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NUREG-1650
Revision 9

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

Office of Nuclear Reactor Regulation

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NUREG-1650
Revision 9

The United States of America Tenth 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 9 to NUREG-1650, “The United States of America Tenth National Report for the Convention on Nuclear Safety,” for submission for peer review at the tenth review meeting of the Convention on Nuclear Safety, to be convened at the International Atomic Energy Agency in Vienna, Austria, in April 2026. This report addresses the safety of land-based commercial nuclear power plants in the United States. It demonstrates how the U.S. Government achieves and maintains a high level of nuclear safety worldwide by enhancing national measures and international cooperation and by meeting the obligations of all the articles established by the Convention. These articles address the safety of existing nuclear installations, the legislative and regulatory framework, the regulatory body, responsibility of the licensee, the priority given to safety, financial and human resources, human factors, quality assurance, assessment and verification of safety, radiation protection, emergency preparedness, siting, design and construction, and operation. This report also addresses the principles of the Vienna Declaration adopted by the contracting parties in February 2015.

Similar to the U.S. National Report issued in 2022, 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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EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) prepared Revision 9 to NUREG-1650, “The United States of America Tenth National Report for the Convention on Nuclear Safety,” for submission for peer review at the tenth 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 joint eighth and ninth review meeting in March 2023. The following U.S. challenges were identified in the International Atomic Energy Agency (IAEA) Convention on Nuclear Safety Joint 8th and 9th Review Meeting – Country Review Report of United States of America, dated March 2023:

- continuing to implement changes in the Reactor Oversight Process, which is discussed in sections 2.2.1 and 6.3.2 of this report
- continuing to prepare for the licensing of new and advanced reactors and fuels, which is discussed in sections 2.3.1, 2.3.2.3, and 2.3.2.4 of this report
- continuing to prepare for the deployment of accident tolerant fuel designs, which is discussed in section 2.3.2.4 of this report
- continuing to focus on international cooperation and support on activities that have the greatest potential for mutual benefit and support embarking countries and countries that are having challenges in fulfilling the CNS obligations, which is discussed in section 2.3.3.5 of this report
- sustaining transformation; in particular, expanding hiring capabilities, optimizing the hybrid work environment, and expanding the use of Be riskSMART, which are discussed in sections 2.3.1.7 and 8.1.6.2 of this report
- determining the scope and timing of hosting a future Integrated Regulatory Review Service mission, which is discussed in section 2.4.2 of this report

Section 2.3.1 of this report discusses the status of safety issues raised in the ninth U.S. National Report, issued August 2022, including advanced reactors; transition to operation of Vogtle Electric Generating Plant, Units 3 and 4; data analytics; licensing and oversight of instrumentation and control digital upgrades; oversight of National Institute of Standards and Technology test reactor fuel event and restart activities; pandemic response; and risk-informed and performance-based regulations. In addition, section 2.3.2 of the report addresses the following safety and regulatory issues that have needed significant attention since 2022:

- Accelerating Deployment of Versatile, Advanced Nuclear for Clean Energy Act (ADVANCE Act)
- use of artificial intelligence
- licensing advanced reactors
- licensing of new/next-generation fuel
- licensing requests for power uprates
- requests to restart nuclear power plants
- subsequent license renewal

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

ΔCDP	change in core damage probability
ABWR	advanced boiling-water reactor
AC	alternating current
ACU	Abilene Christian University
ADAMS	Agencywide Documents Access and Management System (NRC)
ADVANCE Act	Accelerating Deployment of Versatile, Advanced Nuclear for Clean Energy Act
AI	artificial intelligence
ALARA	as low as is reasonably achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
AP	advanced passive
APR	advanced power reactor
ASME	American Society of Mechanical Engineers
ATF	accident tolerant fuel
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
CSC	Convention on Supplementary Compensation
CSV	Continuum Site Visit
CY	calendar year
DG	draft regulatory guide
DHS	U.S. Department of Homeland Security
DI&C	digital instrumentation and control(s)
DOE	U.S. Department of Energy
EIS	environmental impact statement
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
FFD	fitness for duty
FFR	Future Focused Research
FLEX	diverse and flexible coping strategies
FR	*Federal Register*
FY	fiscal year
GE	General Electric

GL	generic letter
GSI	generic safety issue
GTCC	greater-than-Class-C
IAEA	International Atomic Energy Agency
IATA	information assessment team advisory
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
LLW	low-level radioactive waste
LMP	Licensing Modernization Project
LR-ISG	license renewal interim staff guidance
LWR	light-water reactor
MACCS	MELCOR Accident Consequence Code System
MAP	Mission Analytics Portal
MAP-X	Mission Analytics Portal External
MD	management directive
ML	machine learning
MOA	memorandum of agreement
MRP	Materials Reliability Program (EPRI)
MWe	megawatt(s) electric
MWt	megawatt(s) thermal
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
non-LWR	non-light-water reactor
NQA	Nuclear Quality Assurance
NRAN	Nuclear Regulator Apprenticeship Network
NRC	U.S. Nuclear Regulatory Commission
OIG	Office of the Inspector General
OMB	Office of Management and Budget
ONR	Office for Nuclear Regulation (U.K.)
OpESS	Operating Experience Smart Sample
OSART	Operational Safety Assessment Review Team

PRA	probabilistic risk assessment
PWR	pressurized-water reactor
RG	regulatory guide
RIPE	Risk-Informed Process for Evaluations
RIS	regulatory issue summary
RITSTF	Risk-Informed Technical Specification Task Force
RS	review standard
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	subsequent license renewal
SLR-ISG	subsequent license renewal interim staff guidance
SMR	small modular reactor
SRM	staff requirements memorandum
SSHAC	Senior Seismic Hazard Analysis Committee
SSC	structure, system, and component
SSEP	safety related, important to safety, security, and emergency preparedness
SSR	specific safety requirements
STEM	science, technology, engineering and mathematics
Sv	sievert
SWP	strategic workforce planning
TI	temporary instruction
TR	topical report
TRISO	tristructural isotropic
TSTF	Technical Specification Task Force
UCO	uranium oxycarbide
U.K.	United Kingdom
U.S.	United States
VLSSIR	very low safety significance issue resolution process
WANO-AC	World Association of Nuclear Operators–Atlanta Center
WCAP	Westinghouse Commercial Atomic Power
WENRA	Western European Nuclear Regulators' Association

PART 1
INTRODUCTION AND SUMMARY

1 INTRODUCTION

The introduction describes the purpose and structure of the "United States of America Tenth National Report for the Convention on Nuclear Safety" and provides a list 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 tenth 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, agency, or staff) updated the ninth U.S. National Report, principally using agency information that is publicly available. This updated report follows the format of the ninth U.S. National Report published in 2022 and is designed to be a standalone document. Therefore, this report duplicates some of the information presented in the 2022 report. To facilitate peer review, Part 1, Table 1, summarizes 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 2022 report; progress on outstanding challenges and suggestions; safety and regulatory issues that have arisen since the 2022 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 general safety considerations and summarize major safety-related features. Articles 17 through 19 address the safety of installations.

Similar to the 2022 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 a list of nuclear plants in the United States.

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

1.2 Changes from the Ninth U.S. National Report

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

Table 1 Listing of Changes from the Ninth U.S. National Report

Report Section		Change
Abstract		Updated to add discussion about the ninth CNS
Executive Summary		Updated to add discussion about the ninth CNS
PART 1		
Section 1	INTRODUCTION	Updated to add discussion about the ninth CNS
1.1	Purpose and Structure of This Report	Updated to add discussion about the ninth CNS
1.2	Changes from the Ninth U.S. National Report	Updated table
Section 2	SUMMARY	Renumbered and updated to add discussion about the ninth CNS
2.1	The U.S. Policy Toward Nuclear Activities	Updated to include revised NRC's mission
2.1.1	Regulatory Body Organizational Values	No changes
2.1.2	Regulatory Body Challenges	Updated to add discussion on most recent NRC Inspector General report
2.2	National Nuclear Programs	Updated to list new and advanced reactors
2.2.1	Reactor Oversight Process	No changes
2.2.2	License Renewal	Updated references and discussion about subsequent license renewal and units entering the period of extended operation, and updated table listing units with extended operation
2.2.3	Power Uprates	No changes

Report Section		Change
2.2.4	Advanced Reactor Licensing	Modified title and updated to clarify licensing process for new and advanced reactor technologies; updated applications under review; and updated 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 Ninth 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	No changes
2.4.1	Convention on Nuclear Safety	Updated to (1) include discussion on the joint 8th and 9th CNS review meeting 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 include discussion on the CNS peer review challenge for the USA
2.4.3	Operational Safety Review Team	Updated to include references to the results from the last mission
Article 6	EXISTING NUCLEAR INSTALLATIONS	No changes
6.1	Introduction	Updated NRC's mission, added discussion on the most recent NRC budget justification, and included the reference to performance goals results
6.2	Nuclear Installations in the United States	Updated to include status of plants in operation and shutdown and added description of the requests for reauthorization of previously shutdown plants
6.3	Regulatory Processes and Programs	Updated description of the licensing processes and provided updates
6.3.1	Reactor Licensing	Updated to clarify licensing process and include information on applications under review and requests to reauthorize power plants
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 results from the annual report and editorial changes
6.3.4	Operating Experience Program	Section renumbered and editorial changes
6.3.5	Generic Issues Program	Section renumbered and enhanced the program description
6.3.6	Rulemaking	Section renumbered and editorial changes

Report Section		Change
6.3.7	Fire Protection Program	Section renumbered, updated status of plants that implemented the program, and editorial changes
6.3.8	Decommissioning	Section renumbered and updated to discuss recent decommissioning activities
6.3.9	Reactor Safety Research Program	Section renumbered and enhanced the program description
6.3.10	Generic Communications and Orders	Section renumbered and updated to revise types of generic communications and discuss recent examples of generic communications issued
6.4	Vienna Declaration on Nuclear Safety	No changes
Article 7	LEGISLATIVE AND REGULATORY FRAMEWORK	No changes
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7.2	Provisions of the Legislative and Regulatory Framework	No changes
7.2.1	National Safety Requirements and Regulations	Editorial changes only
7.2.2	Licensing of Nuclear Installations	Updated discussion on current regulations and licensing activities
7.2.3	Inspection and Assessment	Editorial changes only
7.2.4	Enforcement	Updated the maximum civil penalty amount
Article 8	REGULATORY BODY	No changes
8.1	The Regulatory Body	No changes
8.1.1	Mandate	Editorial changes only
8.1.2	Authority and Responsibilities	No changes
8.1.2.1	Scope of Authority	Revised to update NRC's mission and editorial changes
8.1.2.2	The NRC as an Independent Regulatory Agency	Updated to clarify selection of the NRC Commission
8.1.3	Structure of the Regulatory Body	No changes
8.1.3.1	The Commission	No changes
8.1.3.2	Component Offices of the Commission	Updated description of roles and responsibilities
8.1.3.3	Offices of the Executive Director for Operations	Updated description of roles and responsibilities
8.1.3.4	Advisory Committees	Updated description of roles and responsibilities
8.1.3.5	Atomic Safety and Licensing Board Panel	Updated description of roles and responsibilities
8.1.3.6	Office of the Inspector General	Updated description of roles and responsibilities
8.1.4	Position of the NRC in the Governmental Structure	No changes
8.1.4.1	Executive Branch	No changes
8.1.4.2	The States (i.e., of the United States)	Updated to describe current state agreement
8.1.4.3	Congress	Updated to reflect the role of the committees
8.1.5	International Responsibilities and Activities	Updated to more accurately reflect NRC participation in international activities
8.1.5.1	International Standards	Updated to describe how NRC regulations consider IAEA safety standards
8.1.5.2	Integrated Regulatory Review Service Mission	Editorial changes only
8.1.5.3	Operational Safety Assessment Review Teams	Editorial changes only

Report Section		Change
8.1.6	Financial and Human Resources	No changes
8.1.6.1	Financial Resources	Updated to include funds for fiscal year 2025
8.1.6.2	Human Resources	Updated throughout
8.1.7	Openness and Transparency	Updated to include most recent numbers associated with public outreach activities and editorial changes
8.2	Independence of the Regulatory Body and Separation of Functions from those Promoting Nuclear Energy	Updated to describe independence of the agency
8.3	Ethics Rules Applying to NRC Employees and Former Employees	No changes
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	Updated program description, reference to most recent NRC policy, and table of enforcement actions
9.3.2	NRC Petition for Enforcement Process	Editorial changes only
9.3.3	Allegation Program	Updated description of the program, provided recent number of allegations, and included references
9.4	Openness and Transparency	Editorial changes only
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
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10.2	Probabilistic Risk Assessment Policy	No changes
10.2.1	Applications of Probabilistic Risk Assessment	Updated to discuss risk-informed initiatives and references
10.2.2	Level 3 Probabilistic Risk Assessment Project	Editorial changes only
10.2.3	Probabilistic Risk Assessment Configuration Control	New section to describe current efforts to develop a program for configuration control
10.3	Safety Culture	No changes
10.3.1	Safety Culture Policy Statement	No changes
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 Monitoring of Licensee Safety Culture	Editorial changes only
10.4	Managing the Safety and Security Interface	No changes
Article 11	FINANCIAL AND HUMAN RESOURCES	No changes
11.1	Financial Resources	No changes
11.1.1	Financial Qualifications for Construction and Operations	No changes

Report Section		Change
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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	Editorial changes only
11.1.2	Financial Assurance for Decommissioning	No changes
11.1.3	Financial Protection Program for Liability Claims Arising from Nuclear Incidents	Updated to indicate next adjustment of financial protection, identified recent insured amounts, updated reference to the most Price-Anderson Report, and described new rulemakings on financial protection
11.1.4	Insurance Program for Onsite Property Damages Arising from Nuclear Incidents	No changes
11.2	Regulatory Requirements for Qualifying, Training, and Retraining Personnel	No changes
11.2.1	Governing Documents and Process	Updated references and editorial changes
11.2.2	Experience	Updated to reflect experience since 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	No changes
12.2.2	Organizational Issues	No changes
12.2.3	Emergency Operating Procedures and Plant Procedures	Editorial changes only
12.2.4	Shift Staffing	Updated information NuScale SMR designs and editorial changes
12.2.5	Human Performance in the Reactor Oversight Process	Updated references
12.2.6	Human Factors Information System	Editorial changes only
12.2.7	Fitness for Duty	Updated to remove description on the impact of COVID-19 on exemptions and editorial changes
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	No changes
12.4.2	Human Performance in Decommissioning Activities	Updated to describe proposed rule and editorial changes
12.4.3	Human Performance Research	Editorial changes only
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
13.3.1	Guidance for Staff Reviews for Licensing	Updated description of the staff guidance

Report Section		Change
13.3.2	Guidance for Design and Construction Activities	Updated references to reflect lates revisions
13.3.3	Guidance for Operational Activities	Updated references
13.4	Quality Assurance Programs	Editorial changes only
13.5	Quality Assurance Audits Performed by Licensees	Editorial changes only
13.5.1	Audits of Vendors and Suppliers	Section removed and information combined with section 13.6
13.6	Vendor Inspection Program	Changes throughout
Article 14	ASSESSMENT AND VERIFICATION OF SAFETY	No changes
14.1	Ensuring Safety Assessments Throughout Plant Life	No changes
14.1.1	Assessment of Safety	No changes
14.1.2	Maintaining the Licensing Basis	Updated description of licensing regulations
14.1.2.1	Governing Documents and Process	Heading removed
14.1.3	Power Uprates	Editorial changes only
14.1.3.1	Governing Documents and Process	Editorial changes and updated to reflect changes since the last reporting cycle
14.1.3.	Experience	Updated discussion on power uprates approved since the last reporting cycle
14.1.4	License Renewal	No changes
14.1.4.1	Governing Rules, Documents, and Process	Updated discussion to include environmental review, reference new rulemaking, and editorial changes
14.1.4.2	Experience	Updated discussion about license renewals approved since the last reporting cycle
14.1.4.3	Operating Beyond 60 Years	Updated discussion on currently licenses being reviewed and new guidance, and described current research programs
14.1.5	The United States and Periodic Safety Reviews	No changes
14.1.5.1	The NRC's Robust and Ongoing Regulatory Process and the Current Licensing Basis	No changes
14.1.5.2	The Backfitting, Forward Fitting, and Issue Finality Processes: Timely Imposition of New Requirements	Editorial changes and updated references
14.1.5.3	License Renewal Confirms Safety of Plants	No changes
14.1.5.4	Risk-Informed Regulation and the Reactor Oversight Process	No changes
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	Editorial changes only
14.2	Verification by Analysis, Surveillance, Testing, and Inspection	No changes
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	Editorial changes only
15.3	Radiation Protection Activities and Control of Radiation Exposure	No changes
15.3.1	Control of Radiation Exposure of Occupational Workers	Updated collective doses measured since the previous reporting cycle
15.3.2	Control of Radiation Exposure of Members of the Public	Editorial changes only
Article 16	EMERGENCY PREPAREDNESS	No changes
16.1	Emergency Plans and Programs	No changes
16.1.1	Background and Overview of Regulatory Requirements	Updated guidance for emergency planning and editorial changes
16.1.2	National Response to an Emergency	No changes
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	Editorial changes only
16.1.2.4	Aspects of Security that Support Response	Editorial changes only
16.1.3	Implementation of Emergency Preparedness Measures	No changes
16.1.3.1	Emergency Classification System and Emergency Action Levels	Editorial changes only
16.1.3.2	Offsite Emergency Planning and Preparedness	Editorial changes only
16.1.3.3	Emergency Preparedness Facilities	Editorial changes only
16.1.3.4	Recommendations for Protective Action in Severe Accidents	Editorial changes only
16.1.4	Emergency Response Exercises	No changes
16.1.5	Regulatory Review and Inspection Practices	Updated to describe performance indicators for emergency preparedness and editorial changes
16.2	Communications Activities	No changes
16.2.1	Communications with Neighboring States and International Arrangements	Updated information on agreements and added discussion on activities since the last review cycle
16.2.2	Communications with the Public	No changes
Article 17	SITING	Editorial changes only
17.1	Background	Editorial changes only
17.2	Safety Elements of Siting	No changes
17.2.1	Background	Updated status references and guidance, and editorial changes
17.2.2	Assessments of Non-Seismic Aspects of Siting	Updated references
17.2.3	Assessments of Seismic and Geological Aspects of Siting	Updated references, described current guidance, and made editorial changes
17.2.4	Assessments of Radiological Consequences from Postulated Accidents	Updated references and editorial changes
17.3	Environmental Protection Elements of Siting	No changes
17.3.1	Governing Documents and Process	Editorial changes only
17.3.2	Other Considerations for Environmental Reviews	Updated description of environmental reviews

Report Section		Change
17.4	Reevaluation of Site-Related Factors	Described current guidance and made editorial changes
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 description and references to inspection programs
18.1.4	Experience and Implementation of Defense-in-Depth Measures	Updated references and editorial changes
18.2	Technologies Proven by Experience or Qualified by Testing or Analysis	Updated experiences with small modular reactors
18.3	Design for Reliable, Stable, and Easily Manageable Operation	No changes
18.3.1	Governing Documents and Process	No changes
18.3.2	Experience	Updated references and editorial changes
18.3.2.1	Human Factors Engineering	No changes
18.3.2.2	Digital Instrumentation and Controls	Updated to reflect current experience, lessons learned, new guidance, and international participation
18.3.2.3	Cybersecurity	Changes throughout
18.4	New Reactor Construction Experience Program	Changes throughout
18.5	Vienna Declaration on Nuclear Safety	No changes
Article 19	OPERATION	No changes
19.1	Initial Authorization to Operate	Updated description of licensing process and status of applications approved
19.2	Definition and Revision of Operational Limits and Conditions	Updated discussion on technical specifications and updated references
19.3	Approved Procedures	No changes
19.4	Procedures for Responding to Anticipated Operational Occurrences and Accidents	Editorial changes only
19.5	Availability of Engineering and Technical Support	Updated to include decommissioning guidance
19.6	Incident Reporting	Updated references and editorial changes
19.7	Programs To Collect and Analyze Operating Experience	Updated discussion on data collection, operating experience and references
19.8	Radioactive Waste	Updated references, waste amounts, and discussion on disposal facilities
19.9	Vienna Declaration on Nuclear Safety	No changes
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

Report Section	Change
APPENDIX B—U.S. COMMERCIAL NUCLEAR POWER REACTORS	Updated to reflect plant license renewals and 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 safety and regulatory issues that have arisen and regulatory accomplishments since the last National Report was issued (see NUREG-1650, Revision 8, "The United States of America Ninth National Report for the Convention on Nuclear Safety," issued August 2022). 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 NRC protects public health and safety and advances the nation's common defense and security by enabling the safe and secure use and deployment of civilian nuclear energy technologies and radioactive materials through efficient and reliable licensing, oversight, and regulation for the benefit of society and 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, predecisional information, and information supplied by foreign governments that is deemed to be sensitive, the NRC makes the documentation that it uses in its decision-making process available in the agency's Public Document Room in Rockville, Maryland, and on the agency's public website 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, issued 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-NRC-25-M-01, "Inspector General's Assessment of the Most Serious Management and Performance Challenges Facing the U.S. Nuclear Regulatory Commission in Fiscal Year 2025," dated October 24, 2024, 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) 2025 management and performance challenges are the following:

- implementing applicable provisions of the Accelerating Deployment of Versatile, Advanced Nuclear for Clean Energy Act of 2024 (ADVANCE Act)

- ensuring safety and security through risk-informed regulation of nuclear technologies and well-supported decisions on the restart of power plants in decommissioning

- overseeing the decommissioning process and the management of decommissioning trust funds

- ensuring the effective protection of information technology and data
- recruiting and retaining a skilled workforce
- overseeing the safe and secure use of nuclear materials and the storage and disposal of waste
- enhancing financial efficiency and resource management
- planning for and assessing the impact of artificial intelligence (AI) on nuclear safety and security programs
- promoting ethical conduct within the agency and protecting regulatory integrity

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 and advanced reactor licensing.

2.2.1 Reactor Oversight Process

The regulatory framework for the NRC's Reactor Oversight Process (ROP) 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. Satisfactory licensee performance in the cornerstones provides reasonable assurance of safe facility operation and accomplishment of the NRC's safety mission.

Inspection reports, including the results of emergency exercise evaluations, are on the NRC public website at https://www.nrc.gov/nrr/oversight/assess/listofrpts_body.html. Article 6 of this report discusses the ROP 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 (SSCs). 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. Accordingly, the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 54, "Requirements for renewal of operating licenses for nuclear power plants," also allow for a renewed operating license to be subsequently renewed for additional terms, provided all applicable license renewal requirements continue to be met. There is no limit on the number of times a license can be subsequently renewed.

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 2025, 97 units in the United States (including those that have since shut down) have had their 40-year operating licenses renewed (initial license renewal for 40–60 years), and 13 units have subsequently renewed their initial renewed operating license for an additional period of operation (subsequent license renewal (SLR) for 60–80 years).

Within the current 2023–2025 period, 19 additional units are expected to enter the period of extended operation, as shown below. Although 10 units with renewed licenses have shut down, a majority of the units are currently operating beyond 40 years.

Table 2 Units that Entered the Period of Extended Operation 2023–2025

Year 2023	Year 2024	Year 2025 (Expected)
• McGuire Nuclear Station, Unit 2 • St. Lucie Plant, Unit 2 • Catawba Nuclear Station, Unit 1 • Catawba Nuclear Station, Unit 2 • LaSalle County Station, Unit 2 • Columbia Generating Station	• Susquehanna Steam Electric Station, Unit 2 • Callaway Plant, Unit 1 • Waterford Steam Electric Station, Unit 3 • Limerick Generating Station, Unit 1 • Byron Station, Unit 1 • Grand Gulf Nuclear Station, Unit 1 • Diablo Canyon Power Plant, Unit 1	• Wolf Creek Generating Station, Unit 1 • Fermi, Unit 2 • Palo Verde Nuclear Generating Station, Unit 1 • River Bend Station, Unit 1 • Millstone Power Station, Unit 3 • Diablo Canyon Power Plant, Unit 2

Section 2.3.2.7 of this report provides an update of the sites that have requested license renewal. Section 14.1.4 discusses the license renewal and SLR process in detail, including the recent rulemaking activity for 10 CFR Part 51, "Environmental protection regulations for domestic licensing and related regulatory functions," and corresponding updates to the generic environmental impact statement (EIS) 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 and Advanced Reactor Licensing

The NRC's new and advanced reactor program focuses on licensing reviews for small and large light-water reactors and non-light-water reactors (non-LWRs), 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 encourages preapplication interactions with advanced reactor developers to provide stability and predictability in the licensing process through early identification and resolution of technical and policy issues that would affect licensing. In March 2024, the NRC staff issued preapplication engagement guidance in appendix A to Interim Staff Guidance (ISG) DANU-ISG-2022-01, "Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications—Roadmap," to enable more predictable and shorter schedules and other efficiencies during the review of an advanced reactor license application. Appendix A also contains guidance for prospective applicants to request a preapplication readiness assessment to assess the readiness of a draft application, such as a design certification, combined license, early site permit, construction permit, or license renewal, before it is formally submitted for the staff's review. The readiness assessment allows the staff to (1) identify any required information that is missing from the application, (2) identify technical or regulatory issues that may complicate the acceptance or technical reviews of the application, and (3) become familiar with the content of the application, particularly in areas where applicants plan to propose new concepts or novel design features.

The NRC staff is also interacting with vendors and utilities on advanced reactor applications and licensing activities. The NRC staff is actively reviewing topical reports associated with potential design certifications, construction permits, and operating license applications. Sections 2.3.1 and 2.3.2.1 of this report discuss current licensing.

In addition to working on domestic issues for advanced reactor licensing, the NRC's advanced reactor program is also actively engaged in several international cooperative activities to promote enhanced safety and awareness for advanced 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.

For example, the NRC was a founding member of the Multinational Design Evaluation Programme, a unique international forum that included 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 Cooperation and Development performed the technical secretariat duties for the Multinational Design Evaluation Programme.

The NRC also chairs the Small Modular Reactor Regulators' Forum, which is an international group that enhances nuclear safety by identifying and resolving common safety issues that may challenge regulatory reviews associated with these advanced reactors. The forum includes representatives from the regulatory authorities of Canada, China, Czech Republic, Finland, France, Japan, the Netherlands, Republic of Korea, Russian Federation, Saudi Arabia, South Africa, and the United Kingdom. The NRC participates in the NEA's Committee on Nuclear Regulatory Activities, which has various working groups focused on licensing advanced reactors, such as the Working Group on New Technologies and the Working Group on Policy and Licensing. The NRC is an observer of the Western European Nuclear Regulators' Association's (WENRA's) Reactor Harmonization Working Group. The NRC also cooperated with the IAEA on its assessment of the applicability of current safety standards to advanced reactors and novel technologies and actively participates in regulatory working groups of the IAEA Nuclear Harmonization and Standardization Initiative, which was established in 2022 to advance harmonization of small modular reactor (SMR) design, construction, regulatory, and industrial approaches.

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

2.3 Safety and Regulatory Issues and Regulatory Accomplishments

This section updates important safety and regulatory issues identified in the ninth 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 Ninth U.S. National Report

In the ninth 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:

- advanced reactors
- transition to operation of Vogtle Electric Generating Plant, Units 3 and 4
- data analytics
- licensing and oversight of digital instrumentation and control (I&C) upgrades
- oversight of National Institute of Standards and Technology (NIST) test reactor fuel event and restart activities
- pandemic response
- accident tolerant fuel
- risk-informed and performance-based regulation

2.3.1.1 Advanced Reactors

Since licensing of advanced reactors remains a regulatory issue, section 2.3.2.3 of this report provides updates and describes the licensing program for advanced reactors.

2.3.1.2 Transition to Operation of Vogtle Electric Generating Plant, Units 3 and 4

The transition to operation of Vogtle Electric Generating Plant is considered a major accomplishment. Section 2.3.3.3 of this report provides the update.

2.3.1.3 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 agency's decision-making process and communications with the public, licensees, and applicants.

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 decision-making, 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 (MAP), and the effort to make data more accessible to external stakeholders through the Mission Analytics Portal External (MAP-X).

<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 aggregated 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 captures information about reactor inspection and licensing activities. The Data Warehouse extracts, transforms, and loads data for developing visualizations outside of the transactional system. All NRC offices and staff are now able to access the Data Warehouse to gather standardized and accurate data. The NRC migrated the Data Warehouse to the Azure cloud in 2023, which decreased maintenance costs and streamlined the usage for data analytics applications.

<u>Mission Analytics Portal</u>. MAP and MAP-X are applications to provide stakeholders with data that enable the NRC to make better and faster regulatory decisions. MAP helps the NRC staff and management quickly access and present mission-related data. It provides critical business analytics to enhance the NRC's ability to make risk informed decisions about how it operates and regulates. MAP provides quicker access to information and 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 decision-making and consistency. The NRC staff can connect to the Data Warehouse to produce dashboards and analytics tools for themselves and produce data and visualizations for external audiences, increasing visibility into NRC data and projects.

The vision for MAP-X is that external stakeholders, including licensees, applicants, and other parties who have business with the NRC, will use it to submit requests for licensing actions, requests for alternatives, and other regulated information in a secure electronic environment. MAP-X is modernizing the way the NRC engages with external stakeholders through the use of technologies that promote openness and transparency while helping the agency become more effective and efficient. MAP-X currently allows licensees to submit licensee event reports (LERs), event notifications, proposed alternatives, and general submissions to the NRC. New modules will continue to be developed based on stakeholder input.

Dashboards. The NRC has developed data analytics tools and dashboards to summarize and highlight trends in the ROP. 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 includes historical data on NRC inspections, licensee events, and budget metrics.

The NRC also continues to use dashboards to quickly meet changing business needs and to replace monthly paper reports with electronic reports, such as public dashboards to provide status for licensing review schedules and resources. Through data analytics and dashboards, the NRC is also improving its openness, efficiency, clarity, and reliability. The agency is giving the public access to currently available information in formats that are easier to understand. Each dashboard gives public access to current information; this website is an example of a dashboard:
https://www.nrc.gov/reactors/new-reactors/smr/licensing-activities/current-licensing-reviews/nuscale-us460.html

These dashboards improve the accessibility of information that previously would have been obtained by manually searching the public website or the NRC's Agencywide Documents Access and Management System (ADAMS).

External Outreach. The NRC continues to organize and lead data science and AI workshops, as discussed in section 2.3.2.2.

2.3.1.4 Licensing and Oversight of Digital I&C Upgrades

The NRC maintains a robust regulatory program for ensuring the safety and security of nuclear facilities protected and operated with both analog and digital instrumentation and control (DI&C) systems. The staff has made significant progress on several key activities that support improved clarity and reliability of the NRC's DI&C regulatory infrastructure, and it continues to engage stakeholders on developing and implementing ongoing improvements. Examples of these activities include the following:

- Staff Requirements Memorandum (SRM)-SECY-22-0076, "Staff Requirements—SECY-22-0076—Expansion of Current Policy on Potential Common-Cause Failures in Digital Instrumentation and Control Systems," issued May 2023, allows the use of risk-informed approaches to justify an appropriate level of defense-in-depth and diversity for DI&C systems with high safety significance.

- Branch Technical Position (BTP) 7-19, Revision 9, “Guidance for Evaluation of Defense in Depth and Diversity to Address Common Cause Failure Due to Latent Design Defects in Digital Safety Systems,” issued May 2024, incorporates the expanded policy in SRM-SECY-22-0076 and provides review guidance for risk-informed defense-in-depth assessments and the use of design techniques or mitigation measures other than diversity.

- Regulatory Guide (RG) 1.152, Revision 4, “Criteria for Programmable Digital Devices in Safety-Related Systems of Nuclear Power Plants,” issued July 2023, endorses, with some exceptions and clarifications, Institute of Electrical and Electronics Engineers (IEEE) Standard 7-4.3.2-2016, “IEEE Standard Criteria for Programmable Digital Devices in Safety Systems of Nuclear Power Generating Stations.”

- RG 1.250, “Dedication of Commercial-Grade Digital Instrumentation and Control Items for Use in Nuclear Power Plants,” issued October 2022, endorses, with clarifications, Nuclear Energy Institute (NEI) 17-06, Revision 1, “Guidance on Using IEC [International Electrotechnical Commission] 61508 Safety Integrity Level (SIL) Certification to Support the Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Related Applications,” and describes an approach that is acceptable to the staff for dedication of commercial-grade digital equipment for use in nuclear power plant safety applications.

- Operating Experience Smart Sample (OpESS) 2023/01, “Digital Instrumentation and Controls,” issued February 2024, supports baseline inspection activities in the area of DI&C modifications, and provides examples of areas where DI&C equipment may be deficient.

As a result of these improvements and previous DI&C regulatory infrastructure, the NRC staff is prepared for the licensing review of major digital upgrades to operating plants and design reviews of advanced reactors with modern DI&C systems. In September 2022, the staff received an application for major digital upgrades for the Limerick Generating Station, Units 1 and 2, protection systems and control room. The staff anticipates that additional operating plants will 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 DI&C systems.

In parallel with these increased licensing activities, the staff will continue to improve the DI&C regulatory infrastructure through extensive engagement with external stakeholders. The staff continues to ensure that any new or revised requirements (1) are performance-based (rather than prescriptive), (2) are technology neutral, (3) apply in the same manner to operating and new reactors, and (4) do not pose an unnecessary impediment to advances in nuclear applications of digital technology.

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

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 shutdown 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 terminated the Alert 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.

As described in Revision 9 of the National Report, the NRC performed several inspections to confirm that the NIST reactor had safely shut down and that the event did not pose a risk to public health and safety and to ensure that the reactor could restart.

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 than previously reported 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 follow-up 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.

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 would need to review before authorizing restart.

During its root cause evaluation, NIST identified, 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 by ensuring proper placement of fuel in the core to provide proper cooling of the fuel. On July 21, 2022, the NRC issued a license amendment to NIST to revise the technical specifications related to fuel element latch verification. This change required NIST to perform both rotational checks and visual inspection following handling of fuel within the reactor vessel and before operation of the reactor.

On August 1, 2022, the NRC issued a supplemental inspection plan to NIST. The supplemental inspections of corrective actions and improvements made by NIST were used by the NRC to inform the decision to authorize restart of the reactor.

On February 1, 2023, the NRC issued a license amendment to NIST approving changes to its safety analysis report to address potential impacts to equipment and changes to the facility radiation sources as a result of some debris remaining in the primary coolant system. On

March 2, 2023, the NRC issued a license amendment to NIST to modify the safety analysis report to describe an alternative fuel management scheme and associated analytic methods.

On March 9, 2023, at the conclusion of the NRC inspection activities and receipt of supplemental information from NIST, the NRC authorized restart of the reactor. The restart decision was informed by numerous activities as described above. The NRC's decision was based on a finding of reasonable assurance that the facility will be operated safely and within its licensing basis. The NRC is continuing its enhanced oversight of the facility in accordance with the supplemental inspection plan.

The NRC provides additional information on the incident, including copies of the event notifications, letters, and inspection reports on its public website at https://www.nrc.gov/reactors/non-power/event-at-nist.html

2.3.1.6 Pandemic Response

As described in Revision 9 of the National Report, the NRC took necessary steps to protect public health and safety in response to the U.S. Department of Health and Human Services declaration of a public health emergency in response to Coronavirus Disease 2019 (COVID-19) on January 31, 2020. This included 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.

The United States officially ended the public health emergency on May 11, 2023, and the NRC staff does not anticipate any further licensing requests related to COVID-19 unless external conditions change. The NRC continues to apply COVID-related efficiencies, lessons learned, and good practices in other areas where surges in licensing requests involve specific topical areas.

2.3.1.7 Risk-Informed and Performance-Based Regulations

The NRC continues to take steps to improve its risk-informed and performance-based regulations and processes.

Resolution of Issues with Very Low Safety Significance. Stemming from the ROP enhancement project, the NRC's Office of Nuclear Reactor Regulation continues to implement the very low safety significance issue resolution (VLSSIR) process following an update to Inspection Manual Chapter (IMC) 0612, "Issue Screening," Appendix B, "Issue Screening Directions," on May 28, 2025. The VLSSIR process is used to discontinue inspection, screening, and evaluation of low-risk issues involving only licensing-basis questions. 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 have greater than very low significance (i.e., not greater than green if the issue was determined to be a finding by use of 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.

In July 2023, the NRC expanded the use of this risk informed concept with the Office of Nuclear Material Safety and Safeguards implementing a VLSSIR process specific to materials, harmonizing, to the extent possible, with the approach applied to reactors.

Risk-Informed Process for Evaluations. The Risk-Informed Process for Evaluations (RIPE) leverages previous risk-informed initiatives to support the evaluation of regulatory issues consistent with the key principles of integrated decision-making in RG 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," issued January 2018. 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.

To implement RIPE, licensees must have adopted (1) Technical Specification Task Force (TSTF)-505, Revision 2, "Provide Risk-Informed Extended Completion Times—RITSTF [Risk-Informed Technical Specifications Task Force] Initiative 4B," dated November 21, 2018, or TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control—RITSTF Initiative 5b," dated March 27, 2009, and (2) 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors," or a RIPE integrated decision-making panel, as documented in NEI 20-XX, "NEI Guidelines for the Implementation of the Risk-Informed Process for Evaluations Integrated Decision-Making Panel," issued August 2020.

RIPE is a voluntary process for licensees, and the NRC continues to evaluate these licensing requests using the streamlined process outlined in the Office of Nuclear Reactor Regulation's Temporary Staff Guidance (TSG)-DORL-2021-01, Revision 4, "Risk Informed Process for Evaluations," dated May 1, 2024.

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 continues to review licensing applications involving the establishment of a risk management approach for certain limiting conditions for operation contained in 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 decision-making 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 regardless of their level of familiarity with risk-informed decision-making 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 risk-informed decision-making approaches, such as probabilistic risk assessment (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 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 decision-making in NUREG/KM-0016, "Be riskSMART: Guidance for Integrating Risk Insights into NRC Decisions," dated March 2021.

In addition to the established processes listed above, NRR Office Instruction LIC-206, Revision 1, "Integrated Risk-Informed Decision-Making for Licensing Reviews," provides key guidance in the implementation of the Be riskSMART framework. LIC-206 establishes the expectation for project managers and reviewers to leverage Integrated Review Teams to ensure that multidisciplinary review teams engage early to develop insights and review synergies, and to maintain cohesion throughout the review with a consolidated safety evaluation. LIC-206 also provides tools and guidance for technical reviewers of all disciplines to expand their use of risk and probabilistic information in their reviews, and to consider it in concert with their traditional deterministic reviews. The NRC staff is working on updating LIC-206, including expanding the use of risk-informed decision-making (RIDM), to offer tools that staff can leverage to support using risk in their work, as well as proposing a graded approach in conducting licensing activity reviews.

To bolster staff awareness, understanding, and engagement with the Be riskSMART framework in NRR, the staff offered four workshops in September 2024 that were individually tailored to (1) management at headquarters and the regions, (2) technical staff, (3) project managers, and (4) mission support corporate staff. The workshops featured examples of how the NRC has applied risk-informed decision-making to accomplish its safety mission and how the agency can build upon those accomplishments to enhance the use of RIDM. Presenters also highlighted and demonstrated tools that staff could leverage to apply risk insights to their work. Approximately 400 staff across multiple offices attended the workshops. Further workshops that expand on the four previous workshops are underway. These workshops will target more specific topics with a nexus to RIDM and Be riskSMART, including case studies of licensing and oversight activities and available tools to identify risk insights. These workshops will build awareness and acceptance of many ways that risk information can improve efficiency and focus our resources.

2.3.2 Current Safety and Regulatory Issues

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

- ADVANCE Act
- use of artificial intelligence
- licensing new and advanced reactors

- licensing new/next-generation fuel

- licensing requests for power uprate

- requests to restart nuclear power plants

- subsequent license renewal

2.3.2.1 ADVANCE Act

The ADVANCE Act of 2024 was passed with bipartisan support and signed into law July 2024. The Act provides the NRC with additional tools and flexibility to improve how the agency accomplishes its mission while preserving the NRC's role as an independent regulator. The NRC is actively addressing the actions and deliverables directed by the Act. The NRC's Executive Director for Operations has designated a dedicated core team with a lead executive to coordinate the agencywide efforts for implementation of the Act.

The NRC is addressing the Act's requirements through the following actions:

- implementing initiatives to achieve efficient, timely, and predictable license application reviews

- establishing an expedited procedure for reviewing qualifying new reactor license applications

- developing a regulatory framework for fusion technology

- implementing changes to how the agency recovers fees from licensees, including establishing a lower hourly rate for advanced reactor applicants and preapplicants

- assessing the licensing review process for new nuclear facilities at former fossil-fuel power plant sites and brownfield sites

- developing strategies and guidance for microreactor licensing

- removing certain limitations on foreign ownership of some types of licensed facilities

- continuing to support international coordination on nuclear technologies and licensing activities

- implementing new requirements relating to nuclear fuel

As required by the Act, the Commission has updated the NRC's mission statement to state: "The NRC protects public health and safety and advances the common defense and security by enabling the safe and secure use and deployment of civilian nuclear energy technologies and radioactive materials through efficient and reliable licensing, oversight, and regulation for the benefit of society and the environment." The NRC staff is preparing guidance that will be implemented to ensure the agency's performance is fully aligned with the revised mission.

The agency has issued several specific tasks for actions required by the Act, as well as actions consistent with the intent of the Act. Project teams, in several cases spanning multiple offices within the agency, have been established and are actively working on each of these tasks. Recognizing that stakeholder engagement is important in this process, the NRC has established a public website (https://www.nrc.gov/about-nrc/governing-laws/advance-act.html) and is using social media to inform and engage external stakeholders during implementation of the ADVANCE Act. The website includes a dashboard that shows the status of the NRC's implementation of the ADVANCE Act.

The NRC is inviting questions, comments, and ideas from interested external parties on its implementation of the ADVANCE Act via the following form on the agency's public website: https://www.nrc.gov/about-nrc/governing-laws/advance-act/contactus.html

Consistent with the Principle of Good Regulation of openness, the NRC has and will continue to host public meetings to provide information and receive feedback on the agency's actions on specific tasks.

2.3.2.2 Use of Artificial Intelligence

AI preparedness is essential to the NRC's mission of ensuring safety and security. The NRC is taking a holistic approach to proactively prepare for how AI may affect regulated activities and how AI can be used to improve the agency's operations. For more details on the NRC's preparedness activities and plans to use AI in enhancing its mission, please visit the agency's public website at https://www.nrc.gov/ai.html

Artificial Intelligence in NRC-Regulated Activities

AI Strategic Plan. In May 2023, the NRC issued NUREG-2261, "Artificial Intelligence Strategic Plan: Fiscal Years 2023–2027," which establishes the vision and goals for the NRC to cultivate an AI-proficient workforce, keep pace with AI technological innovations, and ensure the safe and secure use of AI in NRC-regulated activities. The overall goal of the AI Strategic Plan is to ensure staff readiness to effectively and efficiently review and evaluate the use of AI in NRC-regulated activities.

The strategic plan includes five goals:

(1) Ensure NRC readiness for regulatory decision-making.
(2) Establish an organizational framework to review AI applications.
(3) Strengthen and expand AI partnerships.
(4) Cultivate an AI-proficient workforce.
(5) Pursue use cases to build an AI foundation across the NRC.

As described below, the AI Strategic Plan's success will depend, in part, on early and frequent industry stakeholder engagement on envisioned AI applications and partnering with domestic and international counterparts to gain valuable information to benchmark the agency's AI activities.

External Outreach. The NRC hosts a series of Data Science and AI Regulatory Applications Public Workshops to provide a forum for the NRC, nuclear industry, and stakeholders to discuss the state of knowledge and research activities related to data science and AI and their application in the nuclear industry. At these workshops, the NRC works with internal and external stakeholders to identify the benefits and risks associated with the use of AI in regulatory activities and discusses ongoing and planned projects in the nuclear industry. To date, the NRC has hosted five workshops, and a symposium; more information can be found on the workshop website at https://www.nrc.gov/public-involve/conference-symposia/data-science-ai-reg-workshops.html

International Activities on AI. The NRC is committed to proactive and collaborative engagement with international regulatory counterparts. The agency maintains strong relationships with international regulatory and research organizations to learn from their experiences, share its own best practices, and contribute to global nuclear safety. The NRC, along with other member states, continues to work with the IAEA on several AI technical exchanges and anticipated documents. In March 2024, the NRC hosted an IAEA Technical Meeting on the near-term deployment of AI solutions in the nuclear power industry.

The NRC has several bilateral agreements on AI activities with Canada, France, Germany, and the United Kingdom of Great Britain and Northern Ireland (U.K.). These bilateral relationships include cooperative research and information exchanges on regulatory approaches and best practices in AI.

In September 2024, the Canadian Nuclear Safety Commission (CNSC), the U.K. Office for Nuclear Regulation (ONR), and the NRC jointly published an AI principles paper titled "Considerations for Developing Artificial Intelligence Systems in Nuclear Applications." This paper outlines guiding principles to consider when using AI to ensure the safe and secure operation of nuclear facilities and other nuclear materials. The principles discuss the need to clarify and address the challenges arising from these fast-developing technologies while encouraging the beneficial uses of AI.

The NRC, France's Institute for Radiation Protection and Nuclear Safety, and Germany's Society for Plant and Reactor Safety have an informal trilateral agreement on two research area proposals. The first involves data and AI platforms for nuclear sector applications. The NRC began working on this proposal by combining LER data and Human Actors Information System data. Ultimately, the NRC plans to develop an AI model using natural language processing that would assist with classifying the human factors causal codes for LERs. This is expected to be completed in the first quarter of 2026. The second proposal concerns overarching principles for AI-enabled autonomy, with a proposal to be finalized during 2025.

Research Activities. The NRC is investing in AI research through the Office of Nuclear Regulatory Research's Future Focused Research (FFR) program to explore how AI can support the mission and build foundational knowledge across the agency. Launched in FY 2020, the FFR program serves as an enabler to identify areas where AI could be used to meet the NRC's mission while simultaneously building foundational knowledge. Some FFR projects include (1) using machine learning (ML) to inform inspection planning, (2) characterizing cybersecurity states using AI/ML, and (3) applying a natural language processing model to analyze regulatory documents.

In October 2024, the NRC completed the report "Regulatory Framework Gap Assessment for the Use of Artificial Intelligence in Nuclear Applications" to assess whether regulations, guidance, inspection procedures, or training needs to be updated or created. The NRC began assessing whether the existing regulatory framework applies to AI in NRC-regulated activities. The report indicated that the current regulatory framework is flexible enough to consider AI systems within nuclear facilities; however, regulatory guidance may need to be enhanced in certain areas.

Advancing the Use of Artificial Intelligence at the NRC

Potential AI Use Cases. AI presents an opportunity for the NRC to improve operations and support for its mission. In April 2024, the NRC published SECY-24-0035, “Advancing the Use of Artificial Intelligence at the U.S. Nuclear Regulatory Commission,” dated May 9, 2024, summarizing potential AI applications and the staff’s overall approach to effectively leverage AI at the NRC.

The NRC identified 61 potential use cases across the agency that could drive value and enhance how it meets its mission. From these 61 potential use cases, the NRC identified 36 that align with the capabilities of current AI tools, while the remaining 25 could be addressed using non-AI solutions.

To effectively use AI and ensure long-term success, the NRC will develop a framework to integrate AI into the agency’s operations. This involves two strategic next steps: (1) developing an enterprisewide strategy to advance the use of AI within the agency and (2) investing in foundational tools to support the implementation of several use cases. As the agency begins implementing the use cases, the NRC will prepare and share an inventory of AI use cases with the public and other Government agencies.

Workforce. In August 2024, the NRC’s Office of the Chief Human Capital Officer completed the NRC’s AI Competency Model. The AI Competency Model supports the objectives of the NRC AI Project Plan to assess the NRC’s current AI-related skills and identify skill gaps to ensure that the agency can effectively and efficiently conduct regulatory reviews involving the use of AI. The AI Competency Model assists in the hiring and upskilling of the existing workforce and identifies key skills and competencies needed for AI positions. Development of the competency model ensures that the NRC has the necessary AI talent and supports the objectives of the AI in Government Act of 2020.

Data. AI played a role in transforming the NRC’s internal document database known as ADAMS. The NRC used AI-powered cognitive search technology to create an enhanced ADAMS search engine referred to as ADAMS Content Search, which significantly streamlined search processes for the staff and provided a more modern search interface similar to commonly available public web search tools. This project offered valuable insights into practical uses of available AI tools. The NRC launched a public version of this tool, known as ADAMS Public Search, in January 2025.

AI Governance. In September 2024, the NRC set goals for AI implementation, appointed a Chief AI Officer, and created an AI Governance Board to ensure adequate management of the agency’s use of AI. The NRC also created an AI Steering Committee and AI Community of Practice to promote cross-office coordination and direction to prepare the agency for the future use of AI in NRC-regulated activities.

2.3.2.3 Licensing Advanced Reactors

As discussed in section 2.2.4 of this report, the NRC’s new and advanced reactors licensing program encompasses licensing, oversight, and construction inspection activities for all advanced reactor technologies. This section discusses safety and regulatory issues for both non-light-water and light-water technologies.

Ensuring Regulatory Readiness. In preparation for the review of license applications and regulation of 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," issued 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 issued July 2017. The NRC staff has achieved many of the goals outlined in these documents and continues to make significant progress on the remaining activities to support the licensing of advanced reactors. Many of these activities relate to the requirements of section 103 of the Nuclear Energy Innovation and Modernization Act (NEIMA), which was signed into law on January 14, 2019. The term "advanced nuclear reactor" as defined by NEIMA includes a nuclear fission reactor, including a prototype plant with significant improvements in areas such as 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 non-light-water technologies intended for use as commercial nuclear power plants producing electricity or processing heat and on non-light-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 nuclear reactor licenses. This rulemaking will significantly improve the NRC's readiness for advanced reactor licensing by establishing a transformative, clear, reliable, and flexible framework that can be used for a variety of new 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 would recognize technological advances in reactor design and allow credit in the form of operational flexibilities when a new nuclear 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 (LMP), which is described in NEI 18-04, Revision 1, "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," issued August 2019. The NRC endorsed the LMP methodology as an acceptable approach for reviewing novel non-LWR technologies in RG 1.233, Revision 0, "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," issued June 2020. The LMP 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.

On March 1, 2023, the NRC staff submitted to the Commission the draft proposed 10 CFR Part 53 rulemaking in SECY-23-0021, "Proposed Rule: Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors," requesting approval to publish the proposed rule in the *Federal Register*. On March 4, 2024, the Commission issued SRM-SECY-23-0021, "Staff Requirements—SECY-23-0021—Proposed Rule: Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors," approving, in part, the NRC staff's draft proposed rule with exceptions and clarifications to be addressed before publishing the proposed rule for public comment. The staff implemented the Commission's direction in SRM-SECY-23-0021, and the proposed rule was published on October 31, 2024

(89 FR 86918). The public comment period ended on February 2, 2025. The NRC staff expects to provide the draft final rule package, including key guidance, to the Commission by May 1, 2026. The final rule is expected to be published no later than December 31, 2027, in accordance with NEIMA. The NRC's rulemaking website (https://www.nrc.gov/reading-rm/doc-collections/rulemaking-ruleforum/active/ruledetails.html?id=1108) presents additional details related to the schedule and major rulemaking milestones.

As part of the 10 CFR Part 53 rulemaking, the NRC is creating a transformative security framework that includes physical security, fitness for duty, access authorization, and cybersecurity. This rulemaking will ensure requirements are graded in a manner that is 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. In addition to 10 CFR Part 53 rulemaking, on August 9, 2024, a limited-scope "Alternative Physical Security Requirements for Advanced Reactors" was published in the *Federal Register* for public comment on August 9, 2024 (89 FR 65226). The purpose of this rule is to provide alternatives to specific prescriptive security requirements in the existing regulations if consequence-based criteria are met. The public comment period closed on October 23, 2024. The NRC staff plans to send the final rule to the Commission for approval by September 2025.

In November 2023, the NRC amended its regulations to include new alternative emergency preparedness requirements for SMRs and other new technologies. This final rule, 10 CFR 50.160, "Emergency preparedness for small modular reactors, non-light-water reactors, and non-power production or utilization facilities," acknowledges technological advances and other differences from large light water reactors that are inherent in SMRs and other new technologies. The new rule adopts a scalable plume exposure pathway emergency planning zone approach, and a performance based, risk informed, consequence oriented, and technology inclusive emergency preparedness framework. The NRC concurrently issued RG 1.242, "Performance-Based Emergency Preparedness for Small Modular Reactors, Non-Light-Water Reactors, and Non Power Production or Utilization Facilities," to provide guidance on the implementation of the new requirements.

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 staff initiated a project to develop a framework document for an advanced reactor construction inspection and oversight program.

The NRC is also developing options for a regulatory framework for fusion energy systems, as required by NEIMA. On January 3, 2023, the NRC staff submitted SECY-23-0001, "Options for Licensing and Regulating Fusion Energy Systems," to the Commission. On April 13, 2023, the Commission issued SRM-SECY-23-0001, "Staff Requirements—SECY-23-0001—Options for Licensing and Regulating Fusion Energy Systems," approving the option for a limited-scope rulemaking, which the staff is currently developing, to establish a regulatory framework for fusion machines that augments the NRC's byproduct material framework, which is discussed in 10 CFR Part 30, "Rules of general applicability to domestic licensing of byproduct material." In addition, on December 12, 2024, the NRC staff submitted SECY-24-0085, "Proposed Rule: Regulatory Framework for Fusion Machines," requesting approval to publish in the *Federal*

Register a proposed rule modifying the existing byproduct material frameworks to establish a regulatory framework appropriate for and specific to fusion machines.

The NRC public website (https://www.nrc.gov/materials/fusion.html) offers additional information on fusion energy systems.

Interim Staff Guidance. In March 2024, the NRC issued content of application guidance for advanced reactors in RG 1.253, Revision 0, "Guidance for a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors." Concurrently, the NRC staff issued nine interim staff guidance (ISG) documents to inform the review of technology-inclusive, performance-based advanced reactor license applications. The NRC public website (https://www.nrc.gov/reactors/new-reactors/advanced/modernizing/guidance/advanced-reactor-content-of-application-project.html) presents details of these guidance documents.

Microreactors. The NRC is developing specific strategies associated with licensing of microreactors. These strategies leverage flexibilities in the 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. On January 24, 2024, the staff issued SECY-24-0008, "Micro-Reactor Licensing and Deployment Considerations: Fuel Loading and Operational Testing at a Factory," dated February 8, 2024, to provide the Commission with options for regulating certain aspects of fuel loading and operational testing of commercial factory-fabricated microreactors. This paper also seeks Commission direction on whether a factory-fabricated microreactor that includes "features to preclude criticality" would require a facility operating license or a combined license when loaded with fuel. SRM -SECY-24-0008 was issued on June 17, 2025.

On June 18, 2025, the staff issued SECY-25-0052, "Nth-of-a-Kind Microreactor Licensing and Deployment Considerations," for Commission consideration related to "nth-of-a-kind microreactor licensing and deployment. In this context, "nth-of-a-kind" refers to a microreactor of a standard common design that has been previously approved by the NRC through a design certification, manufacturing license, or an operating license via proceedings under 10 CFR Part 50, "Domestic licensing of production and utilization facilities," or 10 CFR Part 52, "Licenses, certifications, and approvals for nuclear power plants." The proposed licensing strategies are intended to support efficient and predictable licensing of standardized designs and identify potential policy issues, such as standardization of operational programs, alternative environmental reviews, and siting considerations, including use of a graded approach to site characterization for microreactors of a standard design.

Risk-Informed and Performance-Based Guidance for Non-Light-Water Reactors. As an outcome of the LMP, the NRC published RG 1.233. This RG endorsed the methodology described in NEI 18-04, Revision 1, as an acceptable approach for informing the licensing basis and determining the appropriate scope and level of detail for parts of applications for licenses, certifications, and approvals for non-LWRs. NEI 18-04 describes an expanded role for PRA for non-LWRs beyond current 10 CFR Part 52 requirements or Commission policy for potential applications under 10 CFR Part 50.

The advanced non-LWR PRA standard, ASME/ANS RA-S-1.4-2021, “Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants,” was issued in February 2021 by the American Society of Mechanical Engineers (ASME) and the American Nuclear Society (ANS).

On March 21, 2022, the NRC staff endorsed the standard as Trial RG 1.247, “Acceptability of Probabilistic Risk Assessment Results for Non-Light Water Reactor Risk-Informed Activities.” This new guidance describes one acceptable approach for determining whether the acceptability of the PRA used to support a PRA application is sufficient to provide confidence in the results for non-LWRs and risk-informed activities. This RG endorses ASME/ANS RA-S-1.4-2021 and industry guidance on non-LWR PRA peer reviews in NEI 20-09, “Performance of PRA Peer Reviews Using the ASME/ANS Advanced Non-Light Water Reactor Standard.” The NRC staff expects to issue a final version concurrent with the issuance of the 10 CFR Part 53 final rule (by December of 2027, as previously discussed above).

The NRC public website (https://www.nrc.gov/reactors/new-reactors/advanced/modernizing/guidance.html) presents details of these guidance documents.

Licensing Activities. The NRC staff is engaged in preapplication interactions with many 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. The current status of preapplication activities can be found on the NRC public website (https://www.nrc.gov/reactors/new-reactors/advanced/who-were-working-with/pre-application-activities.html).

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 in Idaho Falls, Idaho. This was the first combined license application for a non-LWR 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’s combined license application because it did not provide information on these key topics.

In February 2025, Oklo notified the NRC that it plans to submit a combined license application for its Aurora design at Idaho Falls by the end of 2025. Oklo indicated that it plans to submit the application in two phases and requested a readiness assessment for the first phase. The first phase would include the environmental report, portions of the final safety analysis report containing site information, financial requirements, and applicable offsite emergency plans. The NRC staff conducted a limited scope readiness assessment for the first phase in April 2025 and issued a report to Oklo with its findings on July 7, 2025. Additional details on the application of Oklo can be found on the NRC public website (https://www.nrc.gov/reactors/new-reactors/advanced/who-were-working-with/pre-application-activities/okla-aurora-powerhouse.html).

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 fluoride salt-cooled, high-temperature test reactor in Oak Ridge, Tennessee. The NRC staff accepted the application on November 29, 2021, and issued a construction permit for the Hermes test reactor to Kairos on December 14, 2023. Similarly, Kairos tendered an application for a construction permits for its Hermes 2 test reactors on July 14, 2023. The NRC staff issued a construction permits to Kairos for the Hermes 2 reactor on November 21, 2024. Section 2.3.3.1 of this report provides additional details on Kairos Hermes. Additional details on the application of Kairos Hermes can be found on the NRC public website (https://www.nrc.gov/reactors/new-reactors/advanced/who-were-working-with/pre-application-activities/kairos.html).

On March 28, 2024, US SFR Owner, LLC, a subsidiary of TerraPower, LLC (TerraPower), submitted a construction permit application for the Kemmerer Power Station, Unit 1, a proposed Natrium sodium fast reactor near Kemmerer, Wyoming. The U.S. Department of Energy (DOE) Advanced Reactor Demonstration Program is supporting this project. On May 21, 2024, the NRC staff notified the applicant that the Kemmerer Power Station, Unit 1, construction permit application had been accepted and docketed for review. The estimated completion schedule for this review is 25 months for the safety review and 24 months for the environmental review. On February 26, 2025, the NRC staff issued a letter to inform US SFR Owner, LLC, of the completion of the draft safety evaluation with open items. Additionally, the letter discussed that the NRC staff is targeting completion of the final safety evaluation by June 2026, ahead of the August 2026 date communicated in the original schedule letter. Additional details on the application status of Kemmerer Power Station, Unit 1, can be found on the NRC public website (https://www.nrc.gov/reactors/new-reactors/advanced/who-were-working-with/applicant-projects/terrapower.html).

On March 31, 2025, Long Mott Energy, LLC, a wholly owned subsidiary of the Dow Chemical Company, submitted a construction permit application for the Long Mott Generating Station (LMGS) which would be sited in Calhoun County, Texas. The LMGS project would be used to demonstrate the X-energy high-temperature gas-cooled reactor technology and is supported by the DOE’s Advanced Reactor Demonstration Program. The NRC staff are currently reviewing this application and plan to complete the safety review by November 2026 and the environmental review by June 2026. Additional details on the application status of the LMGS, can be found on the NRC public website (https://www.nrc.gov/reactors/new-reactors/advanced/who-were-working-with/applicant-projects/long-mott.html).

Also, universities are showing growing interest in licensing new nonpower reactors using advanced reactor technologies. For instance, the NRC issued a construction permit for a molten salt (liquid fueled) research reactor at Abilene Christian University (ACU), in Abilene, Texas, on September 16, 2024, and is conducting preapplication activities related to high temperature gas nonpower reactors planned to be located on university campuses. Section 2.3.3.1 of this report provides additional updates on licensing activities for research and test reactors.

Section 2.2.4 of this report describes the licensing process and status of new and advanced reactors in more detail.

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 CNSC, the NRC staff has worked with the CNSC on several cooperative reviews, advanced reactor and SMR 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,” dated June 30, 2021, which documents the results of their collaboration on the Xe 100 design. Since then, the CNSC and NRC staff have issued seven additional joint reports documenting the results of collaborative reviews of specific technical and regulatory issues for designs being considered in both countries. Details of NRC international cooperation activities related to advanced reactors are available on its public website at https://www.nrc.gov/reactors/new-reactors/advanced/who-were-working-with/international-cooperation/nrc-cnsc-moc.html

As discussed in section 2.3.3.5 of this report, this relationship has expanded to include the NRC and CNSC's regulatory counterpart from the U.K.'s ONR. An area of trilateral focus is the GE-Hitachi boiling water reactor X (BWRX)-300 design, which is being evaluated in all three countries.

As discussed in section 2.2.4 of this report, the NRC participated in various international forums, committees, and initiatives related to advanced reactors.

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 non-LWR regulatory and policy challenges.

Advanced Reactors Construction Oversight. On June 6, 2023, the staff issued SECY-23-0048, "Vision for the Nuclear Regulatory Commission's Advanced Reactor Construction Oversight Program," to inform the Commission of progress in the development of a construction oversight program for advanced reactors. The goal of the Advanced Reactor Construction Oversight Program is to provide reasonable assurance that advanced reactor facilities are built and will operate in accordance with their approved designs and licensing bases. The Advanced Reactor Construction Oversight Program will address each aspect of an effective oversight program (i.e., performance monitoring, enforcement, and assessment). Progress on this program continues with the preparation of a second Commission paper detailing the advances to the program after many workshops with external stakeholders. The paper will discuss how the workshop results were used to substantially develop the areas of inspection, issue dispositioning, and assessment for advanced reactor construction oversight. The NRC staff is currently developing a new suite of inspection manual chapters and inspection procedures dedicated to advanced reactor construction in preparation for the next power reactor construction expected to start in 2026.

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

2.3.2.4 Licensing New/Next-Generation Fuel

The U.S. nuclear industry, with the assistance of the DOE, is seeking to develop and deploy new nuclear fuel technologies to meet the growing demand for economical, safe, and clean energy. Some of these fuels represent evolutionary changes to traditional uranium oxide fuels, and others represent a more notable shift to different fuel forms and compositions.

Accident Tolerant Fuel, Increased Enrichment Fuel, and Higher Burnup Fuel. Among the fuels being developed and tested are those expected to enhance the tolerance to severe beyond design basis accidents; permit higher burnup and enrichment; and improve performance and related economics under normal operations. Near-term accident tolerant fuel (ATF) concepts, which the industry is pursuing for deployment by the late 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. 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. The NRC staff are currently in the rulemaking process to support these efforts. The NRC provides the status of this activity on its public website at https://www.nrc.gov/reading-rm/doc-collections/rulemaking-ruleforum/active/ruledetails.html?id=1124

To support licensing activities, along with enhancing and optimizing NRC review, the staff has developed the "Project Plan to Prepare the U.S. Nuclear Regulatory Commission for Efficient and Effective Licensing of Accident Tolerant Fuels," Version 1.0, which describes a new paradigm for fuel licensing. Version 1.2 of the plan, issued 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 its 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. Moreover, in December 2024, the NRC and DOE entered a memorandum of understanding, as required by the ADVANCE Act, to share technical expertise and knowledge regarding advanced nuclear fuel concepts.

Regulatory activity for ATF can be found on the ATF public website (https://www.nrc.gov/reactors/power/atf.html), including the "ATF, Increased Enrichment, and Higher Burnup Roadmap to Readiness," which the NRC staff last revised in July 2023 and developed to provide regulatory certainty of licensing near-term ATF, increased enrichment, and higher burnup technologies.

Advanced Reactor Fuel Qualification. The NRC staff have developed guidance to establish a performance-based fuel qualification assessment framework that would satisfy regulatory requirements. The staff issued NUREG-2246, "Fuel Qualification for Advanced Reactors," in March 2022. The NRC staff contracted with Oak Ridge National Laboratory for NUREG/CR-7299, "Fuel Qualification for Molten Salt Reactors," issued December 2022. The NRC staff contracted with Idaho National Laboratory for NUREG/CR-7305, "Metal Fuel Qualification: Fuel Assessment Using NRC NUREG-2246, 'Fuel Qualification for Advanced Reactors,'" issued August 2023. Additionally, the NRC staff reviewed and approved, with conditions and limitations, a topical report by the Electric Power Research Institute (EPRI), EPRI-AR-1(NP)-A, "Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO)-Coated Particle Fuel Performance," issued November 2020, which documents a performance demonstration of UCO TRISO-coated fuel particles that are anticipated for use in both helium cooled and molten salt cooled high temperature reactors. The staff will continue generically applicable fuel qualification activities and will engage with advanced reactor developers on specific issues related to the qualification of their fuel designs.

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. The NRC public website (https://www.nrc.gov/info-finder/fc/triso-x.html#panel216) shows the current status of this application.

Regulatory information pertaining to the fuel cycle aspects of new fuels, including the status of specific licensing actions, can be found on the New Fuels public website https://www.nrc.gov/materials/fuel-cycle-fac/new-fac-licensing.html

2.3.2.5 Licensing Requests for Power Uprates

The 2022 Inflation Reduction Act production tax credits and other policy measures are encouraging nuclear utilities to reexamine the benefits of power uprates to their existing nuclear power plants. Over 70 percent of sites surveyed recently by the NEI are planning for one or more power uprates with a combined capacity of greater than 3 GWe, as indicated in the Nuclear Energy Institute, "The Future of Nuclear Power: 2024 Update Survey," dated October 2024.

Given the forecast of multiple power uprate applications and consistent with the 2024 ADVANCE Act, the staff is identifying opportunities to make its reviews more efficient. In September 2024, the staff issued "Office of Nuclear Reactor Regulation Preliminary Recommendations on Improving the Power Uprate Application and Review Process" for improving the quality of power uprate applications and for achieving efficiencies in the review process. The key areas in the memorandum are the following:

- Continue engagement with industry. This includes additional workshops to identify technical challenges and solutions for combining power uprate reviews with other changes such as increased enrichment and high burnup fuel.
- Develop a graded approach that will focus reviews on the most safety- and risk-significant portions of the application. The staff will use three categories to grade the expected level of effort commensurate with its safety significance and the degree to which the power uprate affects those systems.
- Evaluate and continue targeted improvements. This includes setting up the infrastructure to use enhanced project management tools, potentially streamlining the environmental review, delegating the signature authority for extended power uprates, and implementing other efficiencies identified as part of the ADVANCE Act.

The staff plans to use a dedicated core team to focus resources and provide predictability and increased capacity for power uprate reviews. The staff issued regulatory issue summary (RIS) 25-02, "Planned Power Uprate-Related Licensing Submittals for All Power Reactor Licensees," requesting licensee plans related to power uprate applications to align resources for reviews. To ensure transparency of regulatory activities related to power uprates, the NRC maintains a public webpage at https://www.nrc.gov/reactors/operating/licensing/power-uprates.html

Sections 2.2.3 and 14.1 of this report provide details on the NRC's power uprate licensing process, including the governing documents, regulatory process, and recent experience.

2.3.2.6 Requests to Restart Nuclear Power Plants

In 2023, the NRC received a first-of-a-kind request from Holtec Decommissioning International, LLC to reauthorize power operations at the Palisades Nuclear Plant, where the previous operator had certified permanent cessation of operations. Two additional licensees have subsequently submitted similar requests.

Once a licensee submits certifications under 10 CFR 50.82(a)(1) confirming that operation has permanently ceased and fuel has been permanently removed from the reactor, then 10 CFR 50.82(a)(2) applies, stating that the 10 CFR Part 50 license no longer authorizes operation of the reactor or emplacement or retention of fuel in the reactor vessel. This means that the plant still has an operating license, but that license is conditioned such that operation is no longer allowed. Current NRC regulations do not specify a particular mechanism for reauthorizing the operation of a nuclear power plant after these certifications have been submitted.

Given that the licensee still possesses a valid operating license, the NRC has found the use of exemptions and license amendment requests to be a viable path to restarting a plant. Exemptions and license amendment requests are means of changing the conditions of an operating license within the existing regulatory framework.

In the case of the Palisades Nuclear Plant, Holtec Decommissioning International, LLC, submitted an exemption request, a license transfer, and associated license amendment requests to restart the plant. The NRC staff issued approval for most of these requests on July 24, 2025, with several other amendments still under review in support of plant operation. The staff is expecting to issue the amendments that are still currently under NRC review by October 2025. The status of this application can be found on the NRC public website (https://www.nrc.gov/info-finder/reactors/pali.html).

The NRC has also received a letter from Constellation Energy Generation stating its intent to request reauthorization of power operations at Three Mile Island Nuclear Station, Unit 1, and change the licensed name to the Crane Clean Energy Center. In November 2024, the NRC received an exemption request for Three Mile Island, Unit 1, to support restarting the plant. Additionally, in January 2025, NextEra Energy Duane Arnold, LLC provided a letter and initial exemptions to support restarting the Duane Arnold Energy Center. These reviews are underway, and additional submittals are expected for those two sites to support reauthorization of power operations.

The NRC has issued IMC 2562, Revision 1, “Light-Water Reactor Inspection Program for Restart of Reactor Facilities Following Permanent Cessation of Power Operations,” dated July 29, 2024. The purpose of the IMC is—

- to establish oversight policies, requirements, and guidance for transitioning from a decommissioning reactor facility to an operational power reactor facility subject to the ROP

- to detail the requirements for the inspection activities and operational plant readiness to provide reasonable assurance for safe operations following reactivation of an operating license

- to ensure that other Federal agencies, State and local governments, Tribal governments, the public, and other applicable stakeholders are engaged and informed

Each plant that considers restart will have its own unique issues depending on the condition of the plant, the amount of decommissioning work that has occurred at the site, and the amount of time since shutdown. One key component is quality assurance. Decommissioning quality assurance programs do not have all the attributes necessary to cover the activities performed to restore a plant to operational status. Implementing a quality assurance plan that appropriately covers these activities early in the process is important.

Also, in 2024, the staff revised existing security inspection procedures to support physical and cybersecurity inspections at restarting reactors. Security inspection procedures used for newly constructed large light water reactors (i.e., inspection procedure series 81000) were updated, with no identifiable associated challenges, to reflect policies, requirements, and guidance consistent with IMC 2562. Revision of these procedures established a durable inspection framework and will be used to confirm that restarting reactors have designed and implemented, or are prepared to implement programs, processes, and performance objectives in accordance with current NRC requirements.

2.3.2.7 Subsequent License Renewal

The NRC's current regulatory framework in 10 CFR Part 54 supports the receipt and review of an SLR application. Specifically, 10 CFR 54.31(d) states that "a renewed license may be subsequently renewed in accordance with all applicable requirements." The standards for SLR 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 SLR applications, the NRC staff developed guidance documents to address the unique aging management needs for an SLR (https://www.nrc.gov/reactors/operating/licensing/renewal/guidance.html#standard). 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.

Using lessons learned from reviewing initial license renewal applications and SLR applications, the NRC staff aims to complete the safety reviews for SLR applications within 12 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 12 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.

In February 2022, the Commission issued a decision stating that further environmental review was required for SLR applications, and it directed the staff to propose a rulemaking plan to revise NUREG-1437, Revision 1, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants" (the LR GEIS), and 10 CFR Part 51 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. In SRM-SECY-22-0024, the Commission approved the staff's proposal on April 5, 2022.

On August 6, 2024, the NRC published a final rule (89 FR 64166) revising its environmental protection regulations in 10 CFR Part 51, along with Revision 2 to the LR GEIS. The final rule updates the potential environmental impacts associated with the renewal of an operating license for a nuclear power plant for up to an additional 20 years for either an initial license renewal or one period of SLR. Revision 2 to NUREG-1437, issued August 2024, provides the technical basis for the final rule. The revised LR GEIS supports the updated list of environmental issues and associated environmental impact findings contained in Table B-1 in Appendix B to Subpart A, "Environmental Effect of Renewing the Operating License of a Nuclear Power Plant," of the revised 10 CFR Part 51 for both initial license renewal and one period of SLR. The final rule was updated with a correction to Appendix B to Subpart A published on August 21, 2024 (89 FR 67522).

In 2025, the staff issued revisions to the Standard Review Plan—SLR and the Generic Aging Lessons Learned Report SLR to incorporate ISG, newly identified information, and lessons learned from the staff's review of the initial SLR submittals. These revisions are available in the NRC public website at
https://www.nrc.gov/reactors/operating/licensing/renewal/slr/guidance.html

The NRC public website provides the current status of SLR applications
https://www.nrc.gov/reactors/operating/licensing/renewal/subsequent-license-renewal.html

Section 14.1.4.3 of this report further describes the licensing process and status of SLRs

2.3.3 Major Regulatory Accomplishments

Since the issuance of the previous U.S. National Report in 2022, the NRC has had many regulatory accomplishments, including the following:

- issuance of construction permits for Kairos Hermes 1 and 2 and ACU
- completion of the NuScale designs reviews
- transition to operation of Vogtle Electric Generating Plant, Units 3 and 4
- establishment of a memorandum of cooperation and trilateral engagement with Canada and the U.K. and development of joint technical reports

2.3.3.1 Issuance of Construction Permits for Kairos Hermes 1 and 2 and Abilene Christian University

Kairos Hermes Construction Permit Applications. In October 2021, Kairos Power LLC (Kairos) submitted a construction permit application for Hermes 1, a nonpower fluoride salt-cooled, high-temperature reactor. The NRC staff issued the construction permit for the Hermes 1 test reactor facility on December 14, 2023. Application documents and information on the review are available on the NRC's public website at https://www.nrc.gov/reactors/non-power/hermes-kairos.html

On July 14, 2023, Kairos submitted a second construction permit application for Hermes 2, a two-unit test reactor facility that would be located on the same site as the Hermes 1 test reactor. The Hermes 2 test reactors would use the same fluoride salt-cooled, high-temperature reactor technology as the Hermes 1 reactor, but would incorporate some additional features, such as intermediate salt loops and a shared turbine generator system. The NRC staff issued a construction permit for Hermes 2 on November 21, 2024. Application documents and information on the review are available on the NRC's public website (https://www.nrc.gov/reactors/non-power/new-facility-licensing/hermes2-kairos.html).

Abilene Christian University Molten Salt Research Reactor Construction Permit Application Review. On August 12, 2022, ACU submitted an application for a construction permit for a molten salt research reactor (less than 1 MWt) to be located on the ACU campus, in Abilene, Texas. On October 20, 2022, ACU supplemented its application to provide additional information on its instrumentation and control design. On March 14, 2024, the NRC staff issued the final environmental assessment. On September 16, 2024, the NRC issued its safety evaluation and the construction permit to ACU. Application documents and information on the ACU review are available on the NRC's public website (https://www.nrc.gov/reactors/non-power/new-facility-licensing/msrr-acu.html).

2.3.3.2 Completion of the NuScale Designs Review

In 2016, NuScale Power, LLC, submitted a design certification application for a 12-module SMR design (US600), with each power module rated for 50 MWe. In August 2020, the NRC approved the design. Subsequently, in January 2023, the NRC certified the 12-module SMR design (10 CFR Part 52, Appendix G, "Design Certification Rule for NuScale"), which became effective February 2023 (88 FR 3287). Sections 6.3.1, 12.2.4, 18.2, 18.3.2, and 19.1 of this report provide additional information regarding the design. The NuScale US600 design is the first SMR that was reviewed, approved, and certified by the NRC. The lessons learned from this review are being implemented for the NuScale US460 standard design approval application currently under review by the NRC and will also be applied to other new reactor applications.

In addition, 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 2.3.1.4 of this report.

In November and December 2022, NuScale submitted a request for a standard design approval application (SDAA) for its NuScale US460 (6-module SMR). NuScale completed their staged submittal of the SDAA in March 2023. Each NuScale Power Module is rated at 77 Mwe. In May 2025, the NRC staff the staff has determined that the US460 SMR design meets the applicable requirements for standard design approval in Subpart E, "Standard Design Approvals," of the 10 CFR Part 52. The NRC issued a final safety evaluation report and standard design approval for this design. The following NRC website provides additional information on the NuScale SDAA review:
https://www.nrc.gov/reactors/new-reactors/advanced/who-were-working-with/applicant-projects/nuscale-us460.html

2.3.3.3 Transition to Operation of Vogtle Electric Generating Plant, Units 3 and 4

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.

On February 10, 2012, the NRC issued the combined licenses for two AP1000 reactors at the Vogtle Electric Generating Plant, Units 3 and 4, in Waynesboro, Georgia. This project used modules made off site 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, placed its final module for the Vogtle Electric Generating Plant, Unit 4, in April 2021.

Southern Nuclear submitted the “all ITAAC [inspections, tests, and analyses] complete” notification for Vogtle Unit 3 on July 29, 2022, and for Vogtle Unit 4 on July 20, 2023. The NRC completed its review of any remaining ITAAC and concluded that all ITAAC have been met. On July 31, 2023, Vogtle Unit 3 entered commercial operation, and Vogtle Unit 4 entered commercial operation on April 29, 2024.

Vogtle Units 3 and 4 are the first newly constructed nuclear units in the United States in more than three decades. These units are the first U.S. deployment of the Westinghouse AP1000 Generation III+ reactor technology.

2.3.3.4 Establishment of a Memorandum of Cooperation and Trilateral Engagement with Canada and United Kingdom and Development of Joint Technical Reports

In August 2019, the NRC and CNSC signed a “Memorandum of Cooperation on Advanced Reactor and Small Modular Reactor Technologies between the United States Nuclear Regulatory Commission and the Canadian Nuclear Safety Commission” to strengthen bilateral exchange of best practices and technology review information.

Subsequently, in September 2022, the NRC and the CNSC signed a charter documenting their partnership on a new Memorandum of Cooperation project related to the GE Hitachi BWRX-300 SMR design, which is expected to be deployed in both countries. The NRC and CNSC collaboration are intended to reduce duplication of licensing review efforts, jointly use third-party verification, identify areas for collaborative verification, share expertise, and leverage analysis performed by each organization. As respective applicants, the Tennessee Valley Authority and Ontario Power Generation are working together on the industry side to share experience and enhance design standardization.

In January 2024, the U.K.’s ONR, along with the country’s Environment Agency, began a two-step Generic Design Assessment of the BWRX-300 design, which is akin to the NRC’s preapplication process. The NRC, CNSC, and ONR recognized the commonalities in technology and licensing review timelines and acknowledged the potential to increase the effectiveness of the nations’ respective regulatory processes. Therefore, on March 12, 2024, the organizations signed a first-of-a-kind trilateral “Memorandum of Cooperation between the Canadian Nuclear Safety Commission (CNSC), the United Kingdom Office for Nuclear Regulation (ONR), and the NRC.” Under the terms of this memorandum of cooperation, the NRC, CNSC, and ONR may consider each other’s experiences, regulatory information, and results when conducting technology assessments for the purposes of feedback to vendors or making recommendations for regulatory decisions.

Examples of representative collaborative projects under the Memorandum of Cooperation can be found on the NRC’s public website https://www.nrc.gov/reactors/new-reactors/advanced/who-were-working-with/international-cooperation/nrc-cnsc-moc/collaboration.html

2.4 International Peer Reviews and Missions

The United States strongly supports international peer reviews and the IAEA's suite of missions, including the CNS peer review activities and the Integrated Regulatory Review Service (IRRS) and Operational Safety 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 2022 U.S. National Report was performed during the joint eighth and ninth CNS review meeting in March 2023. The conclusions from the review were positive.

2.4.1.1 Items Resulting from the Contracting Parties' Peer Review

A review of the questions raised by other contracting parties on the 2022 U.S. National Report identified the following areas of interest:

- emergency preparedness
- radiation protection and dose assessment
- severe accidents
- licensing
- operators
- vendors and quality assurance
- good performance/good practice
- external events
- risk-informed decision-making
- oversight

The NRC's presentation during the 2022 review meeting focused on these topics. INPO, representing the U.S. nuclear industry, also discussed its role in maintaining and improving nuclear safety.

The United States was a member of Country Group 1 during the joint eighth and ninth CNS review meeting, where the group participants concluded that the United States implemented one good practice—the establishment of a safety basis for license renewal up to 80 years. "Good practices" are defined in the IAEA INFCIRC/571, "Guidelines regarding the Review Process under the Convention on Nuclear Safety," Revision 7, 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. The following definition for “area of good performance” was agreed upon by the CNS Officers during the CNS Officers’ Meeting on 24–25 September 2019 and confirmed by the Officers at the CNS Officers’ Meeting on 18–19 July 2022:

> 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 2023, Country Group 1 identified the following areas of good performance by the United States:

- using first-of-a-kind transformation approaches
- strategic use of data analytics
- adapting risk tolerance
- competence and knowledge management initiatives

In 2023, Country Group 1 identified the following challenges for the United States:

- continuing to implement changes in the ROP to improve the effectiveness of the program (discussed in sections 2.2.1 and 6.3.2 of this report)
- continuing to prepare for the licensing of new and advanced reactors (discussed in sections 2.3.1.1 and 2.3.2.3 of this report) and fuels (discussed in section 2.3.2.4 of this report)
- continuing to prepare for the deployment of accident tolerant fuel designs (discussed in section 2.3.2.4 of this report)
- continuing to focus international cooperation and support on activities that have the greatest potential for mutual benefit and support embarking countries and countries that are having challenges in fulfilling the CNS obligations (discussed in section 2.3.3.5 of this report)
- sustaining transformation, in particular, by expanding hiring capabilities, optimizing the hybrid work environment, and expanding the use of Be riskSMART (discussed in sections 2.3.1.7 and 8.1.6.2 of this report)
- determining the scope and timing of hosting a future IRRS mission (discussed in section 2.4.2 of this report)

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

2.4.1.2 Vienna Declaration on Nuclear Safety

Since the accident at the Fukushima Dai-ichi nuclear power plant in Japan 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 sixth 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 meeting the Convention’s objective to prevent accidents and mitigate their radiological consequences, should they occur. The Vienna Declaration on Nuclear Safety was codified in IAEA Information Circular (INFCIRC) 872, dated February 18, 2015.

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:

- <u>Robustness of Defense in Depth</u>. 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 standards for construction and operation. The NRC governs these aspects through its regulations and programs, including, but not limited to, the 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 those associated with 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. 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, natural phenomena that might affect the design or operation of the plant must be appropriately characterized, so that the plant's design basis 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. The 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, the DOE published "Preliminary Report on Operational Guidelines Developed for Use in Emergency Preparedness and Response to a Radiological Dispersal Device Incident" (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 ROP, 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 provides more information on the use of the ROP.

One of the many inspections that the NRC conducts under the ROP 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, and deadlines for licensee implementation are established. Conditions need to be corrected in a manner commensurate with their safety or security significance, which is informed by processes such as the Office of Nuclear Reactor Regulation Office Instruction LIC-504, Revision 6, "Integrated Risk-Informed Decision-Making Process for Emergent Issues," dated September 12, 2023, to evaluate and communicate risk-informed decisions on emergent issues.

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. Section 14.1.5.2 of this report presents 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 the Fukushima accident, which reflects the agency’s regulatory approach of promptly addressing new information when it is discovered and promptly taking appropriate regulatory action.

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 meets the intent of a periodic safety review and that ensure that licensed facilities are meeting regulatory requirements. Sections 8.1.5.2 and 14.1.5 of this report further discuss the results of the IRRS mission and the alternative program that the United States uses in lieu of conducting periodic safety reviews.

<u>The Third Principle of the Vienna Declaration on Nuclear Safety</u>. The NRC’s regulatory requirements and guidance documents undergo systematic reviews and revisions, which are informed by international standards and guidance documents. Built into the process for updating the NRC’s guidance is an examination of applicable technical basis information, including related guidance available in domestic and international consensus standards, IAEA nuclear safety standards and recommendations, and other relevant documents. NRC regulatory guidance, for example, routinely cites or references 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. These practices ensure that the NRC's regulatory guides 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 auspices 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 the NRC's use of IAEA safety standards.

2.4.1.3 Areas of Focus for the Tenth Convention on Nuclear Safety

During the 2023 CNS review meeting, the contracting parties agreed to continue to hold topical sessions during the review meetings. In preparation for the organization meeting held in October 2024, contracting parties were invited to propose recommendations for the topical sessions to be held at the tenth CNS review meeting. In October 2024, the contracting parties agreed that the area of focus for this session was strengthening national regulatory capabilities, taking into account new and innovative technologies and effective nuclear knowledge management.

Strengthening National Regulatory Capabilities. The NRC is giving special attention to retaining, training, and hiring staff with the skills and knowledge necessary to address the areas identified in section 2.3.2 of this report. The NRC continues to use its programs for developing and hiring staff in critical specialties, including partnerships with colleges and universities and programs to ensure that the NRC captures and preserves knowledge to assist with employee development and organizational performance. Section 8.1.6.2 of this report presents more information on NRC regulatory capabilities.

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 the 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.

During the 2023 CNS review meeting, Country Group 1 identified the following challenge for the United States: "The last IRRS mission in the USA was hosted in 2010 and limited to operating reactors. The USA should decide on the scope and timing of hosting an IRRS mission." The NRC is currently evaluating hosting a future IRRS mission.

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 the senior leadership level, to participate in OSART missions around the world.

In August 2020, the Wolf Creek Generating Station hosted an OSART mission. The team identified nine issues, one recommendation, and eight suggestions. The results of the OSART mission are documented in the 2023 IAEA-NSNI/OSART/217/2023, “Report of the Operational Safety Review Team (OSART) Mission to the Wolf Creek Nuclear Power Plant,” which is available on the NRC’s public website. Wolf Creek’s management expressed its commitment to addressing the issues identified and improving the operational safety and reliability of their station. A followup OSART mission was hosted in August 2024.

The next OSART mission in the United States is scheduled to take place in September 2025 at the Salem Nuclear Generating Station in New Jersey.

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, 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 website at https://www.nrc.gov.

6.1 Introduction

The mission of the NRC is to protect public health and safety and advance the common defense and security by enabling the safe and secure use and deployment of civilian nuclear energy technologies and radioactive materials through efficient and reliable licensing, oversight, and regulation for the benefit of society and the environment. The NRC's strategic goals are (1) ensure the safe and secure use of radioactive materials, (2) continue to foster a healthy organization, and (3) inspire stakeholder confidence in the NRC.

The agency achieves its strategic safety goals 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 four performance goals and indicators in this Annual Performance Plan, which are discussed in NUREG-1100, "Congressional Budget Justification: Fiscal Year 2025," Volume 40, issued March 2024. 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 four, the following three indicators are related to commercial nuclear power plants:

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

[1] All references to the abnormal occurrence criteria in this section refer to the criteria approved by the Commission in SRM-SECY-17-0019, "Staff Requirements—SECY-17-0019—Final Revision to Policy Statement on Abnormal Occurrence Reporting Criteria," dated August 24, 2017.

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

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

In FY 2024, 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 FY 2022 and 2023.

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, "2024–2025 Information Digest," Volume 35, issued February 2025. Since the issuance of the 2022 U.S. National Report, the Vogtle Electric Generating Plant, Units 3 and 4, have transitioned to operational status, as mentioned in section 2.3.1.4 of this report. Consequently, there is a total of 94 operating power reactors in the United States.

Appendix A to NUREG-1350 also lists installations in the United States that are under active construction or deferred plant status. Bellefonte Nuclear Plant, Units 1 and 2, is currently in deferred status in accordance with the "Commission's Policy Statement on Deferred Plants," dated October 14, 1987 (52 FR 38077).

In early 2023, the NRC received a first-of-a-kind submittal from a licensee seeking to reauthorize power operations at the Palisades nuclear power plant where the operator had previously certified permanent cessation of operations. Similarly, in 2024, two additional licensees have submitted similar requests. Section 2.3.2.6 of this report contains additional information on the restart of nuclear power plants.

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 has licensed all but two of the 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. Section 2.3.2.3 provides additional details for ongoing reactor licensing reviews.

In 1989, the NRC introduced a single-step licensing process, which is specified in 10 CFR Part 52, and provided direction for issuing a combined license for construction and operation of a new reactor. Two operating nuclear reactor units, Vogtle Units 3 and 4, were licensed under 10 CFR Part 52. In total, the NRC has issued 14 combined licenses since 2012. 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.

As specified in 10 CFR Part 52, the NRC can issue an early site permit to approve a site for a nuclear power plant independent of an application for a combined license or construction permit.

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. Articles 18 and 19 of this report provide more detail about the 10 CFR Part 52 regulations.

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 seven design certifications, and two have since expired. The most recent certification was in January 2023, when the NRC issued the final rule certifying NuScale Power's US600 SMR design. In May 2021, the NRC renewed the design certification for the ABWR.

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, "Staff Requirements—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. The expiration date is being addressed by a current rulemaking effort at the NRC to extend the duration of design certifications.

On November 14, 2024, the Commission approved revising the duration of a design certification from 15 years to 40 years (SRM-COMDAW-24-00001, "Staff Requirements—COMDAW 24 0001—Revising the Duration of Design Certifications"). The NRC published a direct final rule in the Federal Register to extend Design Certification durations from 15 to 40 years. This will go into effect by December 31, 2025, unless significant public opposition arises. The direct final rule would change the expiration date for the AP1000 design from February 27, 2026, to February 27, 2046.

As mentioned in Section 2.3.2.3 of this report, the NRC is working in a rulemaking to create 10 CFR Part 53, which will be a risk informed, technology inclusive regulatory framework for optional use by applicants for new commercial nuclear reactor licenses. Section 7.2.2 of this report describes the licensing of nuclear facilities.

As discussed above, the NRC has also received a first-of-a-kind submittal from a licensee seeking to reauthorize power operations at a nuclear power plant where the operator had previously certified permanent cessation of operations, and subsequently other licensees of similarly situated plants have expressed the same interest. Section 2.3.2.6 of this report discusses this in more detail.

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 ROP, the NRC provides continuous oversight of nuclear power plants 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 ROP 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 ROP 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 website.

The NRC uses the ROP 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 agency 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 ROP Action Matrix. The Agency Action Review Meeting is an integral part of the agency's evaluative process 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 website, in publicly available assessment letters to licensees, and in annual public meetings. Performance information and additional information about the ROP can be accessed at https://www.nrc.gov/reactors/operating/oversight.html

As of July 1, 2025, the Action Matrix assessment of licensee performance at nuclear reactors was as follows:

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

The ROP has developed into a mature oversight program since its inception in 2000, and several countries follow its model. The results of annual ROP self-assessments indicate that the program remains effective. SECY-24-0030, "Reactor Oversight Process Self-Assessment for Calendar Year 2023," dated April 9, 2024, documents the most recent status of the NRC's self-assessment program[2]. However, the NRC recognizes the value of continuous improvement and over the years the agency 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 received Commission approval (SRM-SECY-22-0086, "Staff Requirements—SECY-22-0086—Recommendations for Revising the Reactor Oversight Process Assessment Program," dated March 10, 2023) to change 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 is also updating the significance determination process for emergency preparedness issues, as approved by the Commission in SRM-SECY-22-0089, "Staff Requirements—SECY-22-0089—Recommendation for Enhancing the Emergency Preparedness Significance Determination Process for the Reactor Oversight Process," dated February 9, 2023.

Further, the NRC staff evaluated IMC 0609, Appendix E, Part I, "Baseline Security Significance Determination Process for Power Reactors" to determine whether the Baseline Security Significance Determination Process (BSSDP) can be enhanced or further risk-informed. The staff is actively developing substantive revisions to the BSSDP to address feedback received from internal and external stakeholders.

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," issued 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

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[2] Given the immediate activities required by the ADVANCE Act, in COMSECY-25-0001, "Proposed Suspension of Some Staff Self Assessment Activities for Calendar Years 2024 and 2025," dated January 17, 2025, the NRC staff requested approval to suspend performing portions of the self-assessment activities for calendar years 2024 and 2025. In SRM-COMSECY-25-0001, "Staff Requirements—COMSECY-25-0001—Proposed Suspension of Some Staff Saelf-Assessment Activities for Calendar Years 2024 and 2025," dated April 14, 2025, the Commission approved this request.

operating experience and a measure for trending the core damage risk; a partial check on dominant core damage scenarios predicted by PRAs; and feedback to regulatory activities.

The Accident Sequence Precursor Program supports the NRC's safety and performance objectives, strategies, and goals. The program's objectives include (1) evaluating operating events and trends and advances in science and technology for safety implications to enhance the regulatory framework; (2) assisting in preventing, mitigating, and responding to accidents; (3) assisting 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 and 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 a ΔCDP greater than or equal to 1×10^{-3}.

The latest program results, trend analyses, and insights are documented in the "U.S. Nuclear Regulatory Commission Accident Sequence Precursor Program 2024 Annual Report," issued April 2025. This report provides the results of the Accident Sequence Precursor Program for 2024. In addition, it notes the following key insights for the past 10 years (2015 through 2024):

> The number of precursors identified remain at historical low values. The 74 precursors identified in the past decade is the lowest 10-year period total since the ASP Program's inception. The number of LERs and potential precursors identified also remain at historical low values.

Other insights include the following:

- There are no statistically increasing significant trends in the occurrence rate of all precursors and all precursor subgroups, which indicates that licensee risk management initiatives are effective in maintaining a flat or decreasing risk profile for the industry and that current agency oversight programs and licensing activities remain effective.

- Although there is no statistically significant trend for precursors associated with emergency diesel generator failures, there has been an increase in these precursors in recent years. Specifically, the 16 emergency diesel generator precursors identified in the past 3 years is tied for the most in ASP Program history.

- There are no indications of increasing risk due to the potential of “cumulative impact of risk-informed initiatives.” In addition, no new component failure modes or mechanisms were identified.

- Natural phenomena caused five precursors, with hurricanes and high winds the most frequent causes.

- The most frequent initiating events that resulted in precursors were loss of offsite power and losses of a condenser heat sink. The most frequent structure, system, and component failures observed in precursors were associated with emergency diesel generators and high-pressure coolant injection failures.

- A review of the precursors associated with inspection findings that had a significant impact on the risk of the event were most likely due to inadequate procedures, and ineffective corrective action programs.

6.3.4 Operating Experience Program

The NRC recognizes that the effective use of operating experience is important for the agency’s safety and security mission. Under the current NRC Strategic Plan (NUREG-1614), 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, which is 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 and security mission. 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 and risk 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, and these 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.

As discussed in section 2.3.1.3 of this report, the NRC uses data analytics to assist in the evaluation and communication of information. Visualizations of trends and inputs 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 the 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 this in more detail.

The agency's public website 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 will exit the Generic Issues Program and handle the immediate safety concern within the applicable office's procedures.

The Generic Issues Program consists of three stages: screening, assessment, and regulatory office implementation. The purpose of the screening stage is to evaluate the proposed generic issue against the seven screening criteria to determine if the issue should proceed to the assessment stage or if the issue should exit the Generic Issues Program. The purpose of the assessment stage is to evaluate the proposed generic issue to determine if it merits further regulatory action. The purpose of the regulatory office implementation stage is to develop and perform the appropriate regulatory action to resolve the generic issue in a timely manner.

During the whole time that the safety concern is in the Generic Issues Program, it must meet the following seven criteria to continue in the process:

(1) significantly impacts public health and safety, the common defense and security, or the environment (with respect to radiological health and safety)

(2) applies to two or more facilities, or licensees and certificate holders, or holders of other regulatory approvals

(3) is not already being addressed by other regulatory programs and processes, existing regulations, policies, or guidance

(4) can be resolved by new or revised regulation, policy, or guidance

(5) has a risk or safety significance that can be adequately determined or estimated in a timely manner

(6) is well defined, discrete, and technical

(7) may involve review, analysis, or action by the affected licensees, certificate holders, or holders of other regulatory approvals

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 website at https://www.nrc.gov/about-nrc/regulatory/gen-issues; a history of generic issues appears in NUREG-0933, "Resolution of Generic Safety Issues."

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.

The NRC staff must prepare a rulemaking plan before starting a rulemaking activity that requires Commission approval; this ensures Commission engagement before significant agency resources are expended. The Commission reviews this plan before approving or denying the new rulemaking activity. The Commission can modify an approved rulemaking plan or deny it with additional direction to the staff (e.g., revise and 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. The public, Congress, other Federal agencies, States, local governmental bodies, Indian Tribes, industry, technical societies, and citizen groups are among those the NRC seeks input from during the rulemaking process to understand and address any stakeholder views. 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 periods usually lasting 30 to 60 days, but sometimes 75 to 90 days. The NRC can also extend comment periods after publication, if appropriate. Generally, all rules are issued for public comment; exemptions from this requirement include rules for which delaying their publication to receive comments would be contrary to public interest, unnecessary, or impracticable. The NRC has discretion to afford an opportunity for comment on these exempted 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 applying rulemaking procedures to certain other documents that generically address the public or regulated entities. The NRC has many forms of these nonlegislative rules (sometimes called “guidance”) that the NRC issues according to rulemaking requirements in 10 CFR 2.804, “Notice of proposed rulemaking.” This provides requirements for issuing 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. Interpretive rules or general statements of policy may sometimes use a post-promulgation comment process, in which 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 website, the public can access thousands of documents related to NRC rulemaking actions from May 1996 to the present. The website 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 agency’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 43 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 updated guidance to conduct fire protection team inspections. Inspection Procedure (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 IMC 0609, “Significance Determination Process,” Appendix F, “Fire Protection Significance Determination Process,” dated May 2, 2018.

RG 1.189, Revision 5, “Fire Protection for Nuclear Power Plants,” issued in October 2023, provides licensees with regulatory guidance on fire protection issues.

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 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.

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, “Radiological Criteria for License Termination,” to 10 CFR Part 20, “Standards for protection against radiation.” The NRC reviews and approves license termination plans, conducts inspections, processes license amendments, and monitors the status of decommissioning activities to ensure that radioactive contamination is reduced or stabilized. In addition, the decommissioning process includes several opportunities for public involvement.

In 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 can use the guidance in RG 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning,” issued 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," issued 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 license, 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 licensees of nonelectric utility default to using a discounted prepayment method for decommissioning funding. NUREG-1577, Revision 1, "Standard Review Plan on Power Reactor Licensee Financial Qualifications and Decommissioning Funding Assurance," issued December 2001, and RG 1.159, Revision 2, "Assuring the Availability of Funds for Decommissioning Nuclear Reactors," issued October 2011, present additional guidance on methods for power reactor licensees to provide decommissioning funding assurance.

The NRC has determined that spent fuel can safely remain stored in the spent fuel pools (SFPs) or in dry cask storage facilities until a geologic repository is built and operating. The NRC regulations in 10 CFR Part 50 and 10 CFR Part 72, "Licensing requirements for the independent storage of spent nuclear fuel, high-level radioactive waste, and reactor-related greater than Class C waste," contain requirements to maintain spent fuel integrity. For the majority of U.S. facilities in decommissioning, the result of license termination activities is to functionally shrink the license to cover only the footprint of the remaining dry cask storage facility, thereby releasing the remainder of the site from regulatory control.

The current NRC reactor decommissioning requirements have been implemented safely for many years. As of July 23, 2025, 23 nuclear power and early demonstration reactors, 3 research and test reactors, 9 complex materials facilities, 6 Title II uranium recovery facilities, 22 sites are under long-term care with the DOE and under NRC jurisdiction. Of the 23 power and early demonstration reactors in decommissioning, 7 have elected the SAFSTOR (long-term storage) option (3 of the sites that selected SAFSTOR submitted applications to restart operation: Crane Clean Energy Center, Duane Arnold, and Palisades), and 16 have chosen 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.

The NRC is proposing to amend its regulations for the decommissioning of nuclear power 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 NRC 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 in SECY-18-0055, "Proposed Rule: Regulatory Improvements for Production and Utilization Facilities Transitioning to Decommissioning," dated May 7, 2018, subject to Commission direction and comments, to amend NRC 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 NRC's staff held several public meetings during the public comment period. On January 31, 2024, the staff submitted SECY-24-0011, "Final Rule: Regulatory Improvements for Production and Utilization Facilities Transitioning to Decommissioning," dated February 14, 2024, requesting the Commission review and approval of the draft final rule.

The final rule would amend the NRC's regulations to establish specific regulatory requirements for the decommissioning process, consistent with the reduced radiological risk of decommissioning facilities when compared to operating production and utilization facilities. As the decommissioning process progresses, the radiological risk further decreases. The rulemaking would adopt a graded regulatory approach in several areas commensurate with the reduction in radiological risk at four steps in the decommissioning process:

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

(2) sufficient decay of fuel in the SFP 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

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; (4) develop, maintain, and validate the NRC's suite of safety analysis codes; and (5) 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, when such activities do not compromise the NRC's independent regulatory decision-making. 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 and licensing reviews for small modular and advanced non-LWR designs. In the preapplication phase, the NRC interacts with prospective applicants to address topics that would benefit both the applicant and the staff in preparing for a new reactor licensing action. On October 14, 2008, the Commission issued its "Policy Statement on the Regulation of Advanced Reactors" (73 FR 60612), which 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 including the development of regulatory guidance and consensus codes and standards to address technical issues that it expects will arise during its review of advanced reactor designs. Additionally, independent analyses using safety analysis codes and staff expertise are performed in support of licensing reviews.

6.3.10 Generic Communications and Orders

Generic communications are the NRC's primary method of communicating with the nuclear industry and the public on matters having generic applicability. Generic communications also allow the NRC to communicate and share industry experiences and send information to specific classes of licensees and interested stakeholders.

The types of generic communications are as follows:

- Bulletins are used to request licensee actions, information, or both to address significant matters of safety, security, safeguards, or environmental significance that also have great urgency.

- Generic Letters (GLs) address either an emergent or routine technical issue with generic applicability and deal with risk-significant compliance matters that need prompt licensee attention.

- Regulatory Issue Summaries (RISs) communicate or clarify NRC technical or policy positions on regulatory matters or request voluntary participation for information collection or other activities that will assist the NRC in the performance of its mission.

- Information Notices (INs) transmit a wide range of information, such as operational experience, analytical experience, or other communications related to NRC-regulated activities.

- Information Assessment Team Advisories (IATAs) provide critical, time-sensitive, threat-related information to specified licensees. IATAs may request that recipients take voluntary precautionary or protective actions in response to a potential threat. They are issued whenever the U.S. Attorney General or the Secretary of the U.S. Department of Homeland Security (DHS) makes changes to the National Terrorism Advisory System. The NRC issues IATAs elevating security at licensed facilities in response to these changes. An IATA may also be issued if the Federal Bureau of Investigation issues a Domestic Threat Advisory considered relevant to an NRC licensee.

- Security 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. Recent examples are described below.

On February 7, 2025, the NRC issued RIS 2025-02, "Planned Power Uprate-Related Licensing Submittals for All Power Reactor Licensees," to assist the NRC in determining resource and budget needs with respect to future licensing action submittals pertaining to power uprates anticipated under 10 CFR Part 50 and 10 CFR Part 52, and to assist in planning the technical resources needed to review power uprate-related licensing activities.

On February 10, 2025, the NRC issued IN 2025-01, "Lessons Learned When Implementing ASME Code Case N-752," to inform licensees and permit holders of recently observed inconsistencies between the language in licensee programs during the implementation of Code Case N-752 and the risk-informed methods the NRC approved to be acceptable to satisfy the requirements of 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

On February 21, 2025, the NRC issued (1) SA 2025-01, "Security Advisory for affected Nonpower Reactor Licensees;" (2) SA 2025-02, "Security Advisory for Licensees (NRC and Agreement States) that Possess, Use, Ship, or Receive Risk-Significant Quantities of Radioactive Materials;" and (3) SA 2025-03, "Security Advisory for Affected Operating Power Reactor and Independent Spent Fuel Storage Installation Licensees." These Security Advisories were issued in response to the Secretary of Homeland Security designating the 2025 Presidential Address to a Joint Session of Congress as a National Special Security Event.

On March 28, 2025, the NRC issued IN 2025-02, "Implementation of the Real ID Act of 2005 and Considerations for the U.S. Nuclear Regulatory Commission Nuclear Power Plant Licensees," to remind licensees and applicants of the requirements of the REAL ID Act of 2005 and its applicability to NRC regulations and guidance associated with personnel access authorization programs for nuclear power plants.

On April 9, 2025, the NRC issued RIS 2025-03, "Preparation and Scheduling of Operator Licensing Examinations," to inform addressees that the NRC staff needs updated information on projected site-specific operator licensing examination schedules and on the estimated number of applicants planning to take operator licensing examinations.

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 actions to address lessons-learned from the accident.

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 IAEA 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 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 on the subject of nuclear safety and security:

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

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

- The Convention on the Physical Protection of Nuclear Material, ratified in 1982, mandates standards for the physical protection of nuclear material during international transport.

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

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

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

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

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

- The Convention on Supplementary Compensation for Nuclear Damage, ratified in 2008, establishes a framework obligating the United States and other contracting parties to contribute to an international fund for compensation for "nuclear damage" resulting from a nuclear incident.

7.2 Provisions of the Legislative and Regulatory Framework

7.2.1 National Safety Requirements and Regulations

In addition to the Atomic Energy Act, several statutes (listed in previous U.S. National Reports and briefly described in section 8.1.2.1) have substantial bearing on the Commission's practices and procedures. Furthermore, various U.S. Presidents have issued executive orders and directives related to 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. The NRC has implemented these statutes and executive orders through regulations, policy, 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 the DOE are not subject to NRC licensing under the Atomic Energy Act and the Energy Reorganization Act except where specifically provided by law. Chapter 10, Section 101, of the Atomic Energy Act 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.

Section 189a of the Atomic Energy Act 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 generally comprises one lawyer and two technical members, but a single licensing board member (i.e., the presiding officer) or the Commission may also conduct hearings.

Currently, NRC licensing of nuclear power reactor facilities can take one of the two approaches described below. The NRC is also in the process of revising its regulations to add a new risk-informed, performance-based, and technology-inclusive regulatory framework for the licensing of commercial nuclear power reactors. Sections 2.2.4 and 2.3.2.3 of this report discuss the "Part 53" rulemaking in more detail.

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 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 for approval of sites for one or more nuclear power facilities separate from the filing of an application for construction permits or combined licenses. The basic concept underlying 10 CFR Part 52 is to resolve licensing issues early in the process.

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 these 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.

The license for a nuclear power plant, which typically has 40-year initial terms, may be renewed for periods of 20 years. The NRC provides the licensing process 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 issues quarterly inspection reports for a specific power plant. As discussed in section 6.3.2 of this report, senior agency managers report to the Commission plants that have performance issues during the annual Agency Action Review Meeting. This meeting provides another opportunity to discuss significant events, licensee performance issues, trends, and actions to mitigate recurrences. 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 give the NRC enforcement authority.

Section 161 of The Atomic Energy Act 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 $372,240.

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

8.1 The Regulatory Body

This section explains the NRC's mandate, authority and responsibilities, structure and position in the Government, and its financial and human resources, as well as its international responsibilities and activities, such as those related to international standards and IRRS and OSART missions.

8.1.1 Mandate

As discussed in Article 7, the U.S. Congress created the NRC as an independent regulatory agency in January 1975, with the passage of the Energy Reorganization Act. In giving the NRC an exclusively licensing and 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 protect public health and safety and advance the nation's common defense and security by enabling the safe and secure use and deployment of civilian nuclear energy technologies and radioactive materials through efficient and reliable licensing, oversight, and regulation for the benefit of society and the environment. 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 (LLW), 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. Commissioners serve fixed 5-year terms. Once appointed, Commissioners may only be removed for "inefficiency, neglect of duty, or malfeasance in office." 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 the Chairman, who acts as the official spokesperson and principal executive officer of the agency. Through the 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:

- <u>Office of the Executive Director for Operations</u>. The Executive Director for Operations is the chief operating officer of the Agency 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 is responsible for the NRC's financial management activities as well as agencywide internal controls. The Chief Financial Officer establishes budgeting and financial management policy for the agency and provides advice to the Chairman and the Commission on these matters.

- 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 require 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 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 issues. 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 planning and scheduling Commission business; manages the processes for Commission decision-making, such as publishing Commission decisions in memoranda directing staff actions; monitors staff compliance with pending issues and commitments; processes and controls Commission correspondence; 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 provides centralized services in the areas of acquisition, facilities and security, property management, and administrative services, including support for agency directives, transportation, parking, translation, audiovisual, food services, mail distribution, labor services, furniture, supplies, 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 provides overall management of the agency's human capital planning and human resources planning, policy, and program development. It leads the development of the agency's strategic human capital plan and develops and implements the human resources strategic plan, performance plan, and operating plan consistent with agencywide programmatic goals and objectives. The office assists and advises NRC management in the planning and implementation of human capital goals consistent with agency policies and mission. It establishes accountability for achievement of human capital goals, monitors performance, and provides feedback. It delivers human resources services in support of the NRC's strategic management of human capital. The office plans and 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. It also administers and manages the NRC work life services program, including oversight of the employee assistance program and the headquarters, childcare facility, health unit, and fitness center. The office provides advice and support for the planning, development, implementation, oversight, and evaluation of human resources information systems. It also formulates, justifies, and executes activities for the agency human capital budget and for the human resources office budget.

- 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. It develops NRC policy and guidance for the management of allegations and oversees and implements the allegation management programs. The office develops policy and guidance and implements the Alternative Dispute Resolution program in conjunction with both the allegation and enforcement programs. 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.

- 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 updating of agencywide information resources management policy and consolidates office responses to the Commission, Office of Management and Budget (OMB), and congressional inquiries. It manages the implementation of the Freedom of Information Act and oversees the agency's information collection activities.

- 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 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. Regulatory functions include the safe and secure decommissioning of reactor and 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 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. The office conducts a broad range of regulatory activities in support of the Commission's safety and security strategic goals. These activities encompass 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 provides leadership; plans, recommends, manages and implements programs of nuclear regulatory research; and interfaces with all NRC offices and the Commission on research issues. The office independently proposes improvements to the agency's regulatory research programs and processes to enhance safety, efficiency, and effectiveness based on research results and coordinates research activities with the program offices, as appropriate. Also, it coordinates the development of consensus and voluntary standards for agency use and appoints staff to committees. It assesses the effectiveness of selected NRC programs, including the agency's regulations and regulatory guidance, with regard to risk reduction potential, burden reduction potential, and the appropriate degree to which safety margins exist in the design and operation of licensed facilities. It leads the agency's initiative for cooperative research with the DOE and other Federal agencies, the domestic nuclear industry, U.S. universities, and international 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 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 DHS, the DOE, the intelligence and law enforcement communities, and other agencies. The office develops emergency preparedness policies, regulations, programs, and guidance for both currently licensed nuclear reactors and potential new nuclear reactors. The office provides technical expertise on emergency preparedness issues and interpretations; conducts and directs the NRC program for response to incidents; and is the agency's emergency preparedness and incident response interface with the DHS, FEMA, and other Federal agencies.

- Office of Small Business and Civil Rights supports the NRC mission in protecting people and the environment by enabling the agency to advance equal employment opportunities 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 conduct inspections and execute established policies and assigned programs related to inspections, licensing and construction, allegations, enforcement, emergency and incident response, Agreement State program activities, and government liaison programs for NRC-licensed nuclear facilities. The four regional offices also oversee and inspect decommissioning activities.

8.1.3.4 Advisory Committees

The NRC uses two advisory committees for 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 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, and other matters referred to it by the Commission.

- 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, which 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, Congress authorized the Commission to establish the Atomic Safety and Licensing Board Panel, which is a panel of administrative judges who conduct hearings for the Commission 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 independent, objective audit and investigative oversight of the operations of the NRC and the Defense Nuclear Facilities Safety Board, to protect people and the environment. The Inspector General performs audits and investigations to prevent and detect fraud, waste, abuse, and mismanagement, and promote economy, efficiency, and effectiveness in NRC programs and operations. In addition, it reviews existing and proposed regulations, legislation, and directives and comments, as appropriate, on any significant concerns. 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 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 with the DOE. The following summarizes the agency's relationships with the other identified components of the Federal Government:

- Executive Office of the President. 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. 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 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 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 the DHS.

- U.S. Department of Homeland Security. The 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 the IAEA and other international organizations of the United Nations, as well as the Organisation for Economic Cooperation 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 the 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. The 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 cede 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 for certain byproduct, source, and small quantities of special nuclear materials. Three states have also sent letters of intent to seek agreements with the NRC, with a fourth State seeking to expand its regulatory authority by amending its Agreement. 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, in February 1989, the NRC issued its Policy Statement on Cooperation with States at Commercial Nuclear Power Plants and Other Nuclear Production or Utilization Facilities (54 FR 7530), 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 confirm 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:

- Senate Oversight. In the U.S. Senate, the Committee on the Environment and Public Works has jurisdiction over domestic nuclear regulatory activities. Within the committee, the Subcommittee on Clean Air, 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.

- House Oversight. In the U.S. House of Representatives, the Committee on Energy and Commerce has jurisdiction over domestic nuclear regulatory activities. Within the committee, the Subcommittee on Energy, Climate, & Grid Security and the Subcommittee on Environment, Manufacturing, & Critical Materials have responsibility for oversight of the NRC.

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

8.1.5 International Responsibilities and Activities

The NRC conducts a variety of bilateral and multilateral activities related to statutory mandates, international treaties and conventions, cooperation and assistance, and research. U.S. law or international treaties and conventions mandate several NRC international activities; other activities are discretionary.

The NRC's international engagement is integral to the agency'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 31, 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 and its Amendment, 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 has 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 and its supplementary guidance. 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 the statutorily mandated 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, as amended, 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 300 IAEA and NEA meetings each year, the United States participates in 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, the NRC supports more than 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 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 supporting on a bilateral basis countries seeking to enhance their nuclear regulatory programs. The NRC staff participates in many capacity-building activities to provide newcomer countries with information and experience on building a robust, independent regulatory infrastructure.

Bilateral Relations. The NRC has arrangements to exchange technical information with nuclear regulatory agencies in more than 50 countries, Taiwan, and the European Atomic Energy Community. These arrangements establish the framework for the NRC's communications with international regulatory authorities regarding cooperative engagement to ensure 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 regulatory counterparts for short-term assignments to enhance 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 when 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.

<u>Capacity-Building Programs</u>. Since the early 1990s, the NRC has continued to expand its bilateral and multilateral assistance to countries developing or enhancing regulatory capacity for their nuclear power programs. The NRC initially focused its activities in Central and Eastern Europe, assisting counterpart regulators responsible for the oversight of Soviet-designed reactors as they developed their regulatory infrastructure and programs. Over the past decade, the NRC's reactor-related capacity-building programs expanded to include new reactors, aging management of existing nuclear facilities, and physical protection. The NRC provides technical expertise, training, and technology-neutral information relevant to organizational infrastructure and regulatory programs relating to nuclear power programs. This includes providing 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); licensing (construction permit application review and site application review); and regulatory oversight process (construction and vendor inspection practices, licensing review methodology, and power uprates).

<u>Research Programs</u>. 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 everything from emerging technologies to aging management. The NRC and other nuclear regulatory and safety organizations carry out cooperative projects to meet mutual research needs with greater efficiency.

Taken together, the suite of international activities—treaty implementation, export-import licensing, bilateral and multilateral cooperation, capacity building, and research—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 the 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, new or recently revised NRC RGs contain a section titled "Consideration of International Standards." This section documents the specific IAEA safety Requirements and Safety Guides considered by the RG author(s) as the guide was being developed or updated. The only exceptions since 2023 are four RGs that discuss ASME Code Cases.

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 2 recommendations, 20 suggestions, and 25 good practices. Subsequently, the NRC developed an action plan to address the team's findings and hosted a followup mission in 2014. The 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," was published in 2014.

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 closeout letter to the IAEA that served as the final update on the 2010 and 2014 IRRS missions and gave the final response to all items.

Section 2.4.2 of this report presents additional information on IRRS missions.

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 ROP based on which technical areas the OSART team reviews.

Section 2.4.3 of this report provides information about the last OSART mission hosted in the United States.

8.1.6 Financial and Human Resources

8.1.6.1 Financial Resources

As of October 1, 2024, 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 2025 total budget authority was $944.1 million, including the budget for the Office of the Inspector General.

8.1.6.2 Human Resources

The NRC holds a critical role in supporting America's nuclear energy future and therefore must ensure the agency has the organizational capacity to support incoming work in licensing, oversight, and policy. In 2025 the NRC began development of an Agency Project Management Initiative (APMI) framework and a supporting tool, the NRC Enterprise eXecution and Utilization System (NEXUS), to enable comprehensive, data-driven program management, project management, workload management, and strategic workforce planning.

Agency staff will use the APMI framework and NEXUS application to:

- Capture, integrate, and analyze workforce, workload, and operational data needed to fulfill new and emerging reporting requirements.
- Leverage agency data to inform NRC budget submissions aligned with workload and workforce projections.
- Track projects and activities at a more granular level, aligned to budget line items
- Align programs with strategic objectives and national priorities, such as clean nuclear energy deployment and public safety, ensuring a direct line of sight from mission goals to resource utilization.
- Use advanced analytics for forecasting workforce needs, attrition, and future workforce requirements.

Although these capabilities are still in development, the NRC's future state will enable the agency to deliver forward-looking, data-driven assessments of critical skill needs, emerging workforce challenges, and strategic priorities in support of the agency's regulatory mission, statutory requirements, and executive mandates.

<u>Recruitment and Hiring Process</u>

The NRC places significant emphasis on recruiting individuals with critical skillsets to ensure we have the expertise necessary to address the agency's mission effectively. This commitment includes ensuring we have the staff required to license advanced reactors and other new technologies. By prioritizing the recruitment of highly skilled professionals, the NRC aims to maintain a workforce capable of meeting the evolving challenges within the nuclear sector.

The NRC is implementing the OPM Merit Hiring Plan, a significant overhaul of federal hiring processes, where the central goal is to prioritize merit-based recruitment and selection, aiming to improve efficiency, accountability, and the quality of the federal workforce. Some key entailments include an emphasis on merit and skills-based hiring, shifting from degree requirements and focusing on technical and alternative assessments to evaluate candidates' actual skills, knowledge, and abilities relevant to a position; streamlining the hiring process to reduce the time-to-hire to under 80 days; and to enhance recruitment efforts, with increased targeted outreach.

Training and Development

The NRC strives to maintain a learning environment in which knowledge is continually acquired, shared, and applied to enhance individual, team, or organizational performance. Such an environment 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 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 succession planning process. Additionally, the professional training program fosters career development for staff with self-paced, virtual, and instructor-led courses.

The NRC uses blended learning strategies that combine educational techniques to optimize course delivery. Examples of various educational techniques used at the NRC include classroom instruction, videos, 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.

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

Centralized Qualifications

The NRC has centralized the tracking of all NRC qualification programs into a new document collection, under the Office of the Chief Human Capital Officer (OCHCO) management and oversight. This risk-informed approach ensures collective oversight while still allowing qualification program owners to maintain control of their individual programs.

This shift represents a significant change from how qualifications have previously been managed at the NRC. Centralization enables agencywide visibility to all staff qualified in technical, specialized, and administrative areas, and allows our employees to easily access information about existing programs and plan their career development accordingly.

As part of this initiative, a governance framework was developed that established the appropriate level of standardized required program elements and signature authority, commensurate with the responsibilities of the work the qualified staff person will be performing. The framework also defines roles and responsibilities for the qualification program owners and the methodology used for tracking and maintaining qualifications.

Because many agency qualifications share similar programmatic elements, this uniform structure will allow qualification programs to be systematically reviewed and compared, enabling a clear understanding of the content, rigor, and scope of each program. This standardization will support the identification of deltas—differences in required competencies, knowledge, or experience—between programs. By analyzing these deltas, the agency can streamline efforts to cross-qualify employees by focusing training and development resources only on the specific gaps. This facilitates more efficient reallocation of talent during surge activities, supports succession planning, and enhances workforce resilience by enabling qualified staff to move seamlessly across similar mission areas with minimal delay or redundant training.

To improve the efficiency of the qualification process, staff and supervisors are encouraged to identify opportunities where requirements are redundant due to demonstrated skills or prior work experience. In these situations, granting equivalency waivers for training requirements is acceptable. By accounting for prior expertise, the NRC can better align employees with opportunities that leverage their skills, reduce redundancy, and boost overall efficiency within the qualification process.

Development of Employee Profiles

In addition to tracking qualifications, employee specialties relevant to their qualifications, educational background, and work experience will be tracked. This comprehensive aggregation of information for each employee enables leadership to easily identify individuals with specific skill sets and pinpoint gaps in qualified staff due to attrition.

Employee profiles capture core data and serve as a matching tool, aligning available staff with the specialty and qualification needs of projects and forecasted workloads. This empowers supervisors and project managers to quickly identify staff with the right skills, credentials, and experience to take on upcoming or emergent work.

In an environment where efficiency is crucial for completing reviews and technical specialties are spread across various position titles within the agency, the ability to search across employee profiles and identify matches ensures that the right person, with the right specialties, has the capacity for the upcoming project. This process helps prevent staffing delays and reduces the administrative burden on supervisors who would otherwise need to forage for available staff.

The benefits for supervisors and project managers include:

- Improved workforce planning by understanding who is qualified and their availability.
- Fostering a culture of mobility and opportunity, where employees can be matched with work aligned with their specialties and expertise.

Finally, capturing each employee's work experience provides additional agency-wide benefits in terms of knowledge management. The NRC is re-focusing knowledge transfer activities to fully prepare and support employees for the future anticipated work by emphasizing the importance of knowledge transfer throughout one's career, not just at the end. To support this re-focus, the

NRC is ensuring employees have awareness of and easy access to authoritative data sources including laws, regulations, Staff Requirement Memo's, management directives, office instructions, standard operating procedures, and guidance documents. The development of the employee profiles helps leaders and staff to quickly find subject matter expertise for specific projects, answer project questions, and leverage best practices for similar projects.

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 (NUREG-1614) emphasizes Open Government principles and includes specific strategies for ensuring that the regulatory process, decision-making, 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.

Access to NRC Documents. From its inception, the NRC has made it a priority to maintain a Public Document Room, to assist the public in finding publicly available NRC information. The Public Document Room'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 website at https://www.nrc.gov/reading-rm/adams.html#wba. 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 2024, more than 171,000 public users accessed ADAMS nearly 416,000 times, and users requested more than 2.4 million pages of documents.

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 website hosting high-value datasets; www.regulations.gov, an access portal for all Federal rulemakings; www.USAspending.gov, a website where the NRC reports monthly all its spending on contracts, small purchases, and grants; www.itdashboard.gov, a website 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 website. In 2024, the NRC's website had more than 3 million individual visitors. The website was visited more than 6 million times, and visitors requested pages more than 55.6 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 website 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 most of its Commission meetings over the Internet. More recently, the agency expanded webcasting to other high-interest meetings, conferences, and adjudicatory hearings. These webcasts are available for viewing live 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, X (formerly 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 X 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. As of July 2025, the NRC had more than 19,000 X followers. The agency has sent over 6,000 posts since the launch of the platform.

The agency launched its Facebook page in August 2014. Since that time, its page has gained more than 16,000 followers and 500,000 engagements on more than 2,300 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 website. 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 600 publicly available videos, which have received more than 590,000 views. More than 4,200 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 5,000 photos and graphics to its Flickr account, which have been collectively viewed approximately 12.5 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 it began tracking in March 2021, the agency has contributed more than 700 posts to the LinkedIn page. As of July 2025, the page had more than 46,000 followers.

The NRC launched its newest platform, Instagram, in late 2022. This platform complements the NRC's other social media pages by reaching different demographics than the other platforms. As of July 2025, the page has over 1,600 followers and over 600 published posts.

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 (NUREG/BR0297) in February 2024, redefining meeting categories; describing the NRC's expectations for respectful, civil discussions during public meetings; and reiterating the agency's commitment to transparency, as described in the agency's "Enhancing Participation in NRC Public Meetings" (86 FR 14964; March 19, 2021).

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 website, 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. Since 2023, the agency has conducted fully hybrid RICs to share the proceedings with the widest possible audience.

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 LLW.

The Energy Reorganization Act of 1974 established the NRC as an independent regulatory agency. The Act provides that Commissioners may only be removed for "inefficiency, neglect of duty, or malfeasance in office." 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. 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 that the agency's interests are always represented when its decisions are challenged by others.

The NRC has historically been exempted from the interagency regulatory planning and review process that occurs within the executive branch. Under this long-established process, executive branch agencies are required to submit planned regulatory actions to the OMB for review prior to issuance, and actions deemed to be "significant regulatory actions" cannot be finalized until the review process is complete. On February 18, 2025, President Trump issued Executive Order 14215, "Ensuring Accountability for All Agencies," revoking this exemption for all independent regulatory agencies. The executive order also directs the chairman of independent regulatory agencies to regularly consult with and coordinate policies and priorities with the Executive Office of the President. The executive order states that these provisions are intended to be implemented consistent with applicable law, and the order is not to be construed to impair or affect the authority granted by law to an executive agency. As of this writing the NRC is currently evaluating the executive order.

8.3 <u>Ethics Rules Applying to NRC Employees and Former Employees</u>

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 depend 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.

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 ROP, 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 (1) is 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 ROP.

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 of this report, the ROP 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 of 1946 (preponderance of evidence) in enforcement proceedings. After an apparent violation is identified, it is assessed in accordance with the NRC's Enforcement Policy, dated August 23, 2024. 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:[3]

(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 penalty that the NRC may impose for violations of NRC requirements. The civil penalty will be based on the severity of the violation.

(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 ROP significance determination process (described in Article 6) are assigned the colors green, white, yellow, and red based on increasing risk significance.

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

The NRC may hold a predecisional enforcement conference or a regulatory conference with a licensee before making an enforcement decision if (1) escalated enforcement action (i.e., a Severity Level III or higher notice of violation or a greater-than-green ROP 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 decision-making 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 ROP, civil penalties (and the use of severity levels) are considered for issues that are willful, that have the potential to affect the regulatory process, or that have actual consequences.

Although each severity level may have several associated considerations, the outcome of the assessment process for each violation or problem (absent the exercise of discretion) results in one of three outcomes—no civil penalty, a base civil penalty, or twice the base civil penalty. A base civil penalty has been established in the NRC's Enforcement Policy for each escalated severity level violation and for each type of licensee. Specific Commission approval is required for proposals to impose a civil penalty for a single violation or problem that is greater than three times the Severity Level I civil penalty value for that type of licensee.

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

Providing interested stakeholders with enforcement information is very important to the NRC. Enforcement conferences open to public observation appear in the list of public meetings on the NRC's public website (https://www.nrc.gov/pmns/mtg). The agency may issue a press release for significant enforcement actions. 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 website (https://www.nrc.gov/reading-rm/doc-collections/enforcement/actions/index.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		
	2022	**2023**	**2024**
Notices of violation without civil penalties	8	15	7
Civil penalties	0	0	0
Orders without civil penalties	0	0	0
Total enforcement actions	**8**	**15**	**7**

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 enforcement 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 public, including public meetings with the petitioner and publicly available petition-related documents. The NRC's procedures governing this petition process emphasize timely responses to the petitioner and encourage 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(s) and determines whether requested enforcement action is warranted. If warranted, the Commission may ultimately grant the petitioner's request for action, in whole or in part; take other action that satisfies the concerns raised by the requester; or deny the petitioner's request. If a request is granted, the NRC may modify, suspend, or revoke a license; or take other appropriate enforcement action, to resolve the issue(s) identified in the petition.

9.3.3 Allegation Program

The NRC encourages workers in the nuclear industry to take their concerns directly to their employers. The agency fosters a safety-conscious work environment (SCWE). The

Commission's policy statement, “Freedom of Employees in the Nuclear Industry to Raise Safety Concerns Without Fear of Retaliation,” (61 FR 24336; May 14, 1996), describes a SCWE as a work environment where employees are encouraged to raise safety concerns and where concerns are promptly reviewed, given the proper priority based on their potential safety significance, and appropriately resolved with timely feedback to the originator of the concerns and to other employees as appropriate. These expectations are also communicated through the NRC’s “The Safety Culture Policy Statement” (76 FR 34773; June 14, 2011), safety-conscious work environment guidance documents, and other related regulatory tools. Section 10.3 of this report discusses 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 Management Directive (MD) 8.8, “Management of Allegations,” dated January 25, 2024. 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 email 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 website at https://www.nrc.gov/about-nrc/regulatory/allegations-resp

9.4 Openness and Transparency

The NRC established openness as one of five Principles of Good Regulation. The regulatory processes, decision-making, 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, the 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 website extensively to keep the public informed. NRC requirements are written in a way that allows the agency to perform its day-to-day regulatory oversight and licensing activities openly and transparently. 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 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 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 and 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). 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 <u>Background</u>

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, "Staff Requirements—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 decision-making.

The United States has made progress in developing and using risk information, as described in section 2.3.1.7 of this report.

10.2 Probabilistic Risk Assessment Policy

Three policy statements form the basis of the NRC's current treatment of PRA and the related regulatory safety goals and objectives: the "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," dated August 8, 1985; the "Safety Goals for the Operation of Nuclear Power Plants; Policy Statement; Republication," dated August 21, 1986; and the "Use of Probabilistic Risk Assessment Methods in Nuclear Activities; Final Policy Statement," dated August 16, 1995.

10.2.1 Applications of Probabilistic Risk Assessment

The NRC has developed extensive guidance on the role of PRA in U.S. regulatory programs and applies risk insights gained from PRAs to complement traditional engineering analyses. The increased use of risk information has improved issue-specific safety regulation, and the agency has used risk information to evaluate proposed changes to the current licensing bases for individual plants. The NRC continues to evaluate ways that risk insights can be used to enhance its regulatory framework. 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.

- 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. 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/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," which the NRC later endorsed in RG 1.200, Revision 3, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities," issued December 2020, and ASME/ANS RA-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. The staff uses the results and risk insights from this PRA to inform and verify vulnerabilities, inform input to operational programs, and verify that the Commission's Safety Goals and containment subsidiary goals are met. 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.

To support licensing of non-LWR reactors, the NRC issued RG 1.233, Revision 0, "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," in June 2020. RG 1.233, Revision 0, provides guidance on using a technology-inclusive, risk-informed, and performance-based methodology to inform aspects of the licensing basis and content of applications associated with off-normal operating conditions for non-LWR designs. RG 1.233, Revision 0, endorses the guidance in NEI 18-04, Revision 1, commonly referred to as the Licensing Modernization Project (LMP) methodology. A key feature of the LMP methodology is the use of PRA in a leading role, which is an expansion of the role of PRA in mandatory requirements under 10 CFR Part 52. The LMP methodology uses PRA to inform the selection of the licensing-basis events, safety classifications of SSCs and related special treatments of them, and adequacy of defense in depth. Further, concepts from the LMP methodology have been used to inform the development of a proposed technology-inclusive, risk-informed, and performance-based alternative regulatory framework designated as 10 CFR Part 53. Section 2.3.2.3 of this report provides additional details of 10 CFR Part 53.

10.2.2 Level 3 Probabilistic Risk Assessment Project

A 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, "Staff Requirements—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," issued 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 decision-making 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 pressurized-water reactor (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 plant modifications since 2012 (e.g., design and procedural changes to implement diverse and flexible coping strategies (FLEX).

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 website at https://www.nrc.gov/about-nrc/regulatory/research/level3-pra-project

10.2.3 Probabilistic Risk Assessment Configuration Control

Given the NRC's goal of moving towards more risk-informed and performance-based regulatory programs and the increased reliance on these programs for regulatory and licensee decision-making, the NRC determined a need to ensure licensees were appropriately maintaining their PRA. The PRA Standard, as endorsed by RG 1.200, describes an acceptable approach for initially defining the technical adequacy for an acceptable PRA and provides requirements for PRA configuration control (PCC) to maintain the approved hazard group such that the PRA continues to reflect the as-built, as-operated plant.

A project team was established to develop recommendations for the enhancement of PCC oversight. In 2023 the team developed an action plan, which represented a balanced, performance-based approach. In the near-term, this included the OpESS 2023-02, "Probabilistic Risk Assessment Configuration Control," dated January 2024, which supplements select inspection procedures and provides additional insights regarding PCC for inspections. After conducting several public meetings to gain feedback and insights from the licensees and public on the approach, the NRC staff issued the final OpESS 2023-02 for PRA CC on January 9, 2024. Information gathered through the OpESS will support long-term actions, which are still in development. The project is targeting to have a permanent enhancement of the PCC oversight considered for the 2027 ROP cycle.

10.3 Safety Culture

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

10.3.1 Safety Culture Policy Statement

The NRC's "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 INPO, the 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, 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 ROP. 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 ROP 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 ROP 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 declines, inspection and oversight increase to ensure safe plant operation. The ROP continues to allow licensees to self-diagnose and implement corrective actions for their performance problems before the NRC performs followup inspections.

The ROP applies the safety culture traits and attributes of NUREG-2165 to the inspection and assessment of licensee performance as described in IMC 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 ROP 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 October 31, 2023, 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 ROP 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 Performance column, Column 3, of the ROP 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 Response to Action Matrix Column 3 (Degraded Performance) Inputs,” dated April 1, 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/Repetitive Degraded Cornerstone column, Column 4, of the ROP 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 independently evaluate the licensee’s safety culture through a specific supplemental IP 95003, “Supplemental Inspection Response to Action Matrix Column 4 (Multiple/Repetitive Degraded Cornerstone) Inputs,” dated June 7, 2022, 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 ROP 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 ROP 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 ROP 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 decision-making 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 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 factors in 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 the following:

(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 decision-making 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 agency 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, "Managing the Safey/Security Interface," 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 ROP 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 ROP 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 <u>Financial Resources</u>

The NRC's 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.

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 the 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. Non-electric-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 the 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 asks the 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 has the staff (1) determine whether the proposed holder of the license will remain an electric utility following the direct or indirect transfer; (2) for non-electric-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 review under 10 CFR 50.80, "Transfer of licenses."

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 the following, 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 to a licensee's current method of providing financial assurance made 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 2024, when it renewed the insurance requirements for nuclear facilities until 2065. 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 $500 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 2028.

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 $7.901 million, up to a total of $158.021 million for each reactor. These payments are required if a nuclear incident exhausts the first layer of financial protection, currently

$500 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 ($158.021 million). Based on the number of large commercial nuclear power reactors operating as of October 2024, the nuclear power industry is insured to a maximum per incident dollar level of $15.012 billion under the Price-Anderson framework. As of 2024, the maximum amount of standard retrospective premium for each reactor is $158.021 million per incident (i.e., the $158.021 million maximum deferred premium plus a 5 percent surcharge). With 95 reactors currently participating in the secondary financial protection program, the total financial protection available under the Price-Anderson Act for any one incident is approximately $16.263 billion (i.e., $500 million of primary coverage for the site plus the secondary financial protection program ($158.026 million per reactor times 95 reactors) plus 5 percent additional SFP assessment ($7.901 million per reactor times 95 reactors), which is also the limit on liability. The limit of insurance coverage fluctuates as reactor licensees join or withdraw from the retrospective premium pool. If the second tier is depleted, Congress will determine whether additional disaster relief is required to protect public health and safety. 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. Also, the NRC published the following rulemakings amending the regulations in 10 CFR Part 140 to reflect the revised amounts of financial protection.

- "Inflation Adjustments to the Price-Anderson Act Financial Protection Regulations," dated October 5, 2023 (88 FR 60565)

- "Increase in the Maximum Amount of Primary Nuclear Liability Insurance," dated January 1, 2024 (88 FR 71988)

The public benefits significantly from another feature of the Price-Anderson Act. Specifically, 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 2024, claims for more than 249 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 IAEA Convention on Supplementary Compensation (CSC) for Nuclear Damage, which was developed under the auspices of the IAEA to be the basis for a global nuclear liability regime. The CSC provides an additional amount of liability coverage for nuclear incidents in the United States (currently up to approximately $57 million) resulting in damages in excess of the first-tier amount specified in the CSC. 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, "Operators' licenses." 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, Revision 12, "Operator Licensing Examination Standards for Power Reactors," issued 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, Revision 4, "Qualification and Training of Personnel for Nuclear Power Plants," issued June 2019, contains guidance to implement the regulations.

The NRC continues to endorse the training accreditation process that INPO manages. The NRC 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 periodically observes selected INPO performance-oriented evaluation activities that are relevant to a utility-accredited training program, and the NRC staff observes 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 December 29, 2021, 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, Revision 1, “Training Review Criteria and Procedures,” issued January 1993, to verify compliance with SAT requirements.

11.2.2 Experience

The NRC continually reviews operating experience information (e.g., event reports, inspection reports, reactor scrams, safety system actuations and failures, and forced plant outages) and monitors for trends concerning human performance, decision-making, and training, among other areas. Since the last CNS report was issued in 2022, 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 ROP and how feedback and experience in human factors are 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, Revision 3, Chapter 18, “Human Factors Engineering,” issued December 2016; NUREG-0700, Revision 3, “Human-System Interface Design Review Guidelines,” issued July 2020; and NUREG-0711, Revision 3, “Human Factors Engineering Program Review Model,” issued November 2012. These documents provide guidance for the review of human-system interface issues. The NRC also uses NUREG-1764, Revision 1, “Guidance for the Review of Changes to Human Actions,” issued 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 the NRC's Review Standard (RS)-001, "Review Standard for Extended Power Uprates," issued December 2003. 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.

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, "Issuance of Order to Modify Licenses with Regard to Requirements for Mitigating Strategies for

Beyond-Design-Basis External Events," 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, known as FLEX, 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 that 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.

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 validating time sensitive manual actions documented in Appendix E to NEI 12-06, Revision 4, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," issued December 2016. Section 18.1.4 of this report provides further details of the requirements implemented as a result of the Fukushima lessons learned.

12.2.4 Shift Staffing

In 10 CFR 50.54(m), the NRC establishes minimum onsite staffing requirements for licensed operators and senior operators at nuclear power reactor facilities. Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating prior to January 1, 1979," and Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50 contain the NRC staffing requirements for fire brigades and emergency response personnel.

Current staffing requirements are based on assumptions and operating experience from the operation of large light-water reactors. Also, the staffing requirements in 10 CFR 50.54(m) do not address a situation where three or more units are controlled from a single control room, which has been proposed by some designers of SMRs. 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 January 2023, the NRC certified NuScale's SMR design (10 CFR Part 52, Appendix G), 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 recently approved NuScale US460 design includes control of up to 6 units from a single main control room. The approved staffing plan established adequate staffing levels for NuScale designs “up to 12 units” with a minimum of 3 operators during high workload scenarios. Since the NuScale US460 only has 6 units (operated by 3 operators), the TR was sufficient as written and was considered applicable.

The NRC has conducted preapplication activities related to shift staffing with other SMR 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 ROP focuses on safety cornerstones that are assessed through a combination of performance indicators and risk-informed inspections. Section 6.3.2 of this report discusses the ROP and its seven safety cornerstones. In addition to the safety cornerstones, the ROP features three cross-cutting elements that affect the cornerstones: human performance, safety-conscious work environment, and problem identification and resolution.

Human factors experts participate in ROP 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 ROP 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 IMC 0305, “Operating Reactor Assessment Program,” in 2018. IP 95003 was last updated in 2022.

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 ROP 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 disseminates information on human performance issues at individual nuclear power plant sites through its public website: https://www.nrc.gov/reactors/operating/ops-experience/human-factors. Although new reports have not been added since 2011, the staff maintains the website as a historical reference and is working on modernizing the system.

12.2.7 Fitness for Duty

In 10 CFR Part 26, "Fitness for duty programs," the NRC requires each power reactor licensee to implement a fitness-for-duty program for all personnel who have unescorted access to the protected area of its plant or who perform the duties specified in 10 CFR 26.4, "FFD program applicability to categories of individuals." This rule also requires licensees and permit holders authorized to construct a nuclear power plant to implement a fitness-for-duty program for personnel performing certain construction, management, security, and quality control activities. All fitness-for-duty programs must meet the following performance objectives:

- 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," issued March 2009, presents guidance for implementing 10 CFR 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.

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, "Changes, tests and experiments." 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, "Corrective Action."

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 replaced 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.

12.4.2 Human Performance in Decommissioning Activities

As discussed in Section 6.3.8 of this report, on November 3, 2021, the Commission approved publication of a proposed rule on decommissioning activities. The proposed rule, "Regulatory Improvements for Production and Utilization Facilities Transitioning to Decommissioning," was published in the *Federal Register* for public comment on March 3, 2022 (87 FR 12254). The public comment period closed 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, Draft Regulatory Guide (DG)-1347 (proposed Revision 2 to RG 1.184, "Decommissioning of Nuclear

Power Reactors," issued May 2018), includes specific criteria for certified fuel handler training programs to ensure the safe conduct of decommissioning activities, safe handling and storage of spent fuel, appropriate response to plant emergencies, and command and control over these functions.

12.4.3 Human Performance Research

The Human Performance Research Program generates, collects, and evaluates data on human performance for use in human reliability analysis models. The staff evaluates information to gain insights supporting risk-informed regulation and to find human performance data for human reliability analysis. In collaboration with industry, the NRC developed the Scenario Authoring, Characterization, and Debriefing Application database to collect licensed operator performance information in simulator training and job performance measures 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 program and performs onsite inspections to verify aspects of the program implementation.

13.2 Regulatory Policy and Requirements

The NRC establishes the requirements for the design, construction, and operation of commercial nuclear power plants in 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 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, "Contents of applications; technical information," 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 non-safety-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 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, Revision 1, “Quality Assurance Program Description—Design Certification, Early Site Permit and New License Applicants,” issued August 2015, provides guidance to the NRC staff for the review of applications for design certification, construction permits, early site permit, 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 RGs related to quality assurance and apply the industry’s quality assurance codes and standards. The review guidance in NUREG-0800 also includes quality controls to be applied for the non-safety-related SSCs that are identified as being significant contributors to plant safety.

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, Revision 6, “Quality Assurance Program Criteria (Design and Construction),” in September 2023, to endorse NQA-1-2017, NQA-1-2019, and NQA-1-2022.

13.3.3 Guidance for Operational Activities

The NRC has conditionally endorsed the consensus standard ANSI N18.7-1976, “Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants,” issued February 1976, in RG 1.33, Revision 2, “Quality Assurance Program Requirements (Operation),” issued 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 ASME NQA-1 contains information on quality assurance of design and construction.

13.4 Quality Assurance Programs

The NRC inspects quality assurance programs under the ROP for operating reactors and under the Construction ROP (see Article 18 of this report) for new reactors.

The baseline inspection program of the ROP 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 non-safety-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 non-safety-related, equipment is called "augmented quality control."

The Construction ROP 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 February 9, 2022, to verify that the holder of a combined license or a limited work authorization has developed quality assurance procedures, instructions, and other documents that are consistent with the licensee's NRC-approved quality assurance program description and to verify that the licensee has effectively implemented its quality assurance program implementing documents during construction activities.

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

13.5 Quality Assurance Audits Performed by Licensees

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

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' own technical and quality assurance staff perform these activities. 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 verifies that reactor applicants and licensees are fulfilling their regulatory obligations with respect to providing effective oversight of the nuclear supply chain. Also, applicants and licensees are responsible for securing a reliable nuclear supply chain. 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." IMC 2507, "Vendor Inspections," dated May 18, 2023, contains guidance for these inspections. The NRC inspectors use IP 43002, "Routine Inspections of Nuclear Vendors," issued February 2023, to verify that vendors supplying basic components have implemented effective QA procedures, policies, instructions, and plans in accordance with a QA program that complies with the requirements of Appendix B to 10 CFR Part 50. The NRC inspectors use IP 36100, "Inspection of 10 CFR Part 21 and Programs for Reporting Defects and Noncompliance," issued February 2023, to verify that applicants and holders of a certified design, ESP, or applicants and holders of a COL under 10 CFR Part 52, as well as licensees under 10 CFR Part 50, and suppliers of basic components have established a program to effectively implement the requirements of 10 CFR Part 21. The NRC inspectors use IP 35017, "Quality Assurance Implementation Inspection," dated December 10, 2020, to verify that 10 CFR Part 52 applicants with an approved QA program have appropriately translated the QA program into implementing procedures.

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 ROP is the NRC's program to inspect, measure, and assess the safety and security performance of commercial nuclear power plants. The ROP monitors reactor licensee performance in three key areas: (1) reactor safety, (2) radiation safety, and (3) safeguards. The

ROP 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 ROP 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 ROP and results of the regulatory assessment in greater detail.

The Construction ROP monitors and assesses the construction of commercial nuclear power plants in a manner like that used by the ROP. 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, construction permit, or early site permit."

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 contain requirements for the processes by which, under certain conditions, licensees may change their facilities and procedures as described in the final safety analysis report, as updated, without prior NRC approval. In addition to 10 CFR 50.54 and 10 CFR 50.59, there are several other complementary processes for controlling activities that affect other aspects of the licensing basis: 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, "Codes and standards"; 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors"; 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. The following paragraphs provide additional details for each one of these processes.

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 updates of the final safety analysis report.

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.

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," issued July 2016, describes a method the NRC staff considers acceptable for implementing the requirements of 10 CFR 50.54(q).

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 through appropriate engineering and technical evaluations that a proposed activity is safe and effective, 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, “Guidance for Implementation of 10 CFR 50.59, ‘Changes, Tests, and experiments,’” issued June 2021, endorsed, with clarifications, industry guidance document NEI 96-07, Revision 1, “Guidelines for 10 CFR 50.59 Implementation,” issued November 2000, which provides a method that is acceptable to the NRC staff for complying with the provisions of 10 CFR 50.59.

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

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. This is referred to as “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." 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.

As mentioned in section 12.2.1 of this report, the NRC uses RS-001 for evaluating extended power uprates and stretch power uprates. RS-001 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 12 months of accepting the application for review for extended power uprates, within 9 months of acceptance for stretch power uprates, and within 6 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 172 uprates, resulting in a gain of approximately 24,146 MWt or 8,051 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. The NRC staff has held preapplication engagements with several licensees regarding potential power uprates. This includes eight potential measurement uncertainty recapture uprates, one stretch power uprate, and two extended power uprates. These numbers are by reactor units and information about power uprates on its public website at https://www.nrc.gov/reactors/operating/licensing/power-uprates.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 limits commercial power reactor licenses to 40 years but allows that such licenses can 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 is typically 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 expects to complete its review of a license renewal application within 12 months of acceptance of the application if an adjudicatory hearing is not conducted. If there is a hearing, then the schedule for 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, 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 submit 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 EIS.

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 EIS. If there are no 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 issues that the person seeks to have litigated. The Commission will grant the hearing request if it finds that the person has standing (that is, the person 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, Revision 2, "Standard Format and Content for Applications To Renew Nuclear Power Plant Operating Licenses," issued April 2020, which guides applicants preparing an application for a renewed license, and NUREG-1800, Revision 2, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," issued December 2010, which guides the staff in reviewing applications. The standard review plan for license renewal incorporates by reference NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," issued December 2010, which generically documents the basis for determining when existing programs are adequate for license renewal and when they should be augmented. As lessons are learned from the review of renewal applications or generic technical issues are resolved, the NRC issues license renewal interim staff guidance (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 in the conclusion that certain environmental issues can be assessed generically and do not need to be reevaluated in each plant-specific review. Table B-1 of 10 CFR Part 51, 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)," lists these issues. 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 decision-makers. 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, Supplement 1, Revision 1, "Preparation of Environmental Reports for Nuclear Power Plant License Renewal Applications," issued June 2013, provides guidance to applicants preparing environmental reports to be included as part of license renewal applications. NUREG-1555, "Standard Review Plans for Environmental Reviews for Nuclear Power Plants," Supplement 1, Revision 1, "Operating License Renewal," issued June 2013, 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 SLR term (i.e., 60–80 years). No SLRs 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," issued September 2014. The generic impact determinations for the continued storage of spent fuel in NUREG-2157 shall be deemed incorporated into the supplemental EIS for license renewal. Further, the supplemental EIS 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.

In February 2022, the Commission directed the staff to propose a rulemaking plan to revise NUREG-1437 and 10 CFR Part 51 to specifically address SLR applications.

On August 6, 2024, the NRC published its final rule, "Renewing Nuclear Power Plant Operating Licenses-Environmental Review," (89 FR 64166) revising its environmental protection regulations in 10 CFR Part 51, along with Revision 2 of the LR GEIS. The final rule updates the potential environmental impacts associated with the renewal of an operating license for a nuclear power plant for up to an additional 20 years for either an initial license renewal or one period of SLR. Revision 2 of the LR GEIS provides the technical basis for the final rule. The revised LR GEIS supports the updated list of environmental issues and associated environmental impact findings contained in Table B-1 in Appendix B to Subpart A of the revised 10 CFR Part 51 for both initial license renewal and one period of SLR. The final rule was updated with a correction to Appendix B to Subpart A published on August 21, 2024 (89 FR 67522).

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 2025, 97 U.S. nuclear reactors have received initial license extensions, with 13 of the 94 currently operating U.S. nuclear reactors having received subsequent renewed licenses.

Ten units that received initial renewed operating licenses (license renewal for 40–60 years) have subsequently shut down: Crystal River Nuclear Generating Plant, Unit 3; Duane Arnold Energy Center; Fort Calhoun Station, Unit 1; Indian Point Nuclear Generating, Units 2 and 3; Kewaunee Power Station; Oyster Creek Nuclear Generating Station; Palisades Nuclear Plant; Pilgrim Nuclear Power Station; Three Mile Island Nuclear Station, Unit 1; and Vermont Yankee Nuclear Power Station. As noted in section 2.3.2.6, the NRC has received requests to restart Palisades Nuclear Plant, Three Mile Island Nuclear Station, Unit 1 (now Crane Clean Energy Center), and Duane Arnold Energy Center.

For a list of plants that are expected to apply for initial 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 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 an SLR once it enters the initial period of extended operation (i.e., the 20-year renewal period beyond its initial 40-year license period). RG 1.188, Revision 2, supports the SLR process.

For a list of plants that are expected to apply for SLR, see https://www.nrc.gov/reactors/operating/licensing/renewal/subsequent-license-renewal.html

The SLR guidance document NUREG-2191, “Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report,” and technical basis document NUREG-2192, “Standard Review Plan for Review of subsequent license renewal applications for Nuclear Power Plants” (SRP-SLR), both initially issued July 2017, also address SLR. As lessons were learned from the review of SLR applications and generic technical issues were resolved, the NRC issued four LR-ISGs for use by applicants until the guidance was incorporated into the next formal update of the documents.

In July 2023, the staff issued NUREG-2191, Revision 1, and NUREG-2192, Revision 1, which incorporated the following ISGs and included additional information and lessons learned from the review of SLR applications:

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

- SLR-ISG-2021-02-MECHANICAL, “Updated Aging Management Criteria for Mechanical Portions of Subsequent License Renewal Guidance,” issued February 2021
- SLR-ISG-2021-03-STRUCTURES, “Updated Aging Management Criteria for Structures Portions of Subsequent License Renewal Guidance,” issued February 2021
- SLR-ISG-2021-04-ELECTRICAL, “Updated Aging Management Criteria for Electrical Portions of Subsequent License Renewal Guidance,” issued February 2021

Following the issuance of Revision 1 of the guidance and technical basis documents, any additional generic technical issues will be addressed by LR-ISGs for use by applicants until the guidance is incorporated into the next formal update of the documents.

The NRC continues to perform long term confirmatory research and exchange of information with international counterparts to ensure additional generic information that will make reviews more effective and efficient as additional licensees submit SLR applications. Domestically, the NRC maintains joint research roadmaps with the U.S. DOE Light Water Reactor Sustainability Program and EPRI to coordinate the agency’s research efforts to understand materials degradation and develop effective strategies to manage aging (e.g., inspection and monitoring activities, harvesting). Internationally, the NRC holds leadership roles in international cooperative efforts to support long-term operation, including IAEA and NEA programs, to exchange information on operating experiences and proven practices, establish a strong technical foundation for long-term operation, and direct scientific research where uncertainty may exist. The NRC also frequently engages in bilateral discussions with international regulators and plant operators to share information on emergent issues and aging management approaches. These efforts ensure that the NRC’s guidance and safety decisions on long-term operation are technically sound and continue to be enhanced effectively and efficiently. Section 2.2.2 of this report provides additional information.

14.1.5 The United States and Periodic Safety Reviews

Many countries conduct periodic safety reviews (typically 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 decision-making process for long-term operation or license renewal. However, alternate processes, such as the NRC license renewal and SLR 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 SLR, 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 6.3.4 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 non-safety-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 at 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 ROP to ensure that the licensee has properly characterized the changes or modifications.

The ROP 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. This 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 fully describes the ROP.

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 facility-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 organizations 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. NRC MD 8.4, "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests," dated September 20, 2019, describes the Commission's policies for these areas. 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. 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 various 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 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 ROP, 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[4] 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 decision-making processes. A risk-informed approach to regulatory decision-making 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) ROP; (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 ROP using risk insights and lessons learned from more than 30 years of regulating nuclear power plants. The previous oversight process evolved during a period when the nuclear power industry was less mature, and there was much less operational experience on which to base rules, regulations, and oversight approaches. Significant plant operating events occurred with some frequency, and the oversight process tended to be reactive and prescriptive, observing plant performance for adherence to the regulations and responding to operational problems as they occurred.

After more than five decades of operational experience, the ROP now focuses the agency's resources on issues based on their safety significance and on the relatively few plants requiring

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[4] 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.

additional regulatory attention based on their performance. In general, the ROP 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 ROP 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 ROP also features comprehensive quarterly reviews and expanded annual reviews, which include inspection planning and performance reporting (all posted on the NRC's public website at https://www.nrc.gov/reactors/operating/oversight.html). The ROP 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 ROP.

14.1.5.5 Licensee Responsibilities for Safety: Regulations and Initiatives Beyond Regulations

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[5] 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:

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[5] 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," issued March 2013 by the Western European Nuclear Regulators' Association Reactor Harmonization Working Group.

- 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 decision-making 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.

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 contributor to NEA and IAEA safety standards and guidance documents on aging management. For example, the NRC staff have been involved 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 initiative 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. The NRC's involvement in this program 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 that 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 Expert Group on Operating Experience to gather insights on international safety issues. In addition, the NRC promptly evaluates safety events to ensure that 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. Sections 12.2.3 and 18.1.4 of this report give 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 WENRA.

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.

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 ROP 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 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.

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 U.S. 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 the 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"). The 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 ROP 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 ROP 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 and 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 10 CFR.

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 January 1977, and in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers," Volumes 1–8, issued from 1978 to 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 June 1987. The 1991 revision to 10 CFR Part 20 also adopted the same dose limit for a member of the public recommended in ICRP Publication 60, "1990 Recommendations of the International Commission on Radiological Protection," issued November 1990. 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, the 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 only be imposed on existing reactors if the requirement is 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," issued 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. The regulation in 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 to 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," issued 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 (Revision 1).

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 website at https://www.nrc.gov/reading-rm/doc-collections/fact-sheets/tritium-radiation-fs#limits

15.3.1 Control of Radiation Exposure of Occupational Workers

The NRC staff has been collecting 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 by approximately a factor of 10, to the current level of about 0.58 person-Sv (58 person-rem) per reactor, based on a 3-year rolling average between 2021 and 2023.

In 2023, 83,784 workers at nuclear plants were monitored for radiation exposure. In 2023, the median collective dose for boiling-water reactors (BWRs) and PWRs was 0.81 person-Sv (81 person-rem) and 0.26 person-Sv (26 person-rem), respectively. Of the monitored workers, 34,493 received a measurable dose. The collective measurable dose was 55.52 person-Sv (5,552 person-rem) for an average of 0.0016 Sv (0.16 rem) per worker. Of the workers who received a measurable dose in 2023, 81 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 in 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 website at https://www.nrc.gov/about-nrc/radiation.html. Information posted on the NRC website 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," issued December 1978, and NUREG-0654/FEMA-REP-1, Revision 2, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," issued

December 2019, describe the emergency planning basis. NUREG-0654/FEMA-REP-1, Revision 2, reflects changes to both NRC and FEMA regulations, guidance, policies, and doctrine, as well as advances in technology and best practices that have occurred since the document was originally issued in November 1980. This update also incorporates the four supplemental documents and addenda that have been issued in the intervening years, and modernizes and consolidates the guidance, making it easier for users to understand. These criteria provide a basis for licensees and States, Tribal, and local governments to develop radiological emergency plans.

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 alternating current 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 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 in the United States fundamentally changed as a result of the 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, the 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 FEMA "Nuclear/Radiological Incident Annex to the Response and Recovery Federal Interagency Operational Plans" (available at https://www.fema.gov/sites/default/files/documents/fema_incident-annex_nuclear-radiological.pdf), 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), the 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, the 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 (the licensee) and the State, Tribal, or local government as the two primary decision-makers 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 taking 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 plans and procedures detailing its response to incidents involving licensed facilities, material, and activities. 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.

The NRC may decide to activate its Incident Response Program at the NRC Headquarters Operations Center, at 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 the DOE (i.e., field teams and aerial monitoring).

16.1.2.4 Aspects of Security that Support Response

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 partners, and licensees, and continually evaluates this information to assess threats to regulated facilities or activities. The NRC works with other Federal agencies, particularly the 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, Revision 6, “Development of Emergency Action Levels for Non-Passive Reactors,” issued November 2012, is endorsed in RG 1.101, Revision 6, “Emergency Response Planning and Preparedness for Nuclear Power Reactors,” issued June 2021, and provides the latest guidance for the development of emergency action levels. NEI 07-01, Revision 0, “Methodology for Development of Emergency Action Levels Advanced Passive Light Water Reactors,” issued July 2009, as endorsed in RG 1.101, is the guidance for developing emergency action levels for the AP1000 and GE-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 website at https://www.nrc.gov/about-nrc/emerg-preparedness/regs-guidance-comm.html

These documents are all considered generic guidance, as they are not plant specific and may not be entirely applicable for some reactor 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," dated November 19, 2015, 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, "Review and approval of state and local radiological emergency plans and preparedness," 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, the 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 ensuring that the arrangements of the memorandum of understanding are carried out.

16.1.3.3 Emergency Preparedness Facilities

In 10 CFR 50.47, “Emergency plans,” 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, Tennessee, and a second in Phoenix, Arizona. 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 November 2011, and EPA-400/R-17/001, “PAG Manual: Protective Action Guides and Planning Guidance for Radiological Incidents,” issued 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. NUREG/CR-6953, “Review of NUREG-0654, Supplement 3, ‘Criteria for Protective Action Recommendations for Severe Accidents,’” Volumes 1, 2, and 3, documents these analyses.

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, the 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 a role in 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 ROP 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 its significance determination process to assess the significance of inspection findings. Article 6 of this report discusses the NRC's ROP and significance determination process.

Emergency preparedness is one of the seven cornerstones of safety in the ROP. 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 emergency response facility and equipment readiness. 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 emergency response facility and equipment readiness indicator monitor the availability of the technical support center, emergency operations facility, and equipment necessary to implement the site emergency plan.

The emergency preparedness cornerstone of the ROP includes the following areas for inspection:

- Maintenance of Emergency Preparedness Program—NRC inspectors evaluate the licensees' efforts to identify and resolve program weaknesses, adequacy of internal program assessment activities, emergency plan change process, maintenance of equipment important to emergency preparedness, evacuation time estimate population monitoring, and implementation of emergency response facility maintenance.

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

- Exercise Evaluation—NRC inspectors independently observe the licensee's performance in classifying, notifying, and developing recommendations for protective actions and other activities during the exercise. Evaluated exercise scenarios are varied over an 8-year exercise cycle to include a hostile action event, no radiological release, or minimal release not requiring public protective actions, and a rapidly progressing event. The NRC inspectors assess whether the licensee's self-critique is consistent with their observations. The emergency preparedness performance indicators for drill and exercise performance rely on the accurate determination of successful performance and the correction of identified weaknesses during the conduct of drills and exercises. If a licensee either fails to properly critique performance or correct identified weaknesses, then the validity of the drill and exercise performance indicators come into question. Performance problems with classification, notification, dose assessment, and protective action recommendations are the highest priority inspection areas. Exercise evaluation results are provided in inspection reports available on the NRC's public website. 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.

- Emergency Response Facility and Equipment Readiness Evaluation—NRC inspectors verify occurrences when the technical support center or emergency operations facility is nonfunctional, or equipment necessary to implement the emergency plan is not available or functional, such that a risk-significant planning standard function or response could not be performed for greater than 168 hours from the time of discovery and no compensatory measures were implemented.

- Emergency Action Level and Emergency Plan Changes—NRC inspectors review all of the licensee's changes to emergency action levels and a sample of changes to the emergency plan to determine if any of the changes have reduced the effectiveness of the emergency plan.

- Emergency Response Organization Staffing and Augmentation System—NRC inspectors review the augmentation system to determine whether, as designed, it will support augmentation of the emergency response organization in accordance with the goals for activating the emergency response facility.

- Reactor Safety/Emergency Preparedness—NRC inspectors verify that the data reported for the performance indicator values are valid.

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 for Cooperation and Exchange of Information in Nuclear Regulatory Matters between the United States Nuclear Regulatory Commission and the Canadian Nuclear Safety Commission," was most recently renewed in 2023 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 "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 2022 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 as opportunity allows.

The NRC routinely practices emergency communications with its Canadian and Mexican counterparts during its emergency drills. The NRC also participates in the 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. In 2024, the NRC supported information sharing during ConvEx 2e exercise associated with an NRC-licensed facility in the northeast. In March 2025, the NRC participated in a full-scale emergency exercise titled Cobalt Magnet. This once-in-a-decade exercise, focused on immediate response and near-term recovery activities following a radiological or nuclear disaster, brought together Federal, State, Tribal, and local response organizations, with participation by IAEA and involvement of Canadian authorities. The simulated environment fostered collaboration and coordination; validated concepts of operation; and identified best practices, capability gaps, and areas for improvement.

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 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 the IAEA any events resulting in a categorization of level 2 or higher. The NRC participates in the IAEA's Unified

System for Information Exchange for Incidents and Events as the method for rapidly sharing nuclear or radiological event information with the 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 the 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 the 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 website, 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 continues capturing lessons learned and licensing experiences and updating their guidance, as necessary.

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 standard design certifications 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 takes 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. It explains the basic framework for assessing non-seismic, seismic, and other geological factors important to siting. It also 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, transportation routes, and dams); and physical characteristics of the site (such as geological, 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," which 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. Under 10 CFR 100.10(c) and 10 CFR 100.20(c), the physical characteristics of a site (including seismology, meteorology, geology, and hydrology) must be considered in determining its acceptability for a nuclear power reactor. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design bases for protection against natural phenomena," also requires that SSCs important to safety should be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.

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 many RGs to provide guidance on approaches that applicants can use to address issues of site safety and meet applicable requirements. RG 4.7, Revision 4, "General Site Suitability Criteria for Nuclear Power Stations," issued February 2024, 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 provides criteria that the NRC staff uses in reviewing the site safety content of the applicant's safety analysis report and evaluating whether an applicant or licensee meets applicable NRC regulations. 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 RG 4.27, "Use of Plant Parameter Envelope in Early Site Permit

Applications," in June 2023, 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, blast 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, Revision 2, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," issued December 2021; RG 1.91, Revision 3, "Evaluations of Explosions Postulated To Occur at Nearby Facilities and on Transportation Routes Near Nuclear Power Plants," issued November 2021; and NUREG-0800, Section 3.5.1.6, Revision 4, "Aircraft Hazards," issued March 2010.

Radiological dose calculations must use meteorological data representative of 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. RG 1.23, Revision 1, "Meteorological Monitoring Programs for Nuclear Power Plants," issued March 2007, gives acceptable approaches for obtaining meteorological data. The meteorological data are used in the estimation of the onsite and offsite atmospheric dispersion values. RG 1.76, Revision 1, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," issued March 2007, and RG 1.221, Revision 0, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants," issued October 2011, are used in safety analyses or to establish plant design bases for phenomena such as wind loads or impacts from tornado-generated or hurricane-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, Revision 2, "Ultimate Heat Sink for Nuclear Power Plants," issued 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 manmade hazards such as dam failures. RG 1.59, Revision 2, "Design Basis Floods for Nuclear Power Plants," issued August 1977, provides acceptable approaches for conducting flood-hazard evaluations. DG-1290, Revision 1, issued July 2024, contains the proposed Revision 3 of RG 1.59.

Site characteristics are also 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," issued March 2007, combines the site seismic hazard curves and seismic fragility curves for nuclear structures to meet a specified performance target. RG 1.208 also incorporates recent developments in seismic hazard assessment, including the use of 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," issued 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," issued 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.

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, Revision 1, "Volcanic Hazards Assessment for Proposed Nuclear Power Reactor Sites," issued August 2023, to provide guidance for a risk-informed approach for conducting a probabilistic volcanic hazards assessment, including how to apply the guidelines for a SSHAC study in NUREG-2213 to volcanic hazards. This guidance allows for the performance of either detailed hazard analysis or engineering analysis to inform site characterization. In September 2024, the NRC staff completed the review of the first volcanic hazards assessment following the methodology outlined in RG 4.26 and issued the final safety evaluation report of the TerraPower's Topical Report NAT-3226, "An analysis of the potential volcanic hazards at the proposed natrium site near Kemmerer, Wyoming."

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 PRA 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., less than 95 percent) of low probability (i.e., less than 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, Revision 1, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” reissued February 1983

- RG 1.183, Revision 1, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” issued October 2023

- 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 February 1995, provides updated information on light-water reactor accident source terms. In supplying guidance on the implementation of NUREG-1465, RG 1.183, Revision 1, 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 that apply to siting and doses to members of the public and control room operators) 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 of 10 CFR Part 100, along with the alternative source term described in RG 1.183, in its design certification review for a passive light-water reactor, the AP600 design. More recently, the agency applied this approach to the AP1000, ESBWR, APR1400, and NuScale designs with similar results. For non-LWR 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 NRC has been engaged with stakeholders regarding a planned update of RG 1.183 to accommodate the planned deployment of accident tolerant, higher burnup, and increased enrichment fuel designs.

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 EIS. RG 4.2, Revision 3, "Preparation of Environmental Reports for Nuclear Power Stations," issued September 2018, guides applicants in preparing environmental reports (which the NRC uses to prepare the EIS) 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 check 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 regarding 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 published its first environmental standard review plan in the 1970s. Since then, 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.

Following the issuance of the Commission Memorandum and Order CLI-09-21, issued on November 3, 2009, the NRC began to evaluate the effects of greenhouse gas emissions and their implications for global climate change in its environmental reviews. As documented in EISs, the NRC staff considers the potential effects of climate change on the environment and greenhouse gas emissions associated with the proposed action. The staff evaluates the contribution of the proposed action on climate change by quantifying greenhouse gas emissions from the project for the duration of the project life. The staff documents the potential impacts of climate change on environmental resources. This analysis considers how climate change can induce changes in environmental resource conditions. EISs reflect the most recent climate change information available. This includes information published in climate change reports from the U.S. Global Change Research Program, the Intergovernmental Panel on Climate Change, and the National Oceanic and Atmospheric Administration.

17.4 Reevaluation of Site-Related Factors

Although operating nuclear power plants are not reevaluated periodically for site-related factors, the continued safety of nuclear plants and the adequate protection of a licensed plant are ensured under the NRC's regulatory framework. 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 adequately protects public health and safety 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. NRR Office Instruction LIC-208, "Process for the Ongoing Assessment of Natural Hazards Information (POANHI)," dated November 20, 2019, provides guidance on how to collect, integrate, and evaluate new information for consideration in its regulatory decision-making. POANHI is used to continuously evaluate new natural hazard data, such as seismic or flooding information, that could impact nuclear plant safety. To date, the NRC staff have evaluated the seismic hazard characterizations at several central and eastern United States nuclear power plant sites to assess the impact of the 2018 NGA-East ground motion model.

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 344-2013, "IEEE Standard for Seismic Qualification of 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.

In September 2021, The NRC published NUREG/KM-0015, "Considerations for Estimating Site-Specific Probable Maximum Precipitation at Nuclear Power Plants in the United States of America," 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. In addition, in December 2021, the NRC published NUREG/KM-0017, "Seismic Hazard Evaluations for Nuclear Power Plants: Near-Term Task Force Recommendation 2.1 Results," to document the current best knowledges and practices for characterizing the site-specific hazards for each nuclear power plant in the United States.

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 serve as the mechanism for cooperative dialogue.

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

17.6 Vienna Declaration on Nuclear Safety

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

ARTICLE 18—DESIGN AND CONSTRUCTION

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

(i) the design and construction of a nuclear installation provides for several reliable levels and methods of protection (defense in depth) against the release of radioactive materials, with a view to preventing the occurrence of accidents and to mitigating their radiological consequences should they occur

(ii) the technologies incorporated in the design and construction of a nuclear installation are proven by experience or qualified by testing or analysis

(iii) the design of a nuclear installation allows for reliable, stable, and easily manageable operation, with specific consideration of human factors and the man-machine interface

This section explains the defense-in-depth philosophy and how it is embodied in the general design criteria of U.S. regulations. It explains how applicants meet the defense-in-depth goals and how the U.S. NRC reviews applications and conducts inspections before issuing licenses to ensure that this philosophy is implemented in practice. Next, this section discusses measures for ensuring that the applications of technologies are proven by experience or qualified by testing or analysis. This section discusses requirements for reliable, stable, and easily manageable operation, specifically considering human factors and the man-machine interface.

18.1 Implementation of Defense in Depth

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

18.1.1 Overview of Regulatory Requirements and Governing Documents

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

- Appendix A and Appendix B to 10 CFR Part 50
- 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, “Staff Requirements—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,” issued April 2016

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

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

- RG 1.233, Revision 0, “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,” issued June 2020

- NUREG-0800, BTP 7-19, Revision 9, “Guidance for Evaluation of Defense-in-Depth and Diversity to Address Common-Cause Failure Due to Latent Design Defects in Digital Safety Systems,” issued May 2024

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, it must seek NRC approval and 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
- 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

NRC reviews of applications for new nuclear power plants ensure that the design of such plants includes 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 Article 13 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 may incorporate ITAAC depending on the chosen licensing process, along with lessons learned from the inspection program used in the previous construction era (1970–1980); from the construction of Watts Bar Nuclear Plant, Unit 2; and from the recent construction of Vogtle Units 3 and 4. IMC 2503, “Construction Inspection Program: Inspections of Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Related Work,” dated October 6, 2020, and IMC 2504, “Construction Inspection Program—Inspection of Construction and Operational Programs,” dated October 15, 2009, describe these inspections. The NRC staff will verify successful ITAAC completion based on these inspections for those facilities being built under the 10 CFR Part 52 licensing process.

The NRC used these defense-in-depth principles during its construction oversight of two AP1000 units, Vogtle Units 3 and 4, which are located in Waynesboro, Georgia, and operated by Southern Nuclear Operating Company. The combined license enables the licensee to construct a plant and to operate it if certain conditions, or ITAAC, are satisfied.

Southern Nuclear notified the NRC that all ITAAC for Vogtle Unit 3 were complete, as required by 10 CFR 52.99(c)(4), on July 29, 2022, and for Unit 4 on July 20, 2023. In accordance with 10 CFR 52.103(g), the NRC found that the ITAAC for both units’ combined license were met. The NRC staff transitioned its oversight of Unit 3 and Unit 4 from the Construction ROP to the ROP after it made the 10 CFR 52.103(g) finding.

Section 2.3.3.3 of this report provides additional details about the actions the NRC took to transition from construction to operation of Vogtle Units 3 and 4. To enhance knowledge retention, the staff documented its lessons learned from the construction oversight of these plants in a memorandum titled “Vogtle, Units 3 and 4, and V.C. Summer, Units 2 and 3, 10 CFR Part 52 Construction Lessons-Learned Report,” dated January 16, 2024.

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 final safety analysis report ensures that defense in depth is maintained.

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 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 that core cooling, containment, and SFP integrity are 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.

The NRC's response to the Fukushima Dai-ichi accident demonstrated how the staff applied the defense-in-depth philosophy to address and evaluate the lessons learned from that event. Following publication of the SECY-11-0093, "NRC's Near-Term Task Force Recommendations for Enhancing Reactor Safety in the 21st Century," July 12, 2011, 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, "Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," dated March 12, 2012, 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. 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. The rule specifically requires licensees to develop guidelines to mitigate external events from natural phenomena assuming a loss of all alternating current 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.

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 loss-of-coolant accident 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 loss-of-coolant accident mitigation strategy, were enough to demonstrate that the device will perform as stated.

As with large light-water reactors, the NuScale SMR design has a leaktight 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 leaktight 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,” issued 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 January 2023, the NRC certified NuScale Power LLC’s 12-module SMR design. As part of efforts to update and evolve its design, NuScale submitted licensing Topical Report (TR)-0420-69456, Revision 0, “NuScale Control Room Staffing Plan,” 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 TR-0420-69456 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 SMR designs, including its NuScale US460 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 factor's 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 DI&C 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 potential failure may be addressed using diversity and defense in depth in the application of DI&C systems. Depending on the risk significance of the common-cause failure, design techniques, prevention measures, or mitigation measures other than diversity may be acceptable.

- Digital system architecture raises 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 DI&C 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 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 DI&C can be found at https://www.nrc.gov/reactors/digital.html

The staff has completed its safety reviews of the instrumentation and control systems for the AP1000, ESBWR, APR1400, and NuScale US600 and US460 reactor designs. The staff also completed review of the instrumentation and control design for the ABWR design certification renewal. The staff continues to review topical reports on DI&C platforms that can be referenced in subsequent applications for design certification, construction permit, operating license, and site-specific license amendment requests.

To support the review of applications for SMR 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 SMR design have increased the efficiency and effectiveness of the instrument and control licensing review. In January 2023, the NRC certified the NuScale 12-module SMR design (10 CFR Part 52, Appendix G). The NRC staff used the lessons learned in evaluating the NuScale instrumentation and control design to develop the design review guide. The staff used the design review guide to evaluate the SDAA of the NuScale 6-module US460 design, which is described in section 2.3.3.2 of this report.

The NRC issued a technology-inclusive design review guide, "Instrumentation and Controls for Non-Light-Water Reactor (Non-LWR) Reviews," in February 2021. The guidance supports (1) flexible regulatory review processes for non-LWRs within the bounds of existing regulations; and (2) a new non-LWR regulatory framework that is risk-informed and performance-based; and (3) features the NRC staff's review efforts commensurate with the demonstrated safety performance of non-LWR technologies. The staff is using this design review guide in evaluating the instrumentation and control system design of X-Energy Xe-100 and TerraPower Natrium non-LWRs.

The NRC participates in the Working Group on New Technology within the 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 a task group on DI&C assembled under the auspices of the Working Group on New Technology. The primary intent of this task group is to update an existing NEA guidance document to address recent technical developments in assessing common-cause failure for DI&C applicable to both operating and new reactors. This international engagement is beneficial in building international technical consensus among nuclear regulators on common DI&C issues and in developing technical documents that introduce performance-based approaches to the safety analysis of DI&C systems.

Section 2.3.1.4 of this report discusses the DI&C system regulatory program and processes.

18.3.2.3 Cybersecurity

In March 2009, the NRC issued the cybersecurity rule, 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 (SSEP) functions are protected from cyberattacks up to and including the design-basis threat. To meet the cybersecurity rule requirements, operating power reactor licensees were required 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 license application. All operating nuclear power plant licensees met the submission deadline, and the NRC reviewed and approved all licensees' cybersecurity plans. Essential elements of a plan include establishing a cybersecurity assessment team that has the authority to oversee the cybersecurity program, describing the process for identifying digital assets that perform or support SSEP functions (these digital assets are called critical digital assets (CDAs)), describing the defense-in-depth approach (i.e., protective strategy) and requirements for implementation of a comprehensive set of security controls to protect CDAs, and establishing a process for the ongoing and monitoring assessment of the security controls and programmatic areas of the cybersecurity program. The cybersecurity plan also includes commitments to review and maintain the cybersecurity plan and the program, and provide documentation that demonstrates how that process is accomplished.

In 2010, the NRC published Revision 0 of RG 5.71, “Cyber-Security Programs for Nuclear Power Reactors,” which contains implementation guidance to licensees and applicants on an acceptable method for satisfying the requirements of 10 CFR 73.54. In 2023, the NRC published Revision 1 of RG 5.71. This revision provided additional guidance on risk-informed security approaches for cybersecurity plan implementation. In addition, the NRC has reviewed and found acceptable for use industry guidance documents (e.g., NEI 08-09, “Cybersecurity Plan for Nuclear Power Reactors”) that contains implementation guidance to licensees. Guidance reviewed and approved by the NRC is documented via a regulatory guide or via a letter to NEI.

Also in 2010, the NRC and the Federal Energy Regulatory Commission (FERC) and the North American Electric Reliability Corporation (NERC) entered into an agreement via memorandum of agreement (MOA) and memorandum of understanding, respectively, to address common and overlapping nuclear plant cybersecurity roles, responsibilities, and areas of coordination with these organizations. The NRC and FERC renewed the MOA in 2022. This agreement provided the basis for the NRC Commission to determine 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. FERC and NERC found this policy decision acceptable, and they also found the NRC’s regulatory framework was robust to meet or exceed NERC’s cybersecurity requirements for power generation plants. Under the memorandum of understanding, the NRC staff will continue to coordinate with NERC share relevant operating experience and other related technical information.

The NRC has developed a cybersecurity oversight program that includes a Cyber Assessment Team, an inspection program, inspector training, and a process for evaluating the significance of inspection findings. Stakeholders, including members of industry and representatives from DHS, FERC, and NIST, collaborated with the NRC in developing parts of this program. The NRC completed inspection activities related to the first phase of licensees’ implementation schedules (interim milestones) in calendar year 2015. Most NRC licensees implemented the second phase of their implementation schedule (remaining aspects of the program, including controls for a greater number of systems and processes) between 2017 and 2019. The NRC started cybersecurity program full implementation inspection activities in calendar year 2017 and completed the inspections in 2021. After the NRC completed these inspection activities, The NRC developed IP 71130.10, “Cybersecurity,” dated January 1, 2022, to incorporate cybersecurity inspection in the NRC ROP and to focus on changes to the environment (digital upgrades), CDAs, and the cybersecurity program. Based on lessons learned from the last 13 years of operating experience, the results of a 2019 self-assessment of the cybersecurity program implementation, feedback from the industry, and advances in technology, the NRC, is updating NRC and industry available guidance involving stakeholders and the public to improve the regulatory efficiency and effectiveness of cybersecurity at U.S. nuclear power plants.

Consistent with the ROP, the NRC conducts cybersecurity inspections in accordance with section 7.2.3. To date, results from these inspections have identified findings of very low significance. Although these findings were of very low significance, these findings and emergent threats underscore the need for continued improvements in protecting critical digital assets and maintaining strong cybersecurity programs. These findings were enforced in accordance with section 7.2.4 to ensure licensees are in compliance with regulations, license conditions, and to verify that the licensee is safely operating the facility.

The 10 CFR Part 53 rulemaking discussed in section 2.3.1.1 above provides the option to future applicants to either implement the cybersecurity requirements in 10 CFR 73.54 or those in a new section in Part 73.110, "Technology Inclusive Requirements for Protection of Digital Computer and Communication Systems and Networks." The NRC staff, supported by the national laboratories, also developed a draft regulatory guide DG-5075 (proposed RG 5.96), "Establishing Cybersecurity Programs for Commercial Nuclear Plants licensed under 10 CFR Part 53," to provide a commercial nuclear power reactor under a Part 53 license with an acceptable approach for meeting the requirements of 10 CFR 73.110. The new risk-informed, performance-based, and technology-inclusive regulatory framework that can be adapted to the wide range of technologies. Furthermore, the NRC continues conducting research activities to further improve the regulatory efficiency and effectiveness of its current cybersecurity oversight program, assess and evaluate the cybersecurity posture of emergent technologies and advanced reactors.

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

18.4 New Reactor Construction Experience Program

The Operating Experience Program at the NRC also includes the evaluation of issues at operating reactors and new construction projects that could provide lessons for future new and advanced reactor construction. Construction and operating experience events related to latent design and construction deficiencies, significant design changes, installation and testing activities, and heavy loads are shared to prevent recurrence. The nuclear industry in the United States gained significant construction quality and design experience in the 1970s and 1980s. In 1984, the NRC issued NUREG-1055, "Improving Quality and the Assurance of Quality in the Design and Construction of Nuclear Power Plants, A Report to Congress," to document the lessons learned from plant construction. In 2007, the NRC developed a Construction Experience Program to support new light-water reactor construction activities. 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. Lessons learned following the completion of these activities in late 2023 are captured in the report "Vogtle, Units 3 and 4, and V.C. Summer, Units 2 and 3, 10 CFR Part 52 Construction Lessons-Learned Report," dated January 16, 2024.

In anticipation of new construction activity, the NRC is in the early stages of developing a program to identify, communicate, and apply operating experience to the design, licensing, construction, and commissioning phases of the advanced reactor program.

Section 19.7 of this report describes the actions related to evaluation, communication, and application of construction experience as an integral part of the operation experience program. This includes the use and sharing of new construction experience with international counterparts through the 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 <u>Initial Authorization to Operate</u>

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

The two-step licensing process in 10 CFR Part 50 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 EIS. 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 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. As mentioned in section 6.3.1 of this report, the Commission approved revising the duration of a design certification from 15 years to 40 years (SRM-COMDAW-24-00001).The NRC has certified seven standard plant designs under the design certification process: (1) GE's ABWR, (2) Westinghouse Electric Company's System 80+ (originally designed by Combustion Engineering), (3) Westinghouse's AP600 design, (4) Westinghouse's AP1000, (5) GE-Hitachi's ESBWR, (6) Korea Hydro and Nuclear Power's APR1400, and (7) NuScale Power's NuScale US600. In May 2025, the NRC approved the SDAA for NuScale US460.

The NRC has also received a first-of-a-kind submittal from a licensee seeking to reauthorize power operations at a nuclear power plant where the operator had previously certified permanent cessation of operations, and other licensees of similarly situated plants have expressed interest in the same. Section 2.3.2.6 of this report discusses this in more detail.

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. The NRC must hold a hearing before deciding whether to issue a combined license as it would for a construction permit. 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 licenses 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, issued September 2021, and NUREG-2194, Revision 1, "Standard Technical Specifications: Westinghouse Advanced Passive 1000 (AP1000) Plants," Volumes 1 and 2, issued January 2024.

The NRC encourages licensees to use the standard technical specifications as the basis for plant-specific technical specifications. Implementation of the improved standard technical specifications is expected to improve the safety of nuclear power plants through the use of more operator-oriented technical specifications, improved technical specification bases, reduced action statement induced plant transients, and more efficient use of NRC and industry resources. 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 have converted their technical specifications to the improved standard technical specifications.

Consistent with the Commission's policy statements on technical specifications and the use of PRAs, the NRC and the nuclear industry have developed risk-informed improvements to technical specifications. 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 specification 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 to 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," issued November 1980; NUREG-0737, Supplement 1, "Requirements for Emergency Response Capability," issued January 1983; and NUREG-0899, "Guidelines for the Preparation of Emergency Operating Procedures," issued August 1982.

After the 1979 accident at Three Mile Island 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, "Staff Requirements—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 ROP. 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. The Commission approved a final rule, 10 CFR 50.155, 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 inspecting the training programs if they have not been accredited. In addition, the NRC's ROP, 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 ROP 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 ROP. 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 ROP Action Matrix.

The NRC staff issued IMC 2562 to clearly provide oversight policies, requirements, and guidance for when a licensee seeks to transition a decommissioning reactor facility to an operational status subject to the ROP. The associated inspection activities will verify that reactor operations can be conducted safely and securely before the reactor facility is restarted.

19.6 <u>Incident Reporting</u>

The licensee event reporting requirements of 10 CFR 50.72 and 10 CFR 50.73 help to achieve the NRC's safety mission and emergency response responsibilities. The regulation in 10 CFR 50.72 requires immediate notification requirements through the emergency notification system, and 10 CFR 50.73 requires detailed reports of significant events within 60 days of the occurrence. 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 website at https://www.nrc.gov/reading-rm/doc-collections/event-status/index.html.

The NRC staff uses this information 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 public. Evaluations of events as documented in NRC inspection reports are available on the NRC's public website. 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 public health and safety events, is also publicly available at https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0090/index

In September 2024, the NRC issued NUREG-1022, Revision 3, "Event Reporting Guidelines: 10 CFR 50.72 and 50.73," Supplement 2, "Event Report Guidelines 10 CFR 50.72(b)(3)(ii) and 10 CFR 50.73(a)(2)(ii)." The NRC guidance 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, 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 have reflected improvements in reactor safety system performance and decreasing trends in the number of reactor transients and significant events. For example, between 2007 and 2023, two commercial reactor events were identified through the abnormal occurrence criteria in NUREG-0090:

- 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 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 timely and appropriate communication to stakeholders.

Collection. As discussed in Section 2.3.1.3 of this report, the NRC is exploring ways to improve the data collection process using the Mission Analytics Portal External as a platform for licensee submission of notifications. In addition, the Data Warehouse provides an onsite repository of data that allows easier review, analysis, and connections between datasets.

The NRC facilitates the collection, storage, and retrieval of operating experience data through an internal website containing the Operating Experience Hub. This site provides a centralized repository of links to databases, dashboards, and analytics relevant to operating experience and the ROP. 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 been collecting 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.

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. INPO shares these data with all its members to allow for review and assessment, and in a limited fashion, with the NRC.

Screening. The NRC reviews event notifications and lower-level operating experience (e.g., from inspector feedback) biweekly 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 website and from the IAEA International Reporting System for Operating Experience; and any items of potential interest brought forward by the Office of Nuclear Regulatory Research.

Evaluation. Regardless of safety or risk significance, operating experience is collected, reviewed, and screened by the NRC staff for potential followup actions. Followup actions include email notifications to technical subject matter experts for review and possible analysis and trending. In addition, these events are included in dashboards and other analytics tools available on the Operating Experience Hub for easy access and retrieval. Events that may be of broader interest to inspection staff may be summarized in an operating experience internal communication or considered for inclusion in the annual inspection planning and assessment reviews. Items that meet the criteria for both safety significance and generic applicability are evaluated for potential additional actions. 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.

Application. Finally, the Operating Experience Program shares the results of NRC staff evaluations. At a minimum, technical experts are notified of operating experience data in their fields of expertise, but additional followup actions and communications may be warranted such as internal reports, briefings, or training. Other communications may include the issuance of a generic communication, a proposal for rulemaking, a referral for further study as a generic safety issue (GSI), or a revision of IPs.

The NRC also participates in the International Nuclear and Radiological Event Scale and the IAEA International Reporting System for Operating Experience to share operating experience internationally, to communicate the safety significance of events, and to review events from other Member countries. Operating experience personnel review all domestic commercial reactor event notifications 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 website. The NRC screens all international reactor events posted to this website to determine the potential followup actions based on safety significance and applicability to U.S. plants. The NRC submits relevant U.S. 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 Eighth 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), issued August 2024.

The NRC has issued regulations and guidance for nuclear power reactor licensees to ensure the safe management and disposal of LLW. 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 to 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, Revision 4, “Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants,” issued January 2016

- RIS 2004-17, Revision 1, “Revised Decay-in-Storage Provisions for the Storage of Radioactive Waste Containing Byproduct Material,” 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, Revision 2, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," issued 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.

Radioactive Material Effluent Controls, Reporting, and Procedures. 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.

The NRC gives guidance on the measuring and reporting of liquid and gaseous radioactive waste in RG 1.21, Revision 3, "Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste," issued September 2021. The following documents provide the NRC's guidance on measuring radioactive material in the environment:

- RG 4.1, Revision 2, "Radiological Environmental Monitoring for Nuclear Power Plants," issued June 2009

- RG 4.13, Revision 2, "Environmental Dosimetry—Performance Specifications, Testing, and Data Analysis," issued June 2019

- RG 4.14, Revision 1, "Radiological Effluent and Environmental Monitoring at Uranium Mills," issued 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. NUREG/CR-2907 annually summarizes the data from these effluent reports. The most recent effluent and environmental monitoring report for each nuclear power plant is provided on the NRC website 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., ground water). 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, the 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; October 16, 1981) 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 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 15.3 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 LLW disposal facilities. There are currently four commercial LLW disposal facilities in the United States, all of which are regulated by Agreement States. The NRC is proposing to amend its regulations in 10 CFR Part 61, which govern LLW land disposal facilities. The rulemaking is intended to update the regulations, which were first promulgated in 1982, to be more risk informed and performance based. The proposed Integrated Low-Level Radioactive Waste Disposal rule ensures that the LLW streams that are significantly different from those considered during the development of 10 CFR Part 61, such as depleted uranium, continue to be disposed of safely and meet the performance objectives for land disposal of LLW, as well as other regulatory improvements. The proposed rule includes criteria for licensing the near-surface disposal of greater-than-Class C (GTCC) waste, generally within the upper 30 meters of the earth's surface, and provides for Agreement State licensing of those GTCC LLW streams that meet the regulatory requirements and do not present a hazard such that NRC should retain disposal authority. The NRC staff delivered a proposed rule to the Commission in SECY-24-0045, "Proposed Rule: Integrated Low-Level Radioactive Waste Disposal," dated May 29, 2024.

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 GTCC waste[6] in 10 CFR Part 72. Consolidated interim storage facilities are proposed for the interim storage of spent fuel and reactor-related greater than Class C LLW 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, Texas. 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 in Lea County, New Mexico. 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 GTCC LLW for 40 years. In May 2023, the NRC issued a license to Holtec International to receive, possess, store, and transfer up to 8,680 metric tons of commercial spent nuclear fuel for 40 years. The NRC has posted additional information on consolidated interim storage facilities on its website at https://www.nrc.gov/waste/spent-fuel-storage/cis.html

The United States 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, Nevada, for spent fuel and high-level waste disposal. On March 3, 2010, the DOE

6 The NRC's classification system in 10 CFR Part 61 includes Class A, B, and C LLW that is suitable for land disposal. LLW that does not meet the criteria for these classes is considered greater than Class C (GTCC) and eventually will be managed by the DOE in a yet-to-be-determined manner. Until then, licensees must manage (store) such waste. Regulations in 10 CFR Part 72 address the onsite management of GTCC LLW in independent storage facilities.

filed a motion to withdraw its license application, which the Atomic Safety and Licensing Board denied on 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," issued August 2010

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

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

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

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

INPO®

January 2025

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 operations 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 Government."

All organizations that have a license to operate or construct commercial nuclear plants in the United States have maintained continuous membership in INPO, which currently has 22 members. In addition, many utility organizations that jointly own U.S. nuclear power plants are associate members. 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 (PO&Cs), Principles Documents, and Tier 1 INPO Event Reports (IERs).

(2) It measures and compares industry performance and sustainability 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 actions of any one utility has motivated its continuing support of INPO.

The 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 licensee solely responsible for the safe operation of its nuclear plants.

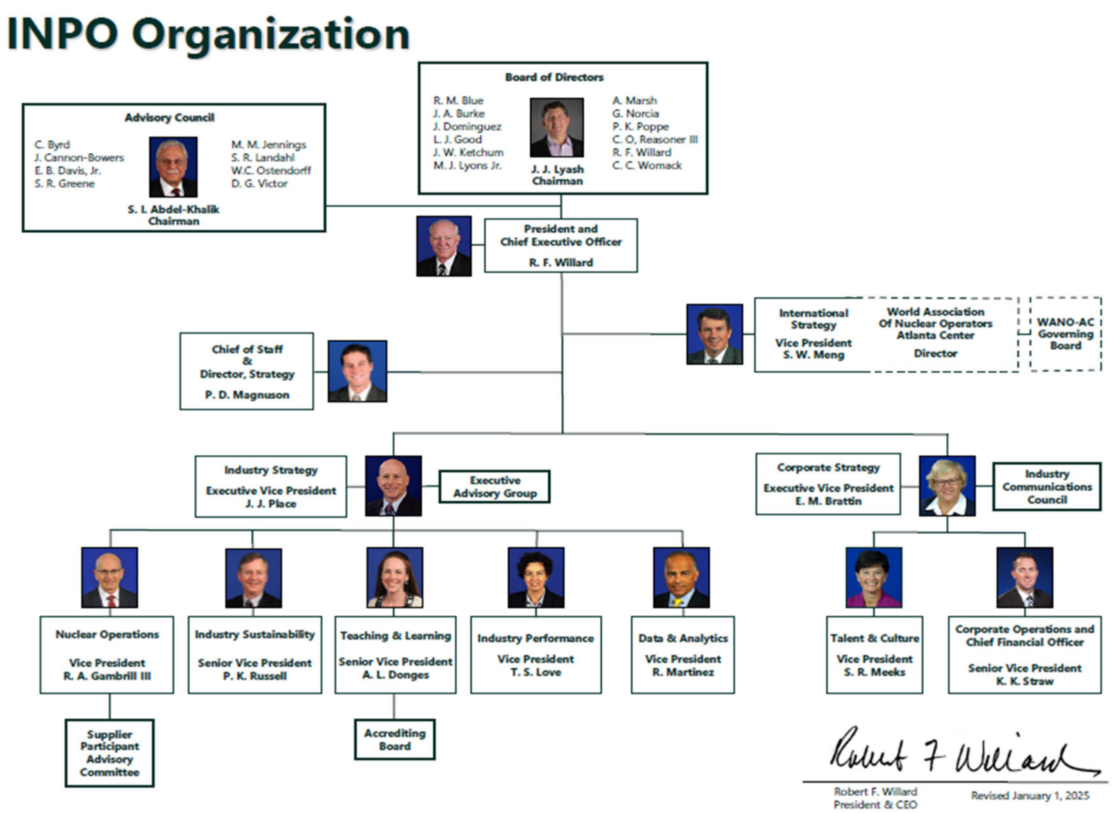
The nuclear industry's commitment to go beyond regulatory compliance and continually strive for excellence, with INPO's support, has resulted in substantial performance improvements over the past 45 years. At the end of 2024, U.S. nuclear stations were continuing to perform at record-high levels. Today, the median industry capability factor is above 93 percent, and collective radiation dose and industrial accident rates are both steady, near all-time lows. Going forward, INPO is at the midpoint of a 10-year strategic design aimed at sustaining this high performance while also promoting continuous improvement.

Despite record performance levels, challenges persist that warrant attention from the industry and INPO. The industry strategy seeks to reduce the number of consequential events attributable to equipment, operations, maintenance, and engineering by half as compared to a 2021 baseline. This is proving to be challenging. Occasional adverse industry trends continue to emerge that demand action from INPO and the industry. For example, in late 2023 and early 2024, the number of scrams increased following 2 consecutive years of record-low scram rates, causing INPO and the industry to enact several corrective measures to reverse this trend.

Finally, two new excellence standards addressing “proficiency” and “resiliency” have been promulgated industrywide to address proficiency contributors that challenge safety and reliability performance and to address a need for greater physical and organizational resilience to external threats, such as extreme weather.

2. Organization and Governance

In many ways, INPO’s organizational structure resembles that of a typical U.S. corporation. A board of directors, comprising INPO’s chief executive officer (CEO) and chief executives from INPO’s member organizations, oversees 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 following chart depicts INPO’s organizational structure.



Because the INPO board consists of 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, consisting of 10 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, global policy, and finance.

The Institute ensures that the industry actively participates in its programs and initiatives. Representatives from member utilities serve on an Executive Advisory Group and Industry Communications Council. The Executive Advisory Group, composed of chief nuclear officers from all member organizations, advises INPO management on nuclear technical areas and INPO's operations. The Industry Communications Council advises on the effectiveness of communicating INPO programs and activities, in addition to sharing best practices and lessons learned that elevate organizational alignment and key stakeholder engagement. The Institute also operates the National Academy for Nuclear Training (NANT). An Academy Council provides advice in the areas of accreditation, teaching and learning, and human performance. Additionally, 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 levels of safety and reliability in the operation of commercial nuclear power plants:

(1) *CEO engagement:* Fundamental 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 regulatory compliance informs the industry's continuous improvements in nuclear safety and reliability.

(3) *Broad industry support:* The nuclear industry plays a significant role in developing INPO standards of excellence and is committed to meeting those standards. Each member organization 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 focused 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 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.

3. 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 World Association of Nuclear Operators (WANO) Atlanta Center (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 its strategic Action for Excellence initiative) and coordinates many U.S. nuclear utility activities with WANO.

INPO's operating model is designed to 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 several hundred units that are operated internationally. The Institute works to remain 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, enhanced performance monitoring, and other WANO activities worldwide and invites international participants from other WANO members to take part in INPO activities.

- The Institute, as 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 all WANO-AC performance data and operating event information and shares this information with WANO. Likewise, INPO receives international event information and disseminates it to all of WANO-AC.

- 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 IAEA, the NEA, 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 high-level engagement during construction and initial startup to instill excellence standards among new entrants. Since 2019, WANO-AC members in the United Arab Emirates (Barakah Nuclear Energy Plant, Units 1 through 4) and China (Haiyang Nuclear Power Plant, Units 1 and 2, and Shidao

Bay Nuclear Power Plant) have successfully transitioned units from construction to operation. In the United States, Units 3 and 4 of the Vogtle Electric Generating Plant have begun operations. 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 organizations:

- Bruce Power (Canada)
- Centrala Nuclearelectrica (Romania)
- China Huaneng Group (China)
- Comisión Federal de Electricidad (México)
- Emirates Nuclear Energy Company (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

INPO/WANO member 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/WANO-AC 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 who handles that specific maintenance task particularly well.

The Institute provides the requesting utility with written reports detailing 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. Effective reviews reveal that the visits are highly valued by station leaders and that they contribute to improved performance.

4. 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 complementary goals, in that they both strive to protect the public by promoting safe and reliable plant operations. Consequently, their efforts in some important areas of nuclear plant performance, such as safety culture, overlap. However, INPO does not supplant 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 Federal Government. In recognition of their differing roles but common objectives, the NRC and INPO established a memorandum of agreement (MOA) 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 understand the quality of INPO products and programs. Consequently, the MOA 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 ROP uses operating data collected 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 NRC executives and 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.

5. Responsibilities of INPO and its Members

INPO member organizations are expected to strive for excellence in the operation of their nuclear plants to meet INPO PO&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, 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.

Members are expected to participate fully in other generic INPO programs designed to enhance industrywide nuclear plant safety and reliability. For example, they should provide INPO with detailed and timely operating experience information and should participate fully in 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 their nuclear operations, and not for other purposes such as obtaining competitive advantage. Members are to avoid including INPO or INPO documents in litigation.

Members of INPO also belonging 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 PO&Cs and other excellence standards are written with industry input and support, but without regard to utility-specific constraints or contracts, such as labor agreements. The Institute expects each member to resolve any barriers that outside organizations may impose on the implementation of INPO standards.

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.

6. Principles of Sharing (Openness and Transparency)

Throughout decades of change for the commercial nuclear power industry, including deregulation, marketplace dynamics, external factors such as the terrorist attacks of September 11, 2001, and the accident at Fukushima Dai-ichi, the industry has reaffirmed INPO's mission and methods. Even with U.S. utilities now in competition in certain locales, plant operators continue to value the sharing of pertinent operational experience and the provision of mutual support in areas impacting safety and reliability. Nuclear utility owners recognize that 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 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 website, written guidelines, and other publications.

Although 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, individual employee performance information, and power market data.

The INPO international strategy calls for it to provide all INPO Principles Documents, guideline documents, and IERs (which are discussed later) to WANO. In recent years, the extent of sharing has been curtailed due to increasingly complex U.S. export control regulations.

7. Commitment to a Strong Nuclear 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 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 from INPO 12-012, 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 IAEA safety culture characteristics.

The industry leverages INPO evaluations and other monitoring activities to identify and correct early signs of decline in the safety culture of a 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 evaluations and continuous monitoring. 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 evaluation team then discusses aspects of safety culture with the CEO of the utility at each evaluation exit briefing. Similarly, continuum leaders review safety culture during their continuous monitoring activities and continuum site visits (CSVs). The results of these reviews are included in the INPO Performance Summary Report (IPSR) to station leaders and are included in CSV exit briefings to the CEO.

8. INPO Strategic Design

INPO maintains an open-ended "Grand Strategy" containing every operation, activity, and/or action the Institute performs. In that way, everything INPO does remains visible and subject to routine continuous improvement.

From the Grand Strategy, INPO also derives a subset of priority corporate, industry, new nuclear, and international industry challenges that combine to define its more focused, long-term strategic design, typically defined with a 10-year timeframe. The current version of this began execution in 2021 and concludes in 2030.

The 2030 strategic design considers the current state of each of these challenges, the desired end states for each in 2030, and potential barriers (contributors) to achieving the desired outcomes in each of the four separate, but interrelated strategies. In executing the strategic design, priorities and measurable targets are defined to guide the application of INPO's limited resources to achieve and sustain strategic progress.

8.1 INPO Corporate Strategy

INPO's operating budget of about $110 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 corporate responsibilities encompass all the supporting capabilities for shaping U.S. and international industry performance—together with the traditional corporate tasks—to equip the INPO workforce with the knowledge, skills, and resources to execute its mission. INPO's 2030 corporate strategy includes the elements described below.

8.1.1 People

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; thus, INPO must maintain a workforce that is recognized as excellent in today's environment.

The Institute's permanent staff of about 305 full-time employees is augmented extensively by U.S. and some international industry professionals who serve as loaned employees for 18 to 24 months. Loaned and liaison employees make up 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 also 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 2025, the industry will provide INPO with an estimated 650 peers for short-term assignments.

8.1.2 Information Technology

The Institute employs a technology infrastructure commensurate with its mission and strategy requirements. Among its extensive information technology capabilities is software development for use by the industry to exchange data. Throughout 2024, the Institute conducted an extensive upgrade of its infrastructure to support new tools and viewing platforms to enable near real-time performance information to be exchanged with its members.

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.

8.2 INPO 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 2030 industry strategy currently addresses performance and sustainability challenges in four areas:

(1) *Performance outliers*: Foundational to a high-performing nuclear industry is the capability to identify and eliminate performance shortfalls that contribute to consequential events and raise risk. Whether it is broad based, isolated to specific units, or involve specific functional areas, low or outlier performance must be corrected wherever it exists.

(2) *Sustainable industry performance*: Underpinning a belief in continuous improvement in pursuit of performance excellence are five core values and underlying behaviors described in INPO 19-003, "Staying on Top—Advancing a Culture of Continuous Improvement," issued August 2019. Four values in particular (self-awareness/self-correction, excellence standards, leadership and talent development, and continuous learning) contribute most to a culture that ensures long-term sustainability.

(3) *Data science and analytics*: Industry demand is increasing for highly applicable Data-driven insights and visualizations from analysis and, modeling to support decision-making, risk management, and performance improvement. In this, the nuclear workforce must be equipped with the competencies to maximize their use of new data-informed tools and the insights they reveal.

(4) *Teaching and learning*: To accommodate a new generation of nuclear workers whose experience in and expectations of modern learning environments and techniques transcend traditional methods that characterize the nuclear industry today, NANT has broadened its approach to learning beyond just accredited training to incorporate innovative methods to teach and learn and raise proficiency up, down, and across the U.S. nuclear industry.

8.3 INPO International Strategy

INPO leverages and supports WANO to improve nuclear safety worldwide and to allow U.S. operators to benefit from worldwide operating experience. The Institute operates WANO-AC and conducts its operations and activities to meet or exceed WANO program requirements.

The Institute seeks to set the global example in its execution of WANO program guidelines, attainment of Action for Excellence strategic measures, and overall industry performance. In addition to supplying its intellectual capital to WANO for its use, the Institute provides direct assistance to other WANO regions in capabilities such as leadership development, continuous monitoring, and performance recovery.

8.4 INPO New Nuclear Strategy

In 2009, INPO established a means for collecting and distributing lessons and best practices from plants under construction through the Nuclear Network®. Nuclear Network® has long been the forum for rapid and secure communications and has hosted the very successful industry Operating Experience Program.

In 2023, INPO established a fourth strategy within its strategic design to ready itself for the likelihood of widespread new nuclear construction in the United States and with its WANO-AC members commencing in this decade. The strategy identifies three challenges that must be addressed, including recruitment of new nuclear operators and technology providers into the nuclear industry community, readying INPO's workforce to support new reactor technologies and operating concepts, and INPO's role in achieving success from construction to operations.

9. Operations, Activities, and Actions

In the execution of its strategic design, INPO conducts a spectrum of large-scale *operations*, such as plant evaluations and continuous monitoring; recurring *activities*, such as onsite accreditation reviews; and one-time *actions*, such as issuance of new standards. INPO categorizes these activities in four ways, including setting excellence standards, measuring and comparing performance and sustainability, facilitating performance improvement, and enacting self-regulatory authority.

9.1 Setting Excellence Standards

Excellence standards set expectations for industry performance beyond regulatory compliance. They come in three forms: PO&Cs, Principles Documents, and Level 1 and Level 2 IERs. In all cases, they are finalized and published by INPO but created in collaboration with industry members. In recent years, INPO and WANO jointly published an updated set of PO&Cs to ensure that evaluation standards for WANO Peer Reviews were aligned worldwide.

9.1.1 Performance Objectives and Criteria

PO&Cs provide a single framework of performance objectives and underlying criteria against which INPO and WANO evaluators assess 15 functional and cross-functional areas that, in combination, characterize nuclear station performance. Over the years, other new or updated standards have been incorporated into PO&Cs to consolidate industry excellence standards in one source, as much as possible.

9.1.2 Principles Documents

Principles Documents generally fill in performance areas that are gapped, or areas in which greater industry alignment is needed. For example, in 2015, INPO 15-005, "Leadership and Team Effectiveness Attributes," was published to align the industry around a common approach to leadership, teamwork, and organizational effectiveness. In another example, in 2019, INPO 19-003 was published to strengthen sustainability of performance. Most recently, in 2024, INPO published two new Principles Documents: INPO 24-001, "Proficiency, Advancing Human and Organizational Performance," and INPO 24-003, "Resiliency, Strengthening Defenses Against External Events," which target gaps in proficiency and resiliency, respectively.

When released, Principles Documents undergo considerable change management to ensure industry members fully understand what is expected in complying with these new excellence standards.

9.1.3 Level 1 and 2 INPO Event Reports

IERs generally address adverse trends in industry performance. Several trends emerged in the past decade that required decisive actions by INPO and the industry to reverse the trends. For example, in 2014, an adverse trend in engineering significant events resulted in publication of IER L1 14-20, "Integrated Risk—Healthy Technical Conscience," which began to immediately improve engineering performance. In 2016, another adverse trend in operations called for issuance of IER L1 17-05, "Line of Sight to the Reactor Core," which reversed the trend and continues to contribute to low numbers of operations events. INPO issued similar IERs for maintenance trends, fuel defect trends, and trends in equipment-related consequential events affecting plant reliability. Each set of IER recommendations established new performance standards that continue to apply today.

9.2 Measuring and Comparing Performance and Sustainability

Members host regular INPO plant evaluations in the form of WANO peer reviews and CSVs 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 PO&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.

9.2.1 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.

A combination of continuum leaders (e.g., monitoring leaders) and technical area points of contact continuously review and analyze performance data from all stations to identify subtle signs of decline. Additionally, a core team of assigned INPO subject matter experts continuously reviews and analyzes performance data pertaining to their specific functional areas. All continuum leaders collaboratively review performance twice per quarter. Additionally, INPO senior leaders review select stations multiple times per month. 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 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 industry feedback and a recognition that industry performance had significantly improved, 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 and to focus monitoring and assistance efforts on the organizations that pose the 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 performance over time together with the values outlined in INPO 19-003.

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 describes updated engagement categories—Monitoring, Augmented Monitoring, and Full Monitoring—which are defined as follows:

- *Monitoring*: Stations and corporations are 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*: Stations and corporations demonstrate exemplary or strong performance and behaviors that largely reflect a culture of improvement, but that also experience performance or sustainability gaps that require increased engagement and 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*: Stations and corporations 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 sustainability without a recovery plan that includes INPO and industry assistance. In specific cases, a station may be placed in Special Focus, a subset of Full Monitoring.

9.2.2 Continuum Site Visits

In 2021, INPO updated its operating model to adapt to a higher performing industry; move away from what was largely a "one size fits all" approach to performance improvement; increase transparency and collaboration; and demonstrate more agility and scalability in its operations. The result was a "continuum" of industry engagement that reduced the number of major operations from six to four, established a new set of engagement categories to reflect both performance and sustainability (and to inform INPO's resource allocations), and established the means for greater collaboration. In addition to strengthening collaboration within INPO, the Performance Continuum promotes higher levels of collaboration across the industry. Central to the continuum is the opportunity for INPO to leverage the wealth of data it collects, while applying updated analytics and visualizations to provide a continuous, near real-time picture of performance to its members.

As part of INPO's transition to the new operating model, it replaced Performance-Based Evaluations with CSVs to be conducted at the 2-year mark between WANO Peer Reviews. Similar to Performance-Based Evaluations, the size of INPO's CSV teams was determined by a station's performance. In this case, insights from continuous performance monitoring and ongoing engagement by the assigned INPO Continuum Leader combine to determine team composition. Base teams consist of the Continuum Leader, an organizational effectiveness leader, an INPO exit representative, and one industry host peer. CSVs also introduced a few minor differences in vernacular, including "areas of concern" to replace "areas for improvement."

Guiding principles for the CSV 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 numerical assessment (1–5) of station performance is determined.

- The utility CEO is informed of results at an exit meeting.

In addition to summarizing station performance, evaluation teams also provide a summary of "sustainability." Using INPO 19-003 as the standard to develop a culture of continuous improvement, the evaluation team assesses five values and underlying behaviors, summarizing each value in the formal report to the CEO.

The Institute's department managers also provide area performance summaries that assess current performance in each of 13 functional and cross-functional areas and trajectory, which forecasts near-term (6 months) future performance.

Subjective team comments are often communicated to the member CEO during the CSV exit meeting. The intent of these comments, which are often 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. Many 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 and extended 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.

9.2.3 WANO Peer Reviews

Historically, teams of approximately 18 to 25 qualified and experienced experts conduct WANO Peer Reviews 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, including—

- 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.

9.2.4 Corporate Peer Reviews

INPO recognizes that the corporate office and both nuclear and nonnuclear corporate leaders have a strong influence on safe and reliable nuclear operations. The Institute conducts corporate evaluations at 6-year intervals, with a followup performance review 2½ to 3 years after each corporate evaluation to verify progress on identified weaknesses.

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

PO&Cs and a Principles document, INPO 17-004, "Principles for excellence in corporate performance," issued October 2017, define the standards for assessing corporate performance. Areas typically evaluated include the following:

- 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 provide the leadership and support necessary to achieve and sustain excellent nuclear station performance. 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 also meets with utility boards of directors to provide an overview of industrywide, fleet, and station performance. The boards compare these briefings with views from independent oversight groups and corporate inputs to complete their assessments of nuclear performance.

9.2.5 National Academy for Nuclear Training: Training Accreditation

The NANT integrates training-related activities of all members, including accreditation of specific training programs that are overseen by an independent National Nuclear Accrediting Board. Each U.S. utility becomes a member of NANT when all its operating plants achieve accreditations for all applicable training programs.

The Institute evaluates accredited training programs, verifies 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 accreditation decisions. If training programs meet accreditation standards, the National Nuclear Accrediting Board awards or renews accreditation. If the program has significant problems, the 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 applicable training program every 6 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," codified in 10 CFR 50.120, "Training and qualification of nuclear power plant personnel," dated April 1993. To assist in maintaining its alignment with regulatory guidelines, the National Nuclear Accrediting Board enlists the participation of ex-NRC executives and permits the NRC to observe Boards and, upon request, join INPO teams that evaluate accredited training programs in the field.

9.3 Facilitating Performance Improvement

Many of INPO's operations, activities, and actions are aimed at assisting members in improving safety and reliability performance wherever and however the need arises. For many years, INPO has led technical teams to review and provide guidance to improve industrywide problem areas. INPO also serves as a clearinghouse for industry data, conducts analysis and trending, and shares lessons learned and best practices. In other cases, it provides direct support by applying its subject matter expertise and considerable industry experience. INPO also facilitates member-to-member assistance, both by leveraging industry peer experts to join INPO review or assist teams, and by functioning as an intermediary to obtain subject matter experts from one utility to assist another. Finally, NANT conducts courses and seminars to help educate and train all levels of industry leadership.

As a means of communicating the above, the Institute frequently sponsors workshops and working meetings for specific groups of industry managers. These allow INPO and its member participants to meet and exchange information broadly. In an average year, more than

4,000 industry personnel participate in more than 100 seminars, workshops, and technical working meetings at INPO.

Another means of communication to the industry are INPO documents and other written products. They are organized by “tiers” to differentiate their application:

- *Tier 1*—Excellence Documents: As previously discussed, these are PO&Cs, Principles Documents, and Level 1 and Level 2 IERs. They constitute the highest tier of INPO communications and are released by INPO’s CEO.

- *Tier 2*—Supporting and Implementing Documents: These include Level 3 IERs and Guidelines that provide additional guidance and detail considered necessary to fully implement objectives and criteria but stop short of prescribing specific methods or processes. Additionally, 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 follow defined processes.

- *Tier 3*—Other Documents: These documents provide references or amplifying information on various topics for review and discretionary use by INPO members and participants. In some cases, 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. Examples include Level 4 IERs; Good Practices, which provide examples of effective methods for accomplishing elements of nuclear plant management and operation; manuals, which are collections of data or other information for use by INPO members and participants; and reports, which provide descriptions and results of INPO or INPO-sponsored activities that may be of interest to the industry, including benchmarking information, cumulative analyses of industry events, and other information that does not fall into a specific document category.

Finally, *Operating Experience Program Descriptions* provide an overview of the INPO-sponsored Operating Experience Program and its expectations for INPO and INPO members.

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.

9.3.1 Technical Review Visits

For many years, INPO has teamed with industry subject matter experts to conduct technical review visits in select industrywide challenge areas. Teams are led by INPO and typically spend a week in preparation followed by a week at a station that has experienced the technical issue. Examples of areas reviewed include materials issues that could affect the structural integrity of the reactor coolant system and vessel internals or systems that contribute significantly to unplanned transients and forced outages. Review visits have sometimes led to detailed technical guidance for an affected utility or for the industry as a whole.

Pressurized-Water Reactor Materials Reviews

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 EPRI 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 asked INPO to 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 system 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 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. 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 this review visit scope and team size are larger, the objective remains the same: ensuring that nuclear safety and plant reliability are not compromised by weaknesses in the primary pressure boundary, including steam generators.

In 2016, the scope of the PWR Materials Review Visit Program was adjusted to look at other aspects of materials degradation, including flow-accelerated corrosion and buried pipe and tank integrity.

Boiling-Water Reactor Materials Reviews

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.

The BWR Vessel and Internals Review Visit Program focuses 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.

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 member corporate offices, consolidating the reviews of BWRs and PWRs.

Main Generator Reviews

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 more 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.

Alternating Current Power Source Reliability Reviews

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 power reliability. Three to five loss-of-offsite power matrix reviews are targeted per year, prioritized on a performance basis. These reviews, termed alternating current 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 programs 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, alternating current power reliability will remain an industry focus area for 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 alternating current 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 alternating current Power Reliability INPO Focus Area was transitioned to monitoring status in 2018.

The Institute also actively partners with the North American Transmission Forum (NATF) to develop common expectations and risk assessment tools for the switchyard and grid system interface. In 2014, INPO, the NATF and EPRI began joint efforts focused on alternating current power reliability, and those efforts continue with periodic alignment meetings to share lessons learned. The Institute also partnered with EPRI in the industry's Flexible Power Operations initiative and developed guidance for plants requested to accommodate renewable resource power contributions to grid load demand. Institute personnel remain engaged in industry working groups and in emergent plant issues related to main generator, turbine, and support systems.

Operator Fundamentals Reviews

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. As a result of the review visits and a related IER, operator fundamentals have sustainably improved.

Fuel Integrity Reviews

The Institute performed Fuel Integrity Review Visits in 2017 and 2018 to gather detailed information on 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 on the causes of the station's fuel rod failures and the corrective actions being taken by each station organization. Recommendations and beneficial practices were identified and documented.

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. In 2019, INPO issued a Level 2 IER that established a new standard for eliminating debris-induced fuel failures.

In 2024, INPO partnered with the industry fuel suppliers to conduct assistance visits at the fuel fabrication facilities. Two visits were conducted in 2024, and one more is planned in 2025 with the purpose of improving fuel reliability.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.

9.3.2 Information Sharing

Station organizations are required to input data and share operating experience and lessons learned with INPO. In turn, The Institute analyzes the data to measure performance, detects industry trends, and communicates findings back to the industry through various means. The Institute also provides members ready access to information it receives on plant and equipment performance and operating experience utilizing a secure, member-only website.

Data Collection and Analysis

The Institute operates and maintains the Industry Reporting and Information System (IRIS) as the central repository for data and information related to nuclear plant performance, including equipment performance. On a monthly basis, members provide routine operational data in accordance with INPO, WANO, and NRC program requirements. Plant data then undergo extensive analysis; are input to indicators, indexes, and models; and provide INPO the means to determine and track performance trends. Members can access these data to monitor performance of their plants, track progress toward performance goals, and compare their performance with other nuclear stations.

In agreement with its members and in accordance with an INPO/NRC MOA, the Institute also makes a subset of these data available to the NRC to support the NRC's regulatory oversight process, equipment performance reviews, and other requirements.

In the past decade, INPO's modeling techniques have grown in sophistication, such that AI methods have been used to improve the capabilities and accuracy of its information. As early as 2014, INPO began to employ the Plant Performance Indicator, which soon began to use neural networks in its formulation. INPO is currently conducting an array of "use cases" leveraging more powerful AI capabilities to advance its analysis and modeling programs.

Operating Experience Program

The Institute employs an Operating Experience Program to review and analyze events reported by domestic and international stations. The purpose of the program is to enable in-depth analysis of events as they are reported and then to communicate lessons learned across the industry. Upon receipt, events are screened, tagged, and analyzed for significance. Those with broad applicability may then be disseminated to the industry in the form of the following IERs:

- Level 1 IER: This provides recommended actions based on one or more significant industry events, a critically important industry issue, or an adverse trend. Level 1 IER recommendations constitute a new industry excellence standard for which compliance is mandatory.

- Level 2 IER: This highlights an area of concern based on one or more industry events or an adverse trend that has broad applicability to the industry, which may or may not derive from significant events but is considered consequential to plant safety or reliability. Level 2 IER recommendations also constitute excellence standards for which compliance is mandatory.

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

- Level 4 IER: This provides analysis of notable trends of equipment or human performance challenges 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 which is then shared among INPO members through

IRIS. INPO applies a 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 of 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 validates the effectiveness of the station's actions in response to these recommendations. Members should also review and take action, as appropriate, on Level 3 and 4 IERs. In all, the Institute evaluates each member's effectiveness in fully exploiting the Operating Experience Program to extract and apply industry lessons learned.

The Institute maintains all operating experience reports on the secure member website. This enables members to leverage historical lessons learned as well as real time event reporting.

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 then communicates the results of analyses and suggested actions to the industry. For example, in 2024, an adverse scram trend was extensively analyzed, and equipment contributors pointed to four specific systems, including feedwater, turbine generator, and main transformer systems, that called for extensive corrective actions that transcended local station actions and included corporate and vendor responses to arrest the trend.

9.3.3 National Academy for Nuclear Training: Teaching and Learning

Beginning in 2021, the NANT embarked on a strategic initiative to institute cutting edge, state-of-the-art teaching and learning techniques and initiatives across the nuclear industry. In part, this was in response to the changing demographics of the nuclear workforce, which includes a younger generation whose teaching and learning experiences are very different from the industry's traditional, longstanding classroom approaches. This was also intended to expand education and training beyond accredited training, which for many years dominated the nuclear industry's approaches to learning. Finally, the initiative was intended to more fully exploit the potential of NANT, in concert with other organizations that contribute to industry learning, such as the NEI, EPRI, and others.

In 2024, INPO revitalized its Academy Council, which serves as an advisory body to NANT. In addition to renewed focus on teaching and learning beyond accreditation, the membership of the Council has been adjusted to include organizations with the knowledge and ability to affect nuclear industry education and training.

9.3.4 Courses and Seminars

Within INPO's headquarters in Atlanta, Georgia, NANT conducts a tiered series of courses and seminars focused on development of leadership skills for first-line supervisors to senior executives. It also sponsors the Reactor Technology Course in partnership with Massachusetts Institute of Technology for nonnuclear utility executives and a course in partnership with Emory University in Atlanta targeting newly appointed directors to nuclear utility boards.

As a result of impacts from the COVID-19 pandemic in 2020, NANT developed virtual courses on topics such as coaching, decision-making, and managing conflict. These programs have continued and have enabled INPO to influence industry members more distantly and more broadly.

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

9.4 Self-Regulatory Authority

The final category of operations, activities, and actions within INPO's operating model defines its responsibilities to exert self-regulatory authority when it must. Since its inception, INPO's authority has been conferred to its CEO by U.S. nuclear utility CEOs through the INPO Board of Directors. What this entails is outlined best in INPO Policy Note 14. Throughout WANO-AC, utilities and stations outside of the United States volunteer to adhere to the tenets of INPO Policy Note 14.

Institute authorities range from its responsibility to assess station performance by assigning scores of 1 through 5, to its management of the Performance Continuum and responsibility to categorize stations in Monitoring, Augmented Monitoring, or Full Monitoring, to its ability to designate "Special Focus" stations as a subcategory of Full Monitoring and to require performance recovery actions. In other words, an expectation that industry members not just participate but comply with the tenets of INPO's operating model is the manifestation of the Institute's self-regulatory authority. In the past, noncompliance has been dealt with CEO to CEO, through "for cause" engagement with utility boards of directors, or utility presence before the INPO Board of Directors. Options for dealing with a member's noncompliance are more fully outlined in an INPO Policy Note approved by the Board of Directors.

10. Conclusion

The U.S. commercial nuclear industry has made substantial, quantifiable, and sustained improvements in plant safety and reliability throughout more than four decades of INPO's existence. The leaders who have guided the U.S. industry over that timeframe showed great insight in recognizing, not just the need for a form of self-regulation, but the benefits of empowering INPO to execute its comprehensive programs and expect compliance from all its members. In this commitment, U.S. industry CEOs acknowledge that nuclear energy will remain a viable contributor to electric generation only if utilities pursue the highest levels of nuclear safety and reliability—excellence—in nuclear plant operation.

Today, the U.S. nuclear industry appears to be on the cusp of recapitalization as unprecedented demands for clean energy increase and the benefits of nuclear power become more broadly recognized and communicated. Multiple initiatives, such as license extensions, power uprates, restarting units previously shut down, and new nuclear construction, are likely to converge with ongoing efforts to continuously improve legacy industry performance and sustainability, testing industry leadership and INPO like never before. In all this, the single nonnegotiable aspect is the continued high performance of the current industry. For INPO and its members, and the NRC and its licensees, nothing is more important.

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APPENDIX B - U.S. COMMERCIAL NUCLEAR POWER REACTORS

SOURCE: U.S. Nuclear Regulatory Commission, NUREG-1350, "2022–2023 Information Digest," Volume 34, February 2023.

NOTE: Since the issuance of the 2022 U.S. National Report, two units started operation (Vogtle Electric Generating Plant, Units 3 and 4), bringing the total to 94 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—Constellation Energy Generation Co., LLC	PWR	3,645	07/88–10/46
Braidwood Station, Unit 2—Constellation Energy Generation Co., 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—Constellation Energy Generation Co., LLC	PWR	3,645	09/85–10/44
Byron Station, Unit 2—Constellation Energy Generation Co., LLC	PWR	3,645	08/87–11/46
Callaway Plant, Unit 1—Union Electric Company	PWR	3,565	12/84–10/44
Calvert Cliffs Nuclear Power Plant, Unit 1—Calvert Cliffs Nuclear Power Plant, Constellation Energy Generation Co., LLC	PWR	2,737	05/75–07/34

Plant Name and Licensee	Reactor Design Type	Licensed Power (MWt)	**Operating Lifetime**
Calvert Cliffs Nuclear Power Plant, Unit 2—Calvert Cliffs Nuclear Power Plant, Constellation Energy Generation Co., 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—Constellation Energy Generation Co., LLC	BWR	3,473	11/87–09/26
Columbia Generating Station—Energy Northwest	BWR	3,544	12/84–12/43
Comanche Peak Nuclear Power Plant, Unit 1—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—Constellation Energy Generation Co., LLC	BWR	2,957	06/70–12/29
Dresden Nuclear Power Station, Unit 3—Constellation Energy Generation Co., 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
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

Plant Name and Licensee	Reactor Design Type	Licensed Power (MWt)	**Operating Lifetime**
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—Constellation Energy Generation Co., 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—Constellation Energy Generation Co., LLC	BWR	3,546	01/84–04/42
La Salle County Station, Unit 2—Constellation Energy Generation Co., LLC	BWR	3,546	10/84–12/43
Limerick Generating Station, Unit 1—Constellation Energy Generation Co., LLC	BWR	3,515	02/86–10/44
Limerick Generating Station, Unit 2—Constellation Energy Generation Co., 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	BWR	1,850	12/69–08/29
Nine Mile Point Nuclear Station, Unit 2—Nine Mile Point Nuclear Station, 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
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

Plant Name and Licensee	Reactor Design Type	Licensed Power (MWt)	**Operating Lifetime**
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—Constellation Energy Generation Co., LLC	BWR	4,016	07/74–08/53
Peach Bottom Atomic Power Station, Unit 3—Constellation Energy Generation Co., 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—Constellation Energy Generation Co., LLC	BWR	2,957	02/73–12/32
Quad Cities Nuclear Power Station, Unit 2—Constellation Energy Generation Co., 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
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

Plant Name and Licensee	Reactor Design Type	Licensed Power (MWt)	**Operating Lifetime**
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
Vogtle Electric Generating Plant, Unit 3—Southern Nuclear Operating Company Inc.	PWR	3,400	08/22–08/62
Vogtle Electric Generating Plant, Unit 4—Southern Nuclear Operating Company Inc.	PWR	3,400	07/23–07/63
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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Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above", if contractor, provide NRC Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address.)

Same as above

10. SUPPLEMENTARY NOTES

This report is an update to NUREG-1650, Revision 8

11. ABSTRACT (200 words or less)

The U.S. Nuclear Regulatory Commission has prepared Revision 9 to NUREG-1650, “The United States of America Tenth National Report for the Convention on Nuclear Safety,” for submission for peer review at the tenth review meeting of the Convention on Nuclear Safety, to be convened at the International Atomic Energy Agency in Vienna, Austria, in April 2026. 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 2022, 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.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)	13. AVAILABILITY STATEMENT unlimited
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