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

September 2004

U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

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THE UNITED STATES OF AMERICA

THIRD NATIONAL REPORT

FOR THE

CONVENTION ON NUCLEAR SAFETY

SEPTEMBER 2004

U.S. NUCLEAR REGULATORY COMMISSION

WASHINGTON, DC 20555-0001

ABSTRACT

The United States (U.S.) Nuclear Regulatory Commission has updated the 2001 *U.S. National Report for the Convention on Nuclear Safety* and will submit this report for peer review at the third Review Meeting of the Convention at the International Atomic Energy Agency in April 2005. The scope of this report is limited to the safety of land-based commercial nuclear power plants. The report demonstrates how the U.S. Government meets the main objective of the Convention — to achieve and maintain a high level of nuclear safety worldwide by enhancing national measures and international cooperation. It also shows how the U.S. Government meets the obligations of each of the articles established by the Convention. Specifically, those articles address the safety of existing nuclear installations, the legislative and regulatory framework, the regulatory body, responsibility of the licensee, priority to safety, financial and human resources, human factors, quality assurance, assessment and verification of safety, radiation protection, emergency preparedness, siting, design, construction, and operation.

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EXECUTIVE SUMMARY

The United States (U.S.) Nuclear Regulatory Commission (NRC) has updated the *U.S. National Report for the Convention on Nuclear Safety* and will submit this report for peer review at the third Review Meeting of the Convention on Nuclear Safety (Convention) hosted by the International Atomic Energy Agency (IAEA) in April 2005. This updated report follows the same format as the previous *U.S. National Report*, which the Commission submitted in 2001, and the scope of the report remains limited to the safety of land-based commercial nuclear power plants. In particular, this report demonstrates how the U.S. Government meets the main objective of the Convention — to achieve and maintain a high level of nuclear safety worldwide by enhancing national measures and international cooperation. It also shows that the U.S. Government meets the obligations of each of the articles established by the Convention. Specifically, those articles address the safety of existing nuclear installations, the legislative and regulatory framework, the regulatory body, responsibility of the licensee, priority to safety, financial and human resources, human factors, quality assurance, assessment and verification of safety, radiation protection, emergency preparedness, siting, design, construction, and operation.

This updated report discusses the national nuclear programs that the United States uses to meet the obligations of the Convention, the most significant current nuclear safety issues, and significant regulatory accomplishments since the previous report was written in 2001. This report also highlights NRC's updated strategic goals, which enable the management and use of radioactive materials and nuclear fuels for beneficial commercial purposes in a manner that protects public health and safety and the environment, promotes the security of the United States, and provides for regulatory actions that are effective, efficient, and open. In addition, this report describes many activities that NRC has initiated to meet those goals.

NRC and its licensees use a number of national programs and processes to protect the health and safety of the public and the environment and to meet the obligations of the Convention. Key programs in the reactor arena comprise a well-established licensing process, which includes power uprates, new reactor licensing and early site permits, reactor oversight, and license renewal. Other programs include the industry trends program; the accident precursor program, the program for resolving generic safety issues; programs for rulemaking, decommissioning, and regulatory research; and programs for public participation and for handling petitions, allegations, and differing opinions of NRC staff.

The main nuclear safety issues that NRC and its licensees are currently addressing involve reactor materials, pressurized-water reactor containment sump performance, electric grid reliability, emergency preparedness, and security.

At the second Review Meeting, the contracting parties concluded that the United States complied with the provisions of the Convention. They also determined that U.S. experience with the further development and application of its regulatory process in the four areas listed below would be of high interest for discussion at the 2005 Review Meeting:

- Risk-Informed Regulation, including the use of safety goals
- Performance-Based Reactor Oversight Process

- License Renewal, and its comparison with the Periodic Safety Reviews of many countries
- Licensing of New Reactors

Accordingly, this report specifically discusses each of these areas. In addition, safety culture is of increasing international interest, and this report discusses the United States' approach for addressing safety culture-related issues.

INTRODUCTION

This section describes the purpose and structure of the *U.S. National Report for the Convention on Nuclear Safety*, the United States' national policy toward nuclear activities, the main national nuclear programs, and the current nuclear safety issues. It then highlights major regulatory accomplishments since the previous *U.S. National Report* was written in 2001. In addition, it references Annex 1 to this report, which lists the nuclear installations in the United States.

Purpose and Structure of this Report

This updated report is the submission by the United States of America at the third Review Meeting of the Contracting Parties to the Convention on Nuclear Safety for peer review. [This meeting is scheduled to be held at the IAEA in Vienna, Austria, in April 2005.] The scope of this report considers only the safety of land-based commercial nuclear power plants, consistent with the definition of nuclear installations provided in Article 2 and the scope of Article 3 of the Convention.

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

(v) 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 in order 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 (which, in this report, is referred to as NRC, Commission[a], agency, or staff) updated the original *U.S. National Report*, principally using agency information that is publicly available. This updated report follows the format of the previous *U.S. National Report for the Convention on Nuclear Safety*, submitted in 2001, and is designed to be a "stand alone" document to facilitate peer review. Some information from the 2001 report was not repeated because of the level of detail, and readers are referred to that report for such information. This report begins with an introduction and a new section on conclusions of the previous Review Meeting. It then continues with Articles 6–19, and includes annexes and references to provide more detailed information where appropriate. 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 by the chapter, and a discussion of how the United States meets the obligations of the article.

[a] "Commission" may also refer to the Chairman and Commissioners who head NRC.

Articles 6–9 summarize the legislative and regulatory system governing the safety of nuclear installations and discuss the adequacy and effectiveness of that system. Articles 10–16 address general safety considerations and summarize major safety-related features. Articles 17–19 address the safety of installations. This report does not include chapters for Articles 1–5. In accordance with Article 1, the report illustrates how the U.S. Government meets the objectives of the Convention. It 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 the report fulfills the obligation of Article 5 on reporting. In addition, the information in this report is available in more detail through NRC's public Web site and, accordingly, uniform resource locators (URLs) are generously given.

The U.S. National Policy Toward Nuclear Activities

NRC was created through enactment of the Energy Reorganization Act of 1974 by the U.S. Congress. As such, NRC is an independent agency of the Federal Government. The mission of NRC is to license and regulate the Nation's civilian use of byproduct, source, and special nuclear materials to ensure adequate protection of public health and safety, promote the common defense and security, and protect the environment. The agency also has a role in combating the proliferation of nuclear materials worldwide. NRC's safety and security responsibilities stem from the Atomic Energy Act of 1954. NRC accomplishes its mission by licensing and overseeing nuclear reactor operations and other activities that apply to the possession of nuclear materials and wastes, ensuring that nuclear materials and facilities are safeguarded from theft and radiological sabotage, issuing rules and standards, inspecting nuclear facilities, and enforcing regulations.

NRC views nuclear regulation as the public's business and, as such, it must be transacted openly and candidly to maintain the public's confidence. The goal to ensure openness explicitly recognizes that the public must be informed about, and have a reasonable opportunity to participate meaningfully in the regulatory process. Except for certain vendor-proprietary business material, facility safeguards information, sensitive predecisional information, certain sensitive, unclassified information that could be useful to a potential terrorist, and information supplied by foreign countries that is deemed to be sensitive, the documentation that NRC uses in its decisionmaking is available in the agency's Public Document Room in Rockville, Maryland, and on the agency's public Web site at http://www.nrc.gov. As a result, nuclear activities and the national policy toward them are open to everyone.

NRC continues to evolve from a prescriptive, deterministic approach toward a more risk-informed and performance-based regulatory approach. Improved probabilistic risk assessment techniques, combined with more than four decades of accumulated experience with operating nuclear power reactors, have led the Commission to revise or eliminate certain requirements. The Commission is also prepared to strengthen the regulatory system where risk considerations reveal the need.

National Nuclear Programs

NRC has a number of national programs and processes to protect the health and safety of the public and the environment and to meet the obligations of the Convention. Key programs and processes in the reactor arena comprise a well-established licensing process, which includes power uprates, new reactor licensing and early site permits, reactor oversight, and license renewal. These programs are described below. Other programs include the industry trends program; the accident precursor program; the program for resolving generic safety issues; programs for rulemaking, decommissioning, and regulatory research; and programs for public participation and for handling petitions, allegations, and differing professional opinions of NRC staff. For details on these programs, see Article 6.

Power Uprate Program

NRC carefully reviews power uprate requests — requests to raise the maximum thermal power level at which a plant may be operated. Improvements of feedwater flow rate instrument accuracy and modifications of plant hardware have allowed licensees to submit power uprate applications for NRC review and approval. The focus of NRC review of these applications has been, and will continue to be, on safety. NRC continues to closely monitor operating experience to identify issues that may affect power uprate implementation.

Requests for power uprates range from small increases based on the recapture of feedwater flow measurement uncertainty, to large increases in the range of 15 to 20 percent. Large increases typically require substantial hardware modifications at the plants. In all instances, NRC must be satisfied that safety margins are maintained. As of August 2004, NRC has approved more than 100 power uprates, which have added approximately 4,179 megawatts electric — the equivalent of about four large nuclear power plants — to the Nation's electric generating capacity.

As of August 2004, NRC has ten power uprate applications under review and expects to receive an additional 18 applications by the end of calendar year 2008. Approval of these applications would add about 1,904 megawatts electric to the Nation's electric generating capacity by calendar year 2009.

Recognizing licensees' increased interest in power uprates, NRC recently issued a review standard for extended power uprates (i.e., uprates that increase the current power by seven percent or more). This document, which is publicly available, provides a plan that establishes standardized review guidance and acceptance criteria for both NRC and its licensees. As such, the review standard enhances NRC's focus on safety and improves the consistency, predictability, and efficiency of its reviews and will foster improved communication with NRC stakeholders and licensees.

NRC is monitoring operating experience at plants that have implemented power uprates. Steam dryer cracking and flow-induced vibration damage affecting components and supports for the main steam and feedwater lines have occurred at some of these plants. NRC conducted inspections to identify the causes of these issues and evaluated many of the repairs performed by the licensees. To date, the staff has determined that these issues do not pose an immediate safety concern. The staff continues to monitor the industry's generic response to these issues and will consider additional regulatory action, as appropriate.

Article 6 describes the power uprate program in more detail.

New Reactor Licensing

Although improved performance of operating nuclear power plants and power uprates has resulted in increases in electrical output and generating capacity, any long-term increase in the demand for electricity is expected to be addressed by construction of new generating capacity. Consequently, the industry has recently expressed interest in new construction of nuclear power plants in the United States. NRC is ready to accept applications for new nuclear power plants. New nuclear power plants may use the licensing process specified in Title 10, Part 52, of the *Code of Federal Regulations* (10 CFR Part 52), which is considered more stable and predictable than that specified in 10 CFR Part 50. This process ensures that most safety and environmental issues, including emergency preparedness and security, are resolved before the construction of a new nuclear power plant. The design certification part of the process resolves the safety issues related to plant design; the early site permit part of the process resolves safety and environmental issues related to a potential site. The issues resolved in these two parts can then be referenced in an application that would lead to a combined construction permit and operating license with conditions (combined license).

The Commission has already certified three new reactor designs pursuant to 10 CFR Part 52, making them readily available for new plant orders. These designs include General Electric's Advanced Boiling Water Reactor and Westinghouse's AP600 and System 80+ designs.

In addition to the three advanced reactor designs already certified, the Commission is currently reviewing the Westinghouse AP1000 design certification application. The staff expects to issue its recommendation regarding the final design approval for the AP1000 design to the Commission in the fall of 2004, followed by the design certification rulemaking in 2005. The NRC staff is also actively reviewing pre-application issues concerning two additional designs and has four other designs in various stages of pre-application review.

In September and October 2003, NRC received three early site permit applications for sites in Virginia, Illinois, and Mississippi where operating reactors already exist. NRC review of these early site permits, if approved, would be the first time this part of the licensing process in 10 CFR Part 52 has been implemented. The staff has established schedules to complete the safety reviews and environmental impact statements in approximately two years. The mandatory adjudicatory hearings associated with the early site permits will be concluded after completion of the NRC staff's technical review. As with the design certification rulemaking, issues resolved in the early site permit proceedings will not be revisited during a combined license proceeding absent new and compelling information.

To prepare for the potential application for a combined license, the Commission is hosting discussions with the U.S. nuclear power industry and other stakeholders to resolve generic issues related to the preparation and review of a combined license application.

NRC is updating its construction inspection program to address the combined license process. As part of that effort, in 2003, NRC published a framework document that discusses the various designs and the construction inspection program activities. Modern design techniques, modular construction, and international suppliers and components raise a number of scheduling and technical challenges. The staff is taking an integrated approach to determine the infrastructure necessary to conduct inspections to support the regulatory decisions that will be needed to license an advanced plant. The staff is informing its approach by considering lessons-learned from foreign counterparts with more recent construction experience. Like plants in other countries, U.S. plants will be built with components from many countries, manufactured by international companies. Consequently, vendor inspection staff have been actively involved in examining standards that are used to fabricate and qualify equipment made abroad. For information on work to harmonize international standards, see Article 13.

Reactor Oversight Process

The NRC's Reactor Oversight Process is now nearly four years old. In its annual self-assessment for calendar year 2003, the staff concluded that the Reactor Oversight Process was generally effective in monitoring operating nuclear power plant activities and focusing NRC resources on significant performance issues; continued to support the agency's performance goals; and remained effective in meeting NRC's program goals of being objective, risk-informed, understandable, and predictable. In addition, there were no statistically significant adverse trends identified in any industry-level performance indicators. However, the reactor vessel head degradation at the Davis-Besse Nuclear Power Station continues to cause a focused look at NRC's oversight efforts and has resulted in several program improvements. (The Davis-Besse event is explained in more detail in Articles 6 and 10). Although significant progress has been made, the staff expects to make continued improvements to the Reactor Oversight Process based on recommendations of the Davis Besse Lessons-Learned Task Force and other stakeholders. The staff also plans to continue to actively solicit input from NRC's internal and external stakeholders to further improve the Reactor Oversight Process, and will continue to evaluate program improvements under the ongoing reactor oversight self-assessment process.

A recent development is that NRC has determined that certain security information formerly included in the Reactor Oversight Process will no longer be publically available, and will no longer be updated on the agency's website. In so determining, the agency balanced a commitment to openness and a concern that sensitive information might be misused by those who wish the country harm. NRC will continue to inspect and assess physical security of nuclear facilities, but will no longer make the results publically available and will exempt them from Freedom of Information Act requests. It will also withhold enforcement information associated with the physical protection of nuclear facilities. However, NRC will continue to provide these types of information to state officials, local law enforcement agencies and other federal agencies.

License Renewal

The focus of the Commission's review of license renewal applications is on maintaining plant safety, with the primary emphasis on the effects of aging on important structures, systems, and components. The review of a renewal application proceeds along two paths — one for the review of 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. With the improved economic conditions for operating nuclear power plants, the Commission has seen sustained strong interest in license renewal, which allows plants to operate up to 20 years beyond their original 40-year operating licenses. The original 40-year term was established in the Atomic Energy Act and was based on financial and antitrust considerations, rather than technical limitations.

The decision to seek license renewal is voluntary and rests entirely with nuclear power plant owners. The decision is typically based on the plant's economic viability and whether it can continue to meet the Commission's requirements. As of August 2004, NRC has issued renewed licenses for 15 sites, totaling 26 units. The staff is currently reviewing applications to renew the licenses for an additional nine sites (totaling 18 units). If the Commission approves all of the applications that are currently under review, approximately 40 percent of the plants in the United States will have had their operating licenses renewed. Judging by statements from industry representatives, the Commission expects virtually all sites to apply for license renewal.

The Commission has established a license renewal process that can be completed in a reasonable period of time with clear requirements to ensure safe plant operation for up to an additional 20 years of plant life. To help achieve consistency in the preparation and review of renewal applications, the Commission has issued guidance documents to assist plant owners in preparing license renewal applications and to guide the NRC staff's review of the applications. Lessons learned from ongoing reviews are documented as they are identified and made publicly available for use by future applicants. These guidance documents provide the framework for an effective, efficient, and technically sound review of renewal applications and help maintain the stability and predictability of the license renewal process. An additional benefit of the license renewal program is that licensees seeking license renewal are willing to spend significant money or upgrade equipment which improves the overall safety of the plant.

Public participation is an important part of the license renewal process. The Commission provides several opportunities for members of the public to question how aging will be managed during the period of extended operation. Concerns may be litigated in an adjudicatory hearing if an adversely affected party appropriately requests a hearing. The staff has also developed a license renewal Web site, which contains key documents associated with license renewal applications as well as information on the license renewal process, regulations, and guidance documents. Although the license renewal program has been highly successful, the Commission continues to seek further improvements in the process. Using lessons learned from past reviews, the Commission is pursuing revisions to the renewal process that should yield additional efficiencies. These efficiencies will help the Commission better accommodate the increasing number of renewal applications being submitted.

Survey of Main Current Safety Issues

NRC and its licensees currently face the following regulatory and safety issues:

- reactor materials issues
- pressurized water reactor containment sump performance
- electric grid reliability
- emergency preparedness and security

Reactor Materials Issues

The current reactor materials issues are primary water stress corrosion cracking in pressurized water reactor vessel upper and lower head penetrations and other locations in the reactor coolant system, as well as boric acid corrosion. Control rod drive mechanism nozzles and other vessel head penetration nozzles welded to the upper reactor vessel head are susceptible to primary water stress corrosion cracking. This susceptibility is a potential safety concern because a nozzle that is sufficiently cracked could break off during operation, compromise the integrity of the reactor coolant system pressure boundary, and possibly cause a control rod to eject.

In March 2002, the licensee for Davis-Besse Nuclear Power Station discovered a significant cavity in the reactor vessel head. The cavity was next to a leaking nozzle with a through-wall crack and in an area of the vessel head that had been covered with boric acid deposits for several years. NRC considers the Davis-Besse case as one of the most significant in the agency's recent history. Davis-Besse has drawn much interest and comment from the agency, the industry, all levels of government, and the public. Consequently, NRC has dedicated much effort to resolving both the technical and programmatic issues that contributed to the degradation. Davis-Besse is discussed in further detail, including steps NRC has taken in Articles 6 and 10. (For history and details, see the extensive documentation on NRC's public Web site.)

Another significant case concerns lower reactor vessel head penetration nozzles. In April 2003, the licensee for South Texas Project Unit 1 discovered small boron deposits around two of the unit's bottom-mounted instrumentation penetration nozzles during a bare metal visual examination of the reactor pressure vessel bottom head. Subsequent nondestructive examination of all 58 nozzles on the bottom head confirmed the existence of leaking, axially-oriented flaws in the two nozzles. As a result of the events at South Texas Project Unit 1, the NRC staff issued Bulletin 2003-02, "Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity," on August 21, 2003.

Pressurized-Water Reactor Containment Sump Performance

Research performed to resolve the strainer plugging issue in boiling-water reactors (BWRs) in the late 1990s led NRC to question the adequacy of sump designs in pressurized-water reactors (PWRs), and to consider the issue generically. The issue is that post-accident debris blockage can impede or prevent the operation of the emergency core cooling system and containment spray system in the sump recirculation mode. Subsequent NRC research concluded that recirculation sump clogging is a credible concern for some PWRs. However, owing to the limitations of plant-specific data and other modeling uncertainties, the research results do not definitively identify whether a particular PWR plant is vulnerable to sump clogging.

In June 2003, NRC issued a bulletin to inform PWR licensees of the results of NRC-sponsored research and to request that licensees either confirm compliance with the regulations regarding long-term cooling and other applicable regulatory requirements, or describe any compensatory measures taken to reduce the potential risk from post-accident debris blockage, while performing evaluations to determine compliance. NRC is currently reviewing each licensee's response to the bulletin, considering international experience, and preparing a generic letter to resolve the issue of PWR sump clogging in the long-term. The generic letter will ask licensees to provide the status of their compliance with the regulations, including their plans for performing analyses and containment walkdowns, and the methodologies used to justify their conclusions. Other ongoing NRC activities to resolve this issue include reviewing and endorsing an industry guidance document on sump clogging, and performing research to characterize chemical and corrosion effects on sump performance.

Electric Grid Reliability

The blackout in the eastern United States and Canada on August 14, 2003, highlighted the need to further consider the impact of grid reliability on nuclear plants, primarily because of its long duration. The blackout affected more than 290 commercial power facilities, including ten nuclear plants, of which nine tripped offline and were subsequently offline for an extended period of time. (Davis-Besse, one of the ten, was already shut down at the time of the blackout.) The affected nuclear plants were returned to service by August 22, 2003. Before this blackout, NRC had been researching the importance of grid stability on the safe operation of nuclear power reactors. Grid stability is important to reduce the chance of a loss of offsite power that could potentially challenge a plant's ability to remove decay heat. Although plants are designed for these occurrences with backup power supplied by emergency diesel generators, a loss of offsite power would reduce a plant's safety margin.

NRC provided extensive technical support to a joint U.S. - Canada Power System Outage Task Force to evaluate nuclear power plant responses to the blackout. NRC continues to interact with other Federal agencies, the electric power industry, coordinating councils, electric reliability councils, economic regulators, and industry groups on electric grid reliability and utility restructuring. NRC is working to better understand the changes in grid performance to develop an appropriate response to ensure continued safe operation of nuclear power plants in a deregulated market. In addition, since grid stability is an international issue, NRC continues to share its insights with its foreign counterparts and to learn from their experiences with loss of power events.

In April 2004, NRC issued a Regulatory Information Summary to inform licensees that grid reliability can have an impact on plant risk and the operability of offsite power. Licensees should be aware of the offsite power needs of the plant and also be aware of situations that can result in unavailability of systems that could impact operability of offsite power. The Regulatory Information Summary stated that a communication interface with the plant's transmission system operator, together with other local means used to maintain an awareness of changes in the plant switchyard and offsite power grid, is appropriate to determine the impact of these changes on operability of the offsite power system.

Emergency Preparedness and Security

Emergency Preparedness

A key component of NRC's mission is to adequately protect public health and safety during a radiological emergency. NRC-regulated emergency preparedness programs successfully support this mission as part of the overall defense-in-depth safety strategy.

After the terrorist attacks on September 11, 2001, NRC required nuclear facility licensees to assess the potential impact of a terrorist-initiated event on the site emergency plan.

Additionally, NRC's emergency preparedness experts have routinely observed security exercises to assess and improve the interface between security plans and emergency plans.

NRC has acknowledged the need to increase communication of its emergency preparedness activities with the public, industry, the international nuclear community, and Federal, State, and local Government agencies. To do so, the agency has formed an Emergency Preparedness Directorate. This Directorate develops emergency preparedness policies, regulations, programs, and guidelines for both currently licensed nuclear reactors and potential new nuclear reactors; provides technical expertise on emergency preparedness issues and coordinates with other parts of NRC and stakeholders; and oversees and provides technical direction for the emergency preparedness cornerstone of the Reactor Oversight Process.

Security

For more than 25 years, NRC's regulations have required that nuclear facilities must maintain rigorous security programs. Commercial nuclear power facilities are among the best defended and most hardened commercial facilities in the Nation. Immediately after September 11, 2001, NRC issued a series of safeguards and threat advisories to the major licensed facilities, placing them on the highest security level, and recommending additional security measures. These actions enhanced security across the nuclear industry, and many of the enhanced security measures are now requirements imposed on the licensees, as a result of subsequently issued NRC orders. Specifically, the security enhancements include measures to provide additional protection against vehicle bombs, as well as water and land-based assaults.

Nonetheless, in light of the terrorist attacks on September 11, 2001, NRC launched a comprehensive review of the security and safeguards programs of all facilities under its jurisdiction. Assessments of potential vulnerabilities and associated mitigative strategies are an integral part of that comprehensive review. The assessments provide additional protection beyond levels achieved by existing safety, security, and safeguards requirements and allow NRC to confirm the adequacy of the existing regulatory framework and form the basis for any regulatory changes that may be necessary. In addition, NRC continues to enhance coordination and collaboration with other agencies on homeland security.

For more details on emergency preparedness and security, see Article 16.

Other Major Regulatory Accomplishments

Since the Commission issued its previous *U.S. National Report* in 2001, NRC has amended its regulations concerning operator training, radiation protection, decommissioning funding, partial site release, importing of components, combustible gas control, electronic maintenance and submission of information, fire protection, and hearing procedures.

Operator Licensing. In October 2001, NRC amended 10 CFR Part 55 to permit applicants for operator and senior operator licenses to fulfill a part of the required experience prerequisites by manipulating a plant-referenced simulator as an alternative to manipulating the controls of an actual nuclear power plant. This change takes advantage of improvements in simulator technology and reduces unnecessary regulatory burden on licensees.

Radiation Protection. NRC amended its regulations in 10 CFR Part 20 to change the definition and method of calculating shallow-dose equivalents. The numerical shallow-dose equivalent limit of 50 rem was not changed; however, effective June 4, 2002, the shallow-dose equivalent must be the dose averaged over the 10 square centimeters of skin receiving the highest dose (rather than 1 square centimeter as stated in the existing rule). NRC believes that this averaging of the dose over a larger area better reflects the risk associated with shallow-dose equivalents to small areas of the skin. The higher effective dose limit will allow licensees to minimize risk-significant whole body doses (by reducing workplace monitoring), and reduce the use of protective clothing (often a cause of increased risk of heat stress). This change results from a U.S. initiative toward risk-based dose limits derived from the National Council on Radiation Protection and Measurements (NCRP) and is consistent with those of the International Commission on Radiological Protection (ICRP).

Decommissioning Funding. In December 2002, NRC amended its decommissioning trust provisions for nuclear power plants, as specified in 10 CFR Parts 50 and 72. For licensees that are no longer rate-regulated, or no longer have access to a non-bypassable charge for decommissioning, NRC is requiring that decommissioning trust agreements must be in a form acceptable to the Agency. This requirement should increase assurance that an adequate amount of decommissioning funds will be available for their intended purpose.

Partial Site Release. In April 2003, NRC amended 10 CFR 50.75 to standardize the process for allowing a power reactor licensee to release part of its facility or site for unrestricted use before NRC approves the license termination plan. This type of release is termed a "partial site release." This amendment identifies the criteria and regulatory framework for a licensee to use to request NRC approval for a partial site release and also ensures that residual radioactivity will meet the radiological criteria for license termination, even if parts of the site were released before license termination. This amendment also clarifies that the radiological criteria for unrestricted use apply to a partial site release.

Import of Nuclear Equipment. In May 2003, NRC amended 10 CFR Part 110 to issue a general license for importing major components of utilization facilities for end-use at NRC-licensed reactors. This amendment is necessary to facilitate imports of major components of domestic nuclear reactors in furtherance of protection of public health and safety, and also reduces unnecessary regulatory burden related to maintaining NRC-licensed reactors.

Combustible Gas Control. In September 2003, NRC amended 10 CFR Parts 50 and 52, as they relate to combustible gas control in power reactors. This amendment is applicable to current licensees, and also consolidates combustible gas control regulations for future reactor applicants and licensees. In particular, this amendment eliminates the requirements for hydrogen recombiners and hydrogen purge systems and relaxes the requirements for hydrogen and oxygen monitoring equipment, commensurate with their risk significance. This action stems from NRC's ongoing effort to risk-inform its regulations, and is intended to reduce the regulatory burden on present and future reactor licensees.

Electronic Maintenance and Submission of Information. In October 2003, NRC published the final rule on "Electronic Maintenance and Submission of Information" (10 CFR Chapter 1) to clarify when and how licensees and other members of the public may use electronic means (such as CD-ROM, email, and fax) to communicate with the agency. This rule change is necessary to implement the Government Paperwork Elimination Act. As such, the rulemaking removed from the regulations language that stated or suggested an unnecessary prohibition of electronic document submission to the agency. As a result, the final rule allows all licensees, vendors, applicants, and members of the public the option, where practicable, to submit documents to NRC in an electronic format.

Alternative Fire Protection Rule. In June 2004, the NRC amended its fire protection requirements for nuclear power plants to allow licensees to voluntarily adopt a new set of requirements that incorporate risk insights. The new rule maintains safety and provides flexibility to existing requirements. The rule endorses the National Fire Protection Association Standard No. 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition." After adoption, utilities can modify their fire protection program consistent with the standard, without prior specific NRC review and approval. Licensee oversight will be provided by inspectors as part of the Reactor Oversight Process. The rule is part of an effort by the agency to incorporate risk information into its regulations.

Hearing Procedures. In January 2004, NRC amended its regulations concerning the rules of practice in 10 CFR Part 2 to fashion hearing procedures that are tailored to the various types of regulatory activities that NRC conducts. These revisions will make NRC's hearing process more effective and efficient and will better focus the limited resources of involved parties.

Nuclear Installations in the United States

Annex 1 contains a list of the commercial nuclear installations in the United States.

CONCLUSIONS ON THE U.S. NATIONAL REPORT FROM THE SECOND REVIEW MEETING

This section presents the conclusions from the review of the *U.S. National Report* (2001) at the Second Review Meeting in April 2002. As stated in the guidelines for national reports promulgated by the International Atomic Energy Agency (IAEA), each national report should contain a section to allow a contracting party to present its conclusions from the discussion of its previous national report at the Second Review Meeting. Additionally, the contracting party should discuss (a) strong features in its current practices, and (b) areas for improvements and major challenges for the future.

Feedback provided to U.S. delegates by other country participants indicated a positive view of the *U.S. National Report*, the comprehensive response to each of the written questions submitted by other countries, the CD-ROM provided with the written questions and answers, the U.S. presentation, and response to the oral questions during the Country Group meeting. This feedback contributed to the view that the U.S. delegation was informative, open, and candid.

Items in the Final Summary Report of the Second Review Meeting

The final Summary Report of the Second Review Meeting listed the following items as needing further attention by the Contracting Parties during the Third Review Meeting. The article of this report that addresses each item is shown in parentheses:

- availability of human and financial resources of regulatory bodies [Articles 8 (budget) and 11]
- deregulation of electricity markets and changes of ownership (Article 11)
- effective independence of the regulatory body (Article 8)
- use of probabilistic safety assessment and performance indicators (Articles 6 and 10)
- challenges associated with licensing new reactors (Article 17)
- international cooperation on a bilateral and multilateral basis among regulatory bodies (Articles 8 and 16)
- implementation of quality management systems (Article 13)
- safety reviews and safety review processes for plant life extension (Article 14)
- important events at nuclear power plants (Article 6, 10, and 19)
- measures for severe accident management (Articles 10 and 16)
- improvements in emergency preparedness (Article 16)

Items Resulting from Country Group Session

The discussion of the *U.S. National Report* focused on the following four areas:

(1) Risk-Informed Regulation, including the use of safety goals
(2) Performance-Based Reactor Oversight Process
(3) License Renewal, and its comparison with the periodic safety reviews of many countries
(4) Licensing of New Reactors

The reviewers determined that the United States complied with the provisions of the Convention. They also determined that U.S. experience in further developing and applying its regulatory process in the four areas listed above would be of high interest for discussion at the next review meeting.

The current *U.S. National Report* addresses these issues under the relevant articles. Specifically, risk-informed regulation is covered under Article 10, the Reactor Oversight Process is covered under Article 6, license renewal and periodic safety reviews are covered under Article 14, and new reactor licensing is covered under Articles 17 and 18.

NRC Major Challenges for the Future

NRC identified major challenges for the future in its strategic plan; those that apply to the reactor safety arena are listed below.

The Changing Regulatory Environment

The many industries that use radioactive materials are changing, particularly with regard to nuclear safety, security and emergency preparedness, risk-informed performance-based regulations, energy production, and waste management, creating challenges that must be met. The section below describes changes expected within the next 5 years.

• NRC strategic initiatives will significantly emphasize strengthening the interrelationship among safety, security, and emergency preparedness.

• The majority of operating nuclear power plants will have applied for license renewal to help meet the country's demand for energy. A primary challenge is to monitor, manage, and control the effects of aging such that safety is ensured for the renewal period.

• The Department of Energy will apply to construct and operate the country's high-level radioactive waste repository. The timing of this action will challenge the allocation of NRC's resources.

• The U.S. nuclear power industry will show a growing interest in licensing and constructing new nuclear power plants to meet the Nation's demand for energy. Challenges include analyzing in detail the vulnerability to accidents and security compromises, as well as developing inspections, tests, analyses, and acceptance criteria for construction.

• NRC, Agreement States (described in Article 8), and licensees will continue to devote increasing attention to the security of radioactive materials and facilities. The primary challenge facing the NRC is to emerge from the period of uncertainty in post-September 11 security requirements; determine what long-term security provisions are necessary; and revise regulations, orders, and internal procedures as necessary to ensure public health and safety and the common defense and security in an elevated threat environment.

• NRC will continue to see increased requirements to coordinate with a wide array of Federal, State, and local agencies related to homeland security and emergency planning. NRC currently conducts emergency preparedness exercises that involve a wide array of governmental agencies and emergency response personnel and uses cooperative intergovernmental relationships to balance and inform national response capabilities.

• The regulatory climate is expected to adjust to both internal and external factors (described below). Challenges include materials degradation at nuclear power plants, new and evolving technologies, and continual review of ongoing operational experience.

Key External Factors

NRC's ability to achieve its goals depends on a changing equation of industry operating experience, national priorities, market forces, and availability of resources. The following section discusses significant external factors, all of which are beyond the control of NRC but could affect the agency's ability to achieve its strategic goals.

Receipt of New Reactor Operating License Applications. The U.S. nuclear industry has indicated a new and growing interest in licensing and constructing new nuclear power plants. If NRC receives a substantial increase in new reactor operating license applications beyond that currently anticipated, the agency would have to significantly reallocate resources to review applications in a timely manner and inspect construction activities. In addition, the high level of public interest likely to be associated with such applications would require significant efforts by NRC to keep stakeholders informed and involved in the licensing process.

Significant Operating Incident (Domestic or International). A significant safety incident could cause an unexpected increase in safety and security requirements that would likely change the agency's focus on initiatives related to its five goals until the situation stabilized. Because NRC stakeholders (including the public) are highly sensitive to many issues regarding the use of radioactive materials, even events of relatively minor safety or security significance can sometimes require a response that consumes considerable agency resources.

Significant Terrorist Incident. A significant terrorist incident anywhere in the U. S. could significantly alter the Nation's priorities. This, in turn, could affect significance levels, a need for new or changed security requirements, or other policy decisions that might impact NRC, its partners, and the industry it regulates. In particular, the impact on State regulatory and enforcement authorities might affect their ability to work with NRC in achieving its goals.

A significant terrorist incident at a nuclear facility or activity anywhere in the world would likely cause similar changes in NRC's priorities and potentially in U.S. policy regarding export activities, NRC's role in international security, and/or requirements for security at U.S. nuclear power plants.

Timing of the Department of Energy Application and Related Activities for the High-Level Waste Repository at Yucca Mountain. The proposed repository for spent nuclear fuel represents a major effort for NRC in planning, review, analysis, and ultimate decision-making regarding the licensing of the facility. The agency has begun to ramp up this effort to respond to preapplication activities by the Department of Energy. The timing of the Department's actions will heavily influence NRC's resource allocation decisions over the next several years. Acceleration or delay in the Department's activities will most likely require reprogramming of NRC resources, which may affect other programs that are directly associated with achieving the agency's goals.

Homeland Security Initiatives. Emergency preparedness activities with Federal, State, and local agencies continue to increase in scope and number. This impacts the agencies' priorities and workloads. As more resources are diverted to external coordination activities, previous work activities must be re-prioritized.

Legislative Initiatives. Numerous legislative initiatives under consideration by Congress could have a major impact on NRC. In particular, pending energy legislation would affect the agency's priorities and workload, if enacted. Increasing interest in diversified sources of energy and energy independence could cause a rise in license applications for nuclear power plants. Any attendant increase in resources devoted to license review and analysis might affect the agency's ability to achieve its goals for the planning period.

Major Management Challenges

By law, the Inspector General of each Federal agency (described in Article 8) is to describe what he or she considers to be the most serious management and performance challenges facing the agency and assess the agency's progress in addressing those challenges. Accordingly, NRC's Inspector General prepared his annual assessment of the major management challenges confronting the agency. The full report can be found on NRC's public Web site at http://www.nrc.gov/reading-rm/doc-collections/insp-gen/2004/04-01.pdf, and the main results are summarized in an appendix to this report.

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 reasonable 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. First, this section summarizes the characteristics of the nuclear industry in the United States. It then explains reactor licensing and discusses the major oversight process in the United States — the Reactor Oversight Process — and supporting programs, including the Industry Trends Program and the Program for Resolving Generic Issues. Next, this section discusses programs for rulemaking, decommissioning, and research, as well as programs for public participation, handling petitions, resolving allegations, and settling differing professional opinions. The “Experience and Examples” subsections cover nuclear installations for which assessments by the NRC showed that corrective actions were necessary. NRC posts the major results of assessments on the agency’s public Web site at http://www.nrc.gov.

The safety performance of the U.S. nuclear power industry has improved substantially over the past 10 years, and nuclear reactors (collectively) are operating above acceptable safety levels consistent with the agency’s Safety Goal Policy. If the agency identifies the need for substantial safety improvements, it will only impose additional requirements consistent with the Commission’s Backfit Rule in Title 10, Section 50.109, of the *Code of Federal Regulations* (10 CFR 50.109).

This section was substantially updated.

6.1 Nuclear Installations in the United States

As of September 2004, 104 commercial nuclear power reactors are licensed to operate in 31 States. Although similar in many ways, each reactor design is unique. The U.S. nuclear power industry comprises 4 reactor vendors (1 BWR vendor and 3 PWR vendors),[b] 27 licensees, 80 different designs, and 65 sites. The 104 operating reactors have accumulated about 2,600 reactor-years of experience; permanently shutdown reactors have accumulated 385 additional reactor-years. For a list of nuclear installations in the United States, see Annex 1.

[b] The BWR vendor is General Electric. Initially there were three distinct PWR vendors representing three unique designs: Westinghouse, Combustion Engineering, and Babcock & Wilcox. There are still three PWR basic designs, but they are all owned by Westinghouse now.

6.2 Regulatory Processes and Programs

This section discusses the processes and programs that NRC uses to ensure that plant safety is maintained. These comprise a well-established licensing process, including power uprates; reactor oversight; programs for responding to and evaluating operational events (including the Industry Trends Program, the Accident Precursor Program, and the Program for Resolving Generic Safety Issues); programs for rulemaking, decommissioning, and regulatory research; and programs for public participation and for handling petitions, allegations, and differing professional opinions.

6.2.1 Reactor Licensing

To construct and operate a nuclear reactor, an entity must submit an application to NRC for safety and environmental review. The public has opportunities to participate through a hearing process. NRC licensed all current operating nuclear plants under the detailed two-step process specified in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," first issuing a construction permit and then an operating license. Although NRC has not received any applications for new reactor operating licenses since 1976, it has recently received three applications for early site permits. The agency is reviewing these applications under the new streamlined one-step process specified in 10 CFR Part 52, "Early Site Permits, Design Certifications and Combined Operating Licenses." The 10 CFR Part 52 regulations are covered in more detail in Article 18.

The reactor licensing process provides for the review and approval of changes after initial licensing. These provisions address amendments to the operating license to support plant changes, license renewal, changes of ownership and license transfer, exemptions and relief from NRC regulations, and increasing the reactor power level ("power uprates"). These provisions are discussed further in other articles, except for power uprates, which are discussed below.

6.2.1.1 Power Uprates

NRC carefully reviews requests to raise the maximum licensed power level at which a plant may be operated, which is called a power uprate. This section describes the program for power uprates and summarizes operating experience.

Description of the Program. The three categories of power uprates can be classified as (1) measurement uncertainty recapture power uprates, (2) stretch power uprates, and (3) extended power uprates. On the order of 1.5 percent, measurement uncertainty recapture power uprates are achieved by implementing enhanced techniques for calculating reactor power. These techniques employ state-of-the-art feedwater flow measurement devices that reduce the degree of uncertainty in measuring feedwater flow and, in turn, provide for a more accurate calculation of power. 10 CFR Part 50, Appendix K, "ECCS (Emergency Core Cooling System) Evaluation Models," which allowed licensees to use a power uncertainty of less than 2 percent in loss-of-coolant accident analyses, facilitated reviews of these uprates. Stretch power uprates, typically on the order of 7 percent, usually involve changes to instrumentation setpoints, and do not generally involve major plant modifications. Stretch power uprates may require high pressure/low pressure turbine upgrades, especially if they already had a measurement uncertainty uprate. Extended power uprates are usually greater than stretch power uprates and require significant modifications to major balance-of-plant equipment, such as the high-pressure turbines and condensate pumps and motors. NRC has approved extended power uprates up to 20 percent.

Licensees have been implementing power uprates since the 1970s to increase the power output of their plants. The staff has completed more than 100 reviews for power uprates. As of August 2004, the staff had approved measurement uncertainty recapture power uprates for 34 units, stretch power uprates for 55 units, and extended power uprates for 12 units. As a result, the United States has gained the power production equivalent of about four 1,000-megawatt electric (MWe) nuclear power plant units by implementing power uprates at existing plants.

Having recognized licensees' increased interest in power uprates, NRC has enhanced its processes for submitting and reviewing power uprates. On January 31, 2002, NRC issued Regulatory Information Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications." This document guides licensees on the scope and detail of the information to provide in measurement uncertainty recapture power uprate applications. In addition, on December 24, 2003, NRC issued Review Standard (RS)-001, "Review Standard for Extended Power Uprates." The purpose of this first-of-a kind document is to guide NRC staff in reviewing extended power uprate applications to enhance consistency, quality, and completeness of reviews. The review standard also informs licensees of the guidance documents that provide acceptance criteria that NRC staff use when reviewing extended power uprate applications. This information should help licensees prepare extended power uprate applications that are complete with respect to the scope of review. NRC provides reference links to these documents and a substantial amount of public information on the NRC public Web page for "Power Uprates," at http://www.nrc.gov/reactors/operating/licensing/power-uprates.html.

NRC surveyed all licensees in January 2004 for information about their future plans for submitting power uprate applications. The survey targeted projections for the size of power uprates and schedule of submittals over the next five years. The results of this survey indicate that licensees plan to submit 25 power uprate applications in the next five years. Of these, they expect to submit 12 for extended power uprates, five for stretch power uprates, and eight for measurement uncertainty recapture power uprates. The sizes reported for the stretch and extended power uprates may also include measurement uncertainty recapture. Judging from the information provided, NRC expects the planned power uprates to increase the Nation's electric power generating capacity by about 1,730 MWe by calendar year 2009.

Operating Experience with Power Uprates. NRC monitors operating experience at plants that have implemented power uprates. Since 2002, steam dryer cracking and flow-induced vibration damage on components and supports for the main steam and feedwater lines have been observed at these plants.

On December 21, 2001, NRC approved increasing the maximum licensed power limits for the Dresden and Quad Cities reactors by about 17 percent. On June 7, 2002, after operating a few months under extended power uprate conditions, Quad Cities Unit 2 began experiencing fluctuations in steam flow, reactor pressure and level, and moisture carryover in the main steam lines. On July 11, 2002, Quad Cities Unit 2 was shut down to investigate the cause of these fluctuations. The licensee discovered that a 1.3-cm (½-inch) thick dryer cover plate on the outside of the steam dryer had broken off, resulting in pieces of the cover plate in a main steamline flow venturi and a turbine stop valve strainer. The steam dryer, located in the upper region of the boiling-water reactor vessel, removes moisture from the steam before the steam is delivered to the turbine. The steam dryer does not perform an accident mitigating role or safety function, but it is required to maintain its structural integrity. General Electric, the reactor vendor, determined that the cover plate most likely failed from high-cycle fatigue. The licensee repaired the steam dryer by installing thicker cover plates, and restarted Quad Cities Unit 2 on July 21, 2002. In late May 2003, the licensee again noted increasing moisture carryover at Quad Cities Unit 2, but did not discern any changes in other reactor parameters. On June 11, 2003, Quad Cities Unit 2 was shut down to inspect the steam dryer. The inspection revealed additional cracking and failures on the dryer hood, internal braces, and tie bars. Following repairs, Quad Cities Unit 2 was restarted on June 28, 2003.

On October 28, 2003, the licensee identified an increase in flow through one of the main steam lines, with a corresponding decrease of about the same amount in the other main steam lines at Quad Cities Unit 1. On October 31, 2003, the licensee began monitoring a steady increase in moisture carryover at Quad Cities Unit 1. These increases were similar to those observed at Quad Cities Unit 2, which resulted from cracking in the steam dryer at that unit. Consequently, on November 12, 2003, Quad Cities Unit 1 was shut down for inspections and repairs to the steam dryer. The unit had been operating at a reduced power level since November 3, owing to indications of higher-than-expected moisture carryover in the reactor steam. On November 13, the steam dryer was found to be damaged. The damage occurred in the upper dryer hood cover plate, which had cracks approximately 129.5 cm (51 inches) in total length and a 15.2-cm by 30.4 cm (6-inch by 9-inch) portion of the plate broke off the steam dryer.

In an effort to locate the lost steam dryer piece(s), the licensee conducted extensive inspections. The licensee did not recover the piece(s); however, it found indications on a recirculation pump impeller, which indicate that the material is most likely in the bottom of the reactor vessel. The licensee completed repairs and modifications on Unit 1, similar to those completed on the Unit 2 steam dryer earlier in 2003. Also during this outage, the licensee discovered that the pilot vent line, on a main steamline electromatic relief valve, was sheared off from the pilot assembly and the solenoid actuator for the valve was significantly damaged. The four electromatic relief valves are safety-related components that automatically depressurize the reactor, if required. Flow-induced vibration on the main steamline, during extended power uprate operating conditions, contributed to this damage. The licensee inspected the other three electromatic relief valves and did not identify any significant issues impacting operability. The licensee replaced the damaged solenoid actuator and rewelded the pilot vent line to the pilot assembly before restarting the unit.

After two years of operating at the extended power uprate power level at Dresden 2, the licensee found cracking on the steam dryer during the November 2003 refueling outage. The licensee did not find any indications of higher-than-expected moisture carryover in the reactor steam during the previous operating cycle at Dresden 2. The licensee completed repairs and modifications on the steam dryer for Dresden 2, similar to those performed at Quad Cities Units 1 and 2, during the Fall 2003 refueling outage. The licensee also found three holes in a feedwater sparger caused by a broken feedwater sampling probe that was retrieved in the sparger. The licensee repaired the feedwater sparger and replaced the sampling probe before restarting the reactor. In December 2003, the licensee also discovered two 10.2-cm (4-inch) through-wall cracks in the steam dryer hood and two broken feedwater sampling probes at Dresden 3. The licensee repaired the dryer and replaced the probes before restarting the plant.

NRC issued Information Notice 2002-26 and Supplements 1 and 2 to alert licensees to the failure of the steam dryer and other plant components at Quad Cities Unit 1 and the steam dryer cracking at Quad Cities Unit 2. NRC continues to evaluate the steam dryer cracking issues and damage to other plant components while considering the generic implications for other plants. The staff remains actively engaged with the industry regarding plans for addressing these issues generically.

Another issue that NRC is closely monitoring is the unexpected, small differences in power level indications that have been observed at Braidwood and Byron, while using the Westinghouse Advanced Measurement Analysis Group “CROSSFLOW” ultrasonic feedwater flow measurement system. NRC is closely monitoring the use of this “CROSSFLOW” system, especially at plants using this system for measuring uprated power levels. NRC is evaluating the potential impact of this issue on power uprates to determine whether regulatory action is necessary.

6.2.2 Reactor Oversight Process

NRC continuously oversees nuclear power plants to verify that they are being operated in accordance with the agency’s rules and regulations. NRC has full authority to take whatever action is necessary to protect public health and safety, and may demand immediate licensee actions, up to and including a plant shutdown.

This section explains the Reactor Oversight Process. It covers the goals, regulatory framework, inspections and performance indicators, significance determination process, enforcement, how the current oversight process differs from the previous process, and experience and examples.

6.2.2.1 Goals

The Reactor Oversight Process is intended to fulfill several strategic goals, including the use and management of radioactive materials and nuclear fuels for beneficial civilian purposes in a manner that protects public health and safety and the environment, promotes the common defense and security of the United States, and provides for regulatory actions that are effective, efficient, and open.

6.2.2.2 Regulatory Framework

The NRC’s regulatory framework for reactor oversight is shown in Figure 1. A risk-informed, tiered approach for ensuring plant safety, this framework has three key strategic performance areas, including reactor safety, radiation safety, and safeguards. Within each strategic performance area are “cornerstones” that apply to the essential safety aspects of plant operation. Satisfactory licensee performance in the cornerstones provides reasonable assurance that plants are being safely operated, and that NRC’s safety mission is being accomplished. Within this framework, NRC’s oversight process for operating reactors provides a means of collecting information about licensee performance, assessing that information for safety significance, and providing for appropriate licensee and NRC responses.

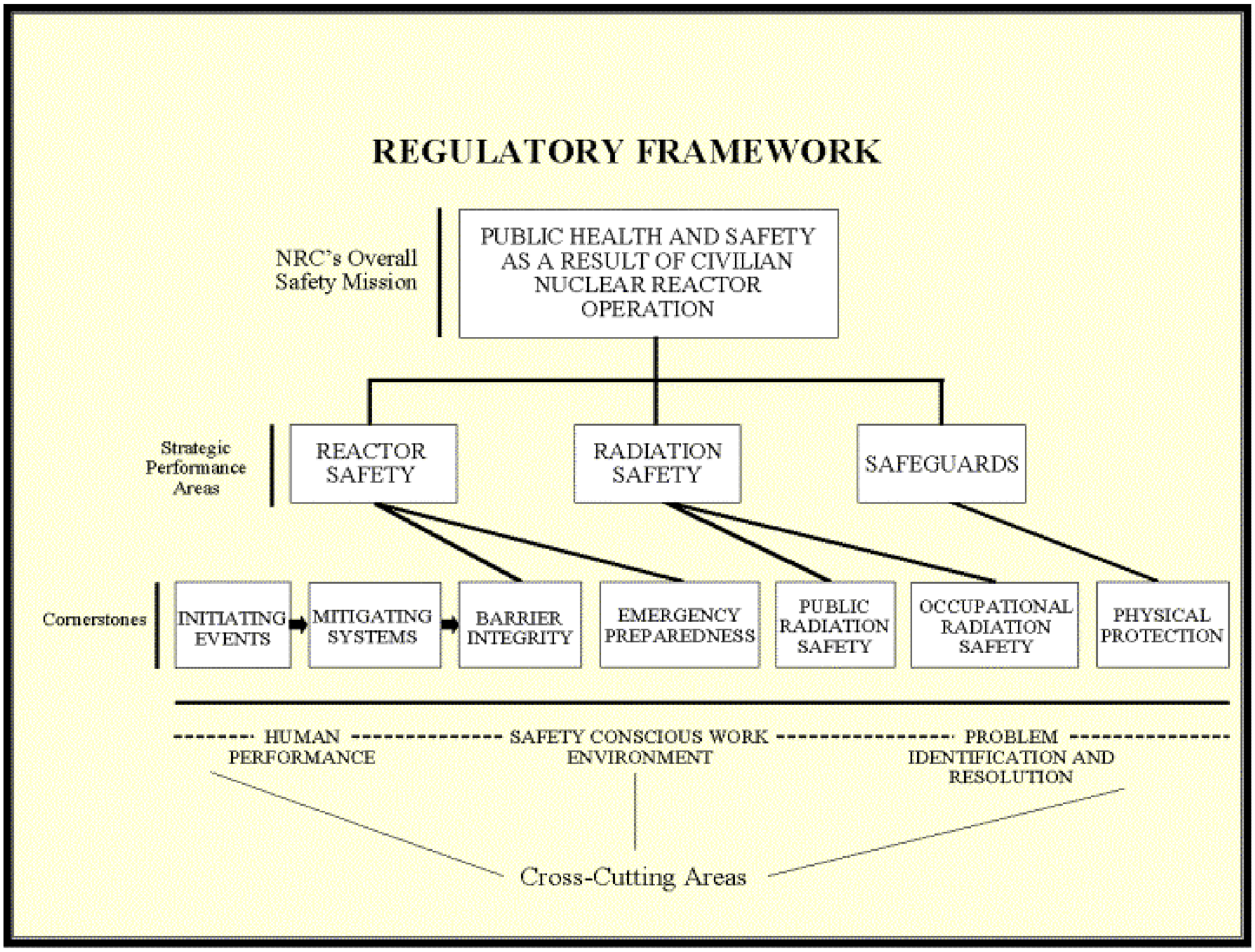


Figure 1: Regulatory Framework

To measure plant performance, the oversight process focuses on the following seven specific "cornerstones" that support the safety of plant operations in the three key strategic areas:

(1) Initiating Events focuses on operations and events at a nuclear plant that could lead to a possible accident if plant safety systems did not intervene. These events include equipment failures leading to a plant shutdown, shutdowns with unexpected complications, or large changes in the plant's power output.

(2) Mitigating Systems focuses on the function of safety systems that are designed to prevent or reduce the consequences of an accident.

(3) Barrier Integrity focuses on the licensee's effectiveness in maintaining the three physical barriers (i.e., fuel rods, reactor vessel and associated piping, and containment).

(4) Emergency Preparedness focuses on the effectiveness of the plant's staff in carrying out its emergency plans.

(5) Public Radiation Safety focuses on the effectiveness of the licensee's programs to meet applicable Federal limits involving the exposure, or potential exposure, of members

of the public to radiation, and ensure that the effluent releases from the plant are as low as is reasonable achievable (ALARA).

(6) Occupational Radiation Safety focuses on the effectiveness of the licensee's program(s) to maintain the worker dose within the regulatory limits and provide occupational exposures that are ALARA.

(7) Physical Protection focuses on the effectiveness of the security, safeguards, and fitness-for-duty programs. (See the introduction to this report for an explanation to withhold certain information from the public.)

In addition to the seven cornerstones, the reactor oversight framework recognizes three "cross-cutting" elements — human performance, safety-conscious work environment, and problem identification and resolution — that affect multiple cornerstones. The assessment of these cross-cutting elements is important to the oversight process.

Within each of the seven cornerstones, NRC designed performance indicators (See Table 1) and inspections to closely focus on plant activities that most affect safety and overall risk.

Table 1: Performance Indicators by Cornerstone

Safety Cornerstone	**Performance Indicator**
Initiating Events	Unplanned reactor shutdowns (automatic and manual)
	Loss of normal reactor cooling system following unplanned shutdown
	Unplanned events that result in significant changes in reactor power
Mitigating Systems	Safety system not available • specific emergency core cooling systems • emergency electric power systems
	Safety system failures
Integrity of Barriers to Release of Radioactivity	Fuel Cladding (measured by radioactivity in the reactor cooling system)
	Reactor cooling system leak rate
Emergency Preparedness	Drill and exercise performance
	Emergency response organization drill participation
	Alert and Notification System Reliability
Public Radiation Safety	Effluent release requiring reporting under NRC regulations and license conditions
Occupational Radiation Safety	Compliance with requirements for the control of access to areas of the plant with dose rates greater than one rem/hr (10 mSv/hr.) Unintended radiation exposures to workers greater than a specified fraction of the dose limits in 20.1201, 20.1207, and 20.1208 [1]
Physical Protection[c]	Security system equipment availability
	Personnel screening program performance
	Employee fitness-for-duty program effectiveness
[1] 2% of the stochastic limit in 10 CFR 20.1201 for total effective dose equivalent, 10% of the non-stochastic limit in 10 CFR 20.120 for individual organ doses, and 20% of the limits in 10 CFR 20.1207 and 10 CFR 20.208 for minors and declared pregnant workers, respectively.	

[c] Although the NRC is actively overseeing the physical protection cornerstone, the Commission has decided that the related inspection and assessment information will not be publically available to ensure that potentially useful information is not provided to a possible adversary.

6.2.2.3 Inspections and Performance Indicators

Although performance indicators provide insights into nuclear power plant performance for selected areas, NRC's inspection program provides a greater depth and breadth of information. (A performance indicator is a quantitative measure of a particular attribute of licensee performance that indicates how well a plant is performing when measured against established thresholds. Licensees submit this data quarterly, and the NRC regularly performs verification inspections of their submittals. As previously mentioned, the NRC uses its analysis of this data with its own inspection data for assessment of a plant's performance.) The key feature of the inspection program is the baseline inspection program, the minimum level of inspection that all plants receive, regardless of performance, to evaluate licensee performance over a 12-month period. The baseline inspection program verifies the accuracy of performance indicator information provided by the licensee, and assesses performance that is not directly measured by the performance indicator data. The overall objective of the program is to monitor all power reactor licensees at a defined level to ensure that their performance meets the objectives for each cornerstone of safety.

Baseline inspections are targeted around the seven cornerstones, and focus on activities and systems that are "risk-significant" (i.e., those that could trigger an accident, mitigate the effects of an accident, or increase the consequences of a possible accident). Baseline inspections include (1) inspection of areas that are not covered or not fully covered by performance indicators, (2) inspections to verify the accuracy of a licensee's reports on performance indicators, and (3) inspections of the licensee's effectiveness in finding and resolving problems on its own. During baseline inspections, inspectors also examine the cross-cutting issues.

In addition, NRC performs supplemental inspections and special inspections at plants where performance falls below established performance indicator thresholds or when baseline inspections reveal significant findings or to respond to a specific event or problem that may arise at a plant. A supplemental inspection is performed to independently evaluate the root causes of performance deficiencies when indications of declining licensee performance are obtained through either the performance indicators or other inspections (principally the baseline inspection program). Significant operational events are evaluated by NRC inspectors to determine whether an agency response is warranted. Plants in extended shutdowns due to performance problems are inspected and assessed by a separate inspection process (Inspection Manual Chapter 0350) because many of the performance indicators and much of the baseline inspection program do not apply to plants in extended shutdowns.

Table 2 shows the relationship between the 36 baseline inspection areas and the 7 safety cornerstones.

Abbreviations used in Table 2 follow:

IE-	Initiating Events
MS	Mitigating Systems
BI	Barrier Integrity
EP	Emergency Preparedness
ORS	Occupational Radiation Safety
PRS	Public Radiation Safety
PP	Physical Protection

Table - 2 Inspectable Areas by Cornerstone							
Inspectable Area	IE	MS	BI	EP	ORS	PRS	PP
Access control to radiologically significant areas					X		
Access authorization program							X
Access control							X
Adverse weather protection	X	X					
ALARA planning and controls					X		
Alert and notification system testing				X			
Drill evaluation				X			
Emergency response organization augmentation testing				X			
Emergency action level and emergency plan changes				X			
Equipment alignment	X	X	X				
Evaluations of changes, tests, or experiments		X	X				
Exercise evaluation				X			
Fire protection	X	X					
Flood protection measures	X	X					
Heat sink performance	X	X					
Identification and resolution of problems	X	X	X	X	X	X	X
Inservice inspection activities	X		X				
Licensed operator requalification		X	X	X			
Maintenance risk assessments and emergent work evaluation	X	X	X				

Table - 2 Inspectable Areas by Cornerstone							
Inspectable Area	IE	MS	BI	EP	ORS	PRS	PP
Maintenance rule implementation	X	X	X				
Operability evaluations		X	X				
Operator workarounds		X					
Permanent plant modifications		X	X				
Personnel Performance during nonroutine evolutions	X	X	X				
Post maintenance testing		X	X				
Radiation monitoring instrumentation					X		
Radiation worker performance					X	X	
Radioactive material processing and transportation						X	
Radioactive Gaseous and liquid effluent treatment and monitoring systems						X	
Radiological environmental monitoring program						X	
Refueling and outage activities	X	X	X				
Response to contingency events							X
Safety system design and performance capability		X					
Security plan changes							X
Surveillance testing		X	X				
Temporary plant modifications		X	X				

Figure 2 presents a graphic overview of the Reactor Oversight Process. For each safety cornerstone, NRC develops findings from inspections, evaluates those findings for safety significance using a significance determination process and compares performance indicator data collected by licensees against prescribed thresholds. NRC then assesses the resulting information in accordance with the Action Matrix (Table 3) to determine whether further regulatory action is required. NRC communicates to the public the results of its performance assessment and its inspection plans and other planned actions in correspondence, on its public Web site, and through public meetings.

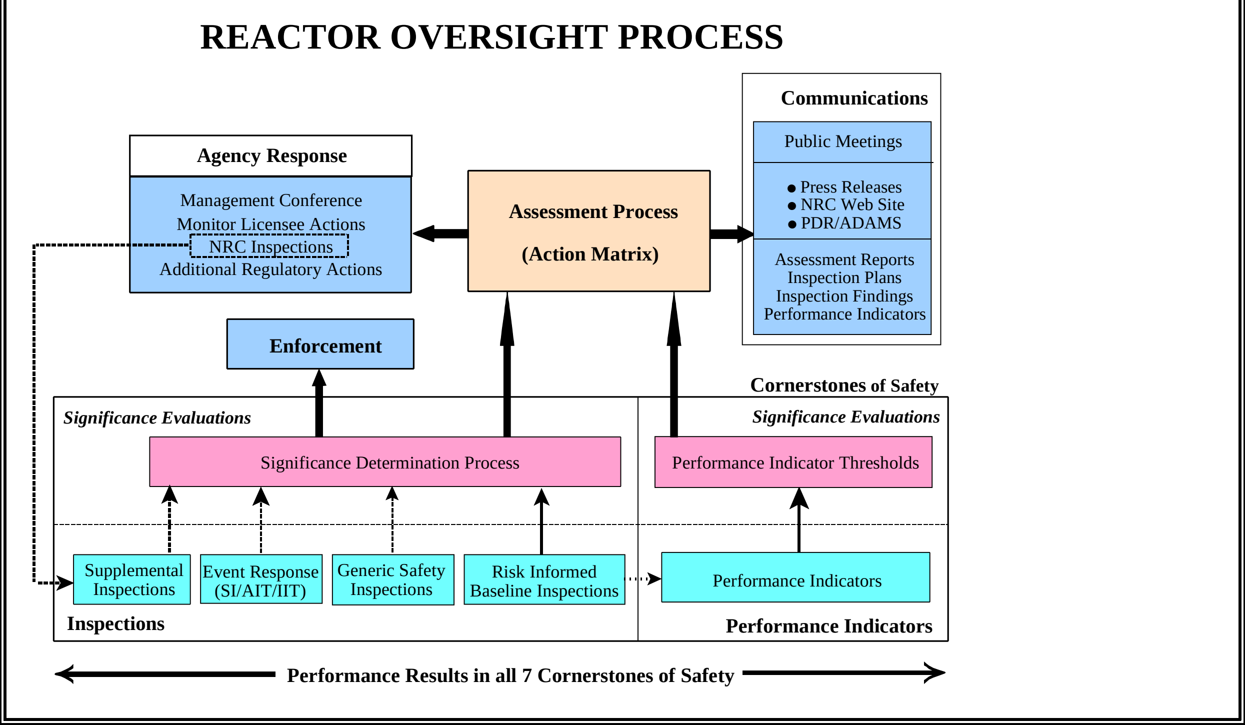


Figure 2: Reactor Oversight Process

6.2.2.4 Significance Determination Process

The Significance Determination Process takes advantage of risk insights to help NRC inspectors and staff determine the safety significance of inspection findings. The process was developed with substantial involvement from both internal and external stakeholders to provide a means for evaluating the significance of inspection findings that is consistent with NRC's regulatory objectives of ensuring that NRC activities are risk-informed, objective, predictable, and understandable. The staff uses this process to screen for inspection findings that do not result in a notable increase in risk and, thus, need not be further analyzed ("green" findings). The staff then subjects the remaining inspection findings to a more thorough risk assessment, according to the next phase of the process, to determine the level of regulatory response needed.

6.2.2.5 Assessment

NRC evaluates the licensee's performance indicator data, and integrates the data with the findings of the agency's inspection program. Each performance indicator has thresholds for measuring acceptable performance. These thresholds indicate risk according to established safety margins, as indicated by a color coding system.

A "green" color code indicates performance within an expected range in which the related cornerstone objectives are met. "White" indicates performance outside an expected range of nominal licensee performance, although related cornerstone objectives are still met. "Yellow" indicates that related cornerstone objectives are met, with some reduction in safety margin. "Red" indicates a significant reduction in the safety margin in the area measured by the given performance indicator. Each licensee reports the performance indicators to NRC on a quarterly basis.

Each calendar quarter, resident inspectors and the inspection staff in each of NRC's four regional offices review the performance of each nuclear power plant in the given region, as measured by the performance indicators and inspection findings. Every six months, NRC staff expand this review to plan for inspections for the following 18-month period.

During the final quarterly review for each year, NRC assesses plant performance over the previous 12 months in more detail and prepares a performance report, and the inspection plan for the following 18-month period. NRC posts these annual performance reports on its public Web site, and holds public meetings with licensees to discuss the previous year's performance at each plant.

In addition, NRC's senior management annually reviews the adequacy of the agency's planned actions for plants with significant performance problems at the Agency Action Review Meeting. Further, they discuss the performance of plants requiring heightened agency scrutiny during an annual public meeting with the Commission at the agency's Headquarters in Rockville, Maryland.

In its quarterly reviews of plant performance, NRC determines what additional action to take if there are signs of declining performance. This approach to regulatory oversight is intended to be more predictable than previous practices by linking regulatory actions to performance criteria. The new process has four levels of regulatory response, according to which NRC regulatory review increases as plant performance declines. (See Table 3.) The cognizant regional office manages the first two levels of heightened regulatory review; the next two levels call for an agency response involving attention from senior management in both Headquarters and Regional offices. NRC's actions for performance below "green" color coding may include meetings with the licensee, additional inspections, and reviews of responses by the licensee. Further declines in performance would warrant stronger action by NRC, possibly including an order that could modify or even suspend the licensee's operating license.

For results of assessments of plant performance, see NRC's public Web site at http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/index.html

Table 3: NRC Action Matrix

		Licensee Response Column	Regulatory Response Column	Degraded Cornerstone Column	Multiple/ Repetitive Degraded Cornerstone Column	Unacceptable Performance Column	IMC 0350 Process
RESULTS		All Assessment Inputs (Performance Indicators (PIs) and Inspection Findings) Green; Cornerstone Objectives Fully Met	One or Two White Inputs (in different cornerstones) in a Strategic Performance Area; Cornerstone Objectives Fully Met	One Degraded Cornerstone (2 White Inputs or 1 Yellow Input) or any 3 White Inputs in a Strategic Performance Area; Cornerstone Objectives Met with Moderate Degradation in Safety Performance	Repetitive Degraded Cornerstone, Multiple Degraded Cornerstones, Multiple Yellow Inputs, or 1 Red Input; Cornerstone Objectives Met with Longstanding Issues or Significant Degradation in Safety Performance	Overall Unacceptable Performance; Plants Not Permitted to Operate Within this Band, Unacceptable Margin to Safety	Plants in a shutdown condition with performance problems placed under the IMC 0350 process
RESPONSE	Regulatory Performance Meeting	None	Branch Chief (BC) or Division Director (DD) Meet with Licensee	DD or Regional Administrator (RA) Meet with Licensee	RA (or Executive Director for Operations(EDO)) Meet with Senior Licensee Management	Commission meeting with Senior Licensee Management	RA (or EDO) Meet with Senior Licensee Management
	Licensee Action	Licensee Corrective Action	Licensee root cause evaluation and corrective action with NRC Oversight	Licensee cumulative root cause evaluation with NRC Oversight	Licensee Performance Improvement Plan with NRC Oversight		Licensee Performance Improvement Plan / Restart Plan with NRC Oversight
	NRC Inspection	Risk-Informed Baseline Inspection Program	Baseline and supplemental inspection procedure 95001	Baseline and supplemental inspection procedure 95002	Baseline and supplemental inspection procedure 95003		Baseline and supplemental as practicable, plus special inspections per restart checklist.
	Regulatory Actions	None	Supplemental inspection only	Supplemental inspection only	-10 CFR 2.204 DFI -10 CFR 50.54(f) Letter - Confirmatory Action Letter (CAL) /Order	Order to Modify, Suspend, or Revoke Licensed Activities	CAL/order requiring NRC approval for restart.
COMMUNICATION	Assessment Letters	BC or DD review/sign assessment report (w/ inspection plan)	DD review/sign assessment report (w/ inspection plan)	RA review/sign assessment report (w/ inspection plan)	RA review/sign assessment report (w/ inspection plan)		N/A. RA (or 0350 Panel Chairman) review/ sign 0350-related correspondence
	Annual Public Meeting	SRI or BC Meet with Licensee	BC or DD Meet with Licensee	RA (or designee) Discuss Performance with Licensee	RA or EDO Discuss Performance with Senior Licensee Management		N/A. 0350 Panel Chairman conduct public status meetings periodically
	Commission Involvement	None	None	None	Plant discussed at AARM	Commission Meeting with Senior Licensee Management	Commission meetings as requested, restart approval in some cases.
	INCREASING SAFETY SIGNIFICANCE ---------->						

6.2.2.6 Enforcement of NRC Requirements

NRC evaluates each violation of agency requirements found during inspections to assess the safety significance. If the evaluation reveals that the violation is of very low safety significance, NRC describes the violation in the inspection report and does not take any formal enforcement action. The licensee is expected to deal with the violation through its corrective action program to prevent a recurrence. NRC may also review the issue during future inspections to determine the effectiveness of the licensee's corrective action. NRC may also issue a notice of violation if the licensee fails to correct a violation of low safety significance in a reasonable period of time.

For more significant violations, the agency will issue a notice of violation, requiring the licensee to formally respond to NRC about its actions to correct the violation, and to identify the steps it will take to prevent the violation from occurring again. The agency then reviews the licensee's actions in a later inspection. Normally, these violations will not be the subject of a monetary civil penalty (fine). However, on rare occasions, there may be violations that warrant a fine because of their high safety significance, such as an accidental criticality.

In addition, the following violations call for additional enforcement action, including the use of severity levels to reflect the significance (rather than the color coding used in the Significance Determination Process) and possible fines:

- willful violations, including discrimination against workers for raising safety issues

- actions that may adversely affect NRC's ability to monitor licensee's activities, including failures to report required information, to obtain NRC approval for certain plant changes, to maintain accurate records, or to give NRC complete and accurate information

- incidents with actual safety consequences, including radiation exposures above NRC limits, doses above NRC limits resulting from releases of radioactive material, or failure to notify government agencies when an emergency response is required

Section 9.2 describes NRC's enforcement program in more detail.

6.2.2.7 Experience and Examples

The Reactor Oversight Process began implementation in April 2000. After the first year of implementation, NRC assessed the efficacy of the process and documented the results in SECY-01-0114, "Results of the Initial Implementation of the New Reactor Oversight Process," issued on June 25, 2001. NRC continues to assess the process annually. The latest report, issued April 6, 2004, presents the results of the calendar year 2003 self-assessment, SECY-04-0053, "Calendar Year 2003 Reactor Oversight Process Self-Assessment."

The self-assessment results indicate that the Reactor Oversight Process was generally effective in monitoring operating nuclear power plant activities and focusing NRC resources on significant performance issues in calendar year 2003. The staff of the NRC maintained its focus on stakeholder involvement and continued to improve various aspects of the Reactor Oversight Process as a result of feedback and lessons learned. In particular, the event at Davis-Besse Nuclear Power Station continues to cause a focused look at the NRC's oversight efforts and has resulted in several program improvements. The responses to the NRC's annual survey of external stakeholders, which solicited feedback on the Reactor Oversight Process, were generally favorable; however, some stakeholders raised concerns about the complexity and subjectivity of the Significance Determination Process, the effectiveness of the performance indicator program, a perceived lack of NRC responsiveness to stakeholder comments, and other areas where improvements have been suggested. NRC is aggressively pursuing improvements to address concerns in the noted areas.

The recommendations led to several changes to the Reactor Oversight Process in calendar year 2003. These changes were made to enhance NRC's ability to detect declining plant performance, including the specific issues that have been identified at Davis Besse. Among the changes already completed are: changes to the inspection program to help identify negative trends regarding equipment performance; an improved frequently asked question process by which performance indicator issues are addressed; enhanced inspector training; and better tracking and management of resident inspector staffing. Although the staff completed the baseline inspection program in calendar year 2003, resource challenges continued. The staff believes that the revised resident inspector staffing policy and additional regional resources allocated in fiscal year (FY) 2004 and beyond will address the site staffing and resource concerns associated with the Reactor Oversight Process. The staff continues to focus on improving the timeliness of the Significance Determination Process and has made significant progress in implementing the significance determination process Improvement Plan. The staff also made several improvements in the assessment program during calendar year 2003, and evaluated other suggested adjustments.

Although significant progress has been made in calendar year 2003, the staff expects to continue to improve the Reactor Oversight Process based on lessons learned and stakeholder feedback. The staff will continue to actively solicit input from NRC's internal and external stakeholders, and evaluate potential program improvements under the ongoing self-assessment process.

Significant Inspection Findings

Some of the significant inspection findings resulting from recent inspections are summarized below. The staff continues to exercise all aspects of the Reactor Oversight Process including, when appropriate, overseeing plants that are in extended shutdowns, using the process in Inspection Manual Chapter 0350.

<u>First Finding</u>: On February 16, 2002, during a refueling outage, the licensee for Davis-Besse found axial flaw indications in three control rod drive mechanism nozzles, as result of the inspections performed pursuant to NRC Bulletin 2002-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles." On March 5 and 6, 2002, workers at Davis-Besse were repairing control rod drive penetration Nozzle 3, and discovered a large cavity, a significant loss of metal adjacent to the control rod drive nozzle in the reactor vessel head, that apparently resulted from boric acid corrosion of the reactor vessel head due to leakage from the cracks in Nozzle 3. The degraded area covered about 30 square inches where the thick low-alloy structural steel was corroded away, leaving only the thin stainless steel cladding layer as a pressure boundary for the reactor coolant system. The licensee reported these findings to NRC. On March 12, 2002, NRC dispatched an Augmented Inspection Team to Davis-Besse under NRC Management Directive 8.3, "NRC Incident Investigation Program." The team was chartered to find out the facts and circumstances related to the significant degradation of the reactor pressure vessel head discovered by the licensee. The team's results and subsequent special inspections were documented in NRC inspection reports (NRC Augmented Inspection

Team - Degradation of The Reactor Pressure Vessel Head - Report No. 50-346/02-03 (DRS) dated May 3, 2002).(For inspection reports, see NRC's public Web site at http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/listofrpts_body.html#davi.)

The performance deficiency, associated with the control rod drive penetration cracking and reactor pressure vessel head degradation, was the licensee's failure to properly implement its boric acid corrosion control and corrective action programs, which allowed reactor coolant system pressure boundary leakage to occur undetected for a long time, and causing degradation of the head and circumferential cracking of the nozzles. NRC assessed the significance of the performance deficiency using the Significance Determination Process.

The performance deficiency increased the risk of reactor core damage through a loss-of-coolant accident caused by either a rupture in the exposed cladding in the head cavity or a circumferential crack causing a control rod drive mechanism nozzle to eject. The significance analysis of the head cavity indicated significance in the red range (change in core damage frequency > 10^{-4} per reactor-year). NRC's significance analysis of the as-found circumferential crack and potential for crack growth indicated significance in the yellow to red range (change in core damage frequency in the range of low 10^{-5} to low 10^{-4} per reactor-year). Consequently, NRC determined that the performance deficiency resulting in the head degradation and control rod drive mechanism nozzle cracking had high safety significance in the red range that will result in increased NRC inspection and other NRC actions. Since Davis-Besse was under the Inspection Manual Chapter 0350 process, NRC staff performed special augmented inspections to review licensees' corrective actions, root causes, and preventive measures.

<u>Second Finding</u>: At Davis-Besse, a violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," was identified for the failure to effectively implement corrective actions for design control issues related to deficient containment coatings, uncontrolled fibrous material, and other debris. This deficiency resulted in the inability of the emergency core cooling system sump to perform its safety function under certain accident scenarios owing to clogging of the sump screen.

The significance of the finding is based on the increased likelihood of the emergency core cooling systems to fail following a loss of coolant accident. After injecting cooling water into the reactor following an accident, those systems begin recirculating cooling water to the reactor from the containment sump. The unqualified coatings, fibrous material and other debris could clog the screen on the sump, blocking the water supply to the emergency core cooling system pumps. This increased likelihood of emergency core cooling system failure increases the probability of damage to the reactor following an accident. The increased probability was evaluated using the Davis-Besse Standardized Plant Analysis Risk Model. The results of the evaluation indicated an increase in reactor core damage frequency, using the Significance Determination Process, in the yellow range (i.e., 10^{-5} to 10^{-4} per reactor-year change in CDF). This increased risk existed from the time the facility began operation in 1977 until early 2002.

<u>Third Finding</u>: At Point Beach Unit 2, a violation was identified, in part, through a self-revealing event when decreased auxiliary feedwater pump recirculation flow was noted during post-maintenance testing. Subsequent licensee and NRC review determined that the licensee had installed incorrectly designed orifices in each of the pump recirculation lines. The orifices, due to small clearances, were susceptible to plugging. The primary causes of this finding were inadequacies in the licensee's design process and the licensee's implementation of the process, including identifying system design requirements and developing supporting safety evaluations.

Following installation of the inadequately designed orifices, the entire auxiliary feedwater system was susceptible to a common mode failure. Failure of auxiliary feedwater during several initiating events could lead to core damage. The significance of this issue, evaluated using the Significance Determination Process, was determined to have high safety significance (red; change in core damage frequency > 10^{-4} per reactor-year). The installation of the incorrectly designed orifices in the recirculation lines was a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control."

Fourth Finding: At Point Beach Units 1 and 2, the licensee identified a potential common mode failure of the auxiliary feedwater pumps due to operator actions specified in plant procedures. A NRC inspection team identified that procedural guidance was inadequate to prevent such a common mode failure. In addition, the team identified that the licensee had seven opportunities, from 1981 through 1997, to identify the problem and take appropriate corrective actions. The failures to provide adequate procedural guidance and to take appropriate corrective actions are both a violation of 10 CFR Part 50, Appendix B, Criteria V and XVI. A common mode failure of the auxiliary feedwater pumps would result in substantially reduced mitigation capability for safely shutting down the plant in response to certain transients. The significance was determined to be high (red; change in core damage frequency > 10^{-4} per reactor-year), largely owing to the relatively high initiating event frequencies associated with the involved transients and the high likelihood of improper operator actions due to the procedural inadequacies.

A supplemental inspection was performed to evaluate corrective actions to the auxiliary feedwater pumps because of the plugging of the recirculation line pressure reduction orifices. The inspection also reviewed the corrective action, emergency preparedness, and engineering programs. The overall root and contributing causes for the two red auxiliary feedwater findings were the lack of understanding of the design, corrective action program weaknesses, and poor operations/engineering interface. And while overall, corrective actions taken for the findings were adequate, several important corrective actions to prevent recurrence had not been adequately implemented.

6.2.3 Industry Trends Program

6.2.3.1 Description of the Program

The NRC staff implemented the Industry Trends Program in 2001, and is continuing to develop the program as a means to confirm that the nuclear industry is maintaining the safety of operating power plants and to increase public confidence in the efficacy of NRC's processes. NRC uses industry-level indicators to identify adverse trends and provide feedback to the Reactor Oversight Process. It assesses adverse trends for safety significance and responds as necessary to any safety issues identified. One important output of this program is to report to Congress each year on the performance goal measure of "no statistically significant adverse industry trends in safety performance" as part of NRC's Performance and Accountability Report. Based on the information currently available from the industry-level indicators originally developed by the former Office for Analysis and Evaluation of Operational Data and the Accident Sequence Precursor Program, no statistically significant adverse industry trends have been identified through FY 2003.

The staff is continuing to use the indicators while it develops additional industry-level indicators that are more risk-informed and better aligned with the cornerstones of safety in the Reactor Oversight Process. These additional indicators will be developed in phases and qualified for use in the Industry Trends Program.

The Reactor Oversight Process uses both plant-level performance indicators and inspections to provide plant-specific oversight of safety performance, whereas the industry trends program provides a means to assess overall industry performance using industry-level indicators. NRC evaluates issues that are identified from either program using information from agency databases, and addresses those assessed as having generic safety significance by existing NRC processes, including generic safety inspections under the Reactor Oversight Process, the generic communications process, and the generic safety issue process.

6.2.3.2 Recent Experience and Examples

As part of the program, the staff adopted a statistical approach using "prediction limits" to provide a consistent method to identify potential short-term year-to-year emergent issues before they manifest themselves as long-term trends. Three indicators: automatic scrams while critical, safety system actuations, and equipment-forced outage rate reached or exceeded their prediction limit during FY 2003. For more details, see SECY-04-0052, "FY 2003 Results of the Industry Trends Program for Operating Power Reactors and Status of Ongoing Development," available on NRC's public Web site in the electronic reading room.

6.2.4 Accident Sequence Precursor Program

6.2.4.1 Description of the Program

The Accident Sequence Precursor Program views U.S. nuclear plant operating experience from a perspective of safety significance. The primary aim of the program is to systematically identify, document, and rank operating events that are potentially most significant to cause inadequate core cooling and core damage. The secondary objectives of the program are (1) categorizing precursors for plant-specific and generic implications, (2) providing a measure to trend the risk of core damage, and (3) providing a partial check on dominant core damage scenarios predicted by probabilistic risk analyses.

An accident sequence precursor is an historically observed element or condition in a postulated sequence of events leading to some undesirable consequence. For purposes of the program, the undesirable consequence is usually severe core damage. Applying probabilistic risk techniques, the staff evaluates initiating events, degradation of plant conditions, and failures of safety equipment that could increase the probability of postulated accident sequences. Accident sequences of interest are those that, if additional failures occurred, would have resulted in inadequate core cooling that could have caused severe core damage. For example, a loss-of coolant accident with a reported failure of a high-pressure injection system may be postulated. In this example, the precursor would be the failure of the high-pressure injection system.

To identify potential precursors, the staff reviews licensee event reports or other documents such as inspection reports and reports by incident investigation teams of plant problems, equipment failures, or other operational incidents. The staff analyzes the events it considers to be potential precursors and calculates the conditional probability (i.e., probability, given an initiating condition) of core damage by mapping failures observed during the event onto the program's core damage models. The staff identifies and documents the events with conditional probabilities of subsequent severe core damage greater than 1×10^{-6} as precursors. It considers precursors with values greater than 1×10^{-4} important.

The Accident Sequence Precursor Program began in 1979. Since then, the staff has evaluated and documented in excess of 600 precursors from reported experience for 1969 through 2003.

The precursor events identified by the Accident Sequence Precursor Program form a unique database of historical system failures, multiple losses of redundancy, and infrequent core damage initiators. The program has evolved to the point where NRC routinely uses the methodology and results. NRC continues to improve the methodology to better account for plant design and operational differences, human reliability, changes in equipment, and to provide user-friendly analytical tools. For example, NRC has recently incorporated the analysis of modeling and data uncertainty into event analysis, and has established a more complete set of accident sequences into the risk models. Planned future improvements include procedures to better evaluate risk from external event initiators, low power and shutdown modes of operation, and evaluations of containment response and large early release frequencies.

Commercial nuclear power reactors in the United States now have a combined total of more than 2000 years of operating experience. The Accident Sequence Precursor Program uses information gained from this experience to continually assess plant operation. The assessment helps to determine how well plant designs and capabilities can cope with actual operational events or conditions.

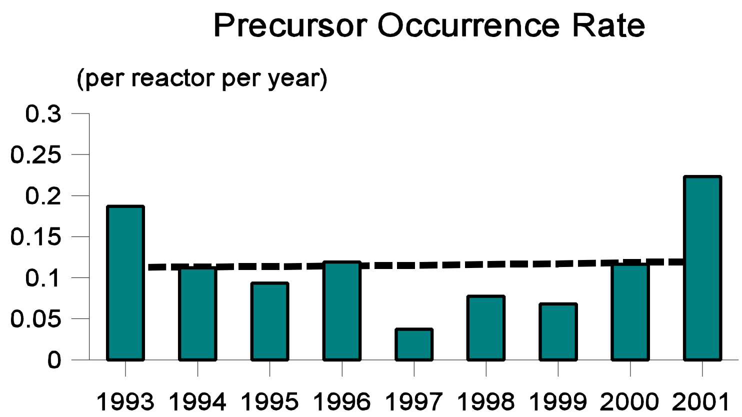
6.2.4.2 Recent Experience and Examples

The occurrence rate (number of precursors per reactor year) for all precursors by fiscal year is shown in Figure 3. NRC did not observe any statistically significant trend in the occurrence rate of precursors during the 1993–2001 period. Although the staff did not detect a trend based on data for 1993 through 2001, data for the 5-year period from 1997 through 2001 show an increasing number of precursors. The staff will investigate the nature of the precursors to determine whether there is an explanation for the relatively low number of precursors between 1997 and 1999, and for an increasing number of potential precursors in 2000 and 2001.

Over the past two years, NRC has worked on several complex analyses in the identification of potential precursors. These analyses include the emergency service water debris clogging condition at D.C. Cook, the potential common-mode failure of all auxiliary feedwater pumps at Point Beach Units 1 and 2, the reactor pressure vessel head degradation at Davis-Besse, the through-wall cracks in the control rod drive mechanism housing at several plants, and the losses of offsite power at nine plants caused by the Northeast U.S. blackout on August 14, 2003. The analysis of the later three events are ongoing. The events at Point Beach and D.C. Cook are described below.

The Point Beach analysis involves a January 2002 event in which the licensee identified a design deficiency in the auxiliary feedwater pumps air-operated minimum flow recirculation valves. The valves fail closed on loss of instrument air which could potentially lead to pump deadhead conditions and a common mode, non-recoverable, failure of the auxiliary feedwater pumps. Because the pressurizer power-operated relief valves are also dependent on instrument air, a loss of instrument air event may also result in the loss of feed and bleed cooling capability. The program's results for this condition showed a change in the mean core damage probability of 6×10^{-4}. The important accident sequences are dominated by loss of service water, loss of offsite power, loss of instrument air, and seismic events.

The D. C. Cook analysis involves the entrainment of debris in the emergency service water system as the result of a deformed strainer basket associated with the emergency service water pump. Because of system cross-ties, both the Unit 1 and Unit 2 emergency service water systems could be affected by this strainer condition. During the August 2001 event, the plant experienced low emergency service water flow to both the Unit 1 and Unit 2 emergency diesel generator heat exchangers as a result of clogging of the emergency service water system. This condition resulted in a change in the mean core damage probability of approximately 1×10^{-5}. The dominant sequences associated with this scenario involve a loss of offsite power initiating event, common cause failure of the emergency diesel generators due to debris clogging of the emergency service water system, and failure to recover offsite power before battery depletion. One of the bigger challenges of this analysis is the determination of equipment functionality given various degrees of debris intrusion. The program analysis accounted for this by the performance of uncertainty and sensitivity analyses.



Occurrence rate for all precursors by fiscal year.
No statistically significant trend in the occurrence rate of precursors was observed during the 1993-2001 period. (Data is current as of 06/30/04)

Figure 3: Occurrence rate for all precursors by fiscal year.

6.2.5 Program for Resolving Generic Issues

The program for resolving generic issues is governed by Management Directive 6.4, "Generic Issues Program," issued in December 2001. In addition, a complete list of all generic issues, including a description, screening writeup, and disposition of each issue, is maintained in NUREG-0933, "A Prioritization of Generic Safety Issues." The most recent version of NUREG-0933, published in June 2003 and including Supplements 1–27, is available on NRC's public Web site at http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0933/.

The sources of generic issues include safety-related research, risk-assessment analyses, operating experience, event investigations, and public and industry concerns. In the past, the generic issues program has addressed major safety issues affecting multiple facilities, including the Three Mile Island (TMI) Action Plan requirements, the Chernobyl Issues, the Unresolved Safety Issues, and all other multi-plant actions.

To assess each issue, NRC has developed a screening assessment method that is predicated on risk. In screening issues, NRC considers both operating and future plants. NRC identifies issues that could significantly improve safety, and, where appropriate, also considers the cost of resolution. The purpose of the screening is to allocate resources to the safety issues that have a high potential for reducing risk. Issues that demand immediate attention (e.g., issues that may require plant shutdown) are generally not addressed by means of the Generic Issues Program, because the agency must quickly make decisions on such issues.

The generic issues screening activity primarily focuses on issues that affect all, several, or a class of nuclear power plants, and issues that may lead to improvements in safety. The screening assessment method can also be used to identify changes in requirements that could significantly reduce the effect (usually the cost) on licensees without substantially changing public risk. NRC classifies issues of this type as burden reduction issues to clearly differentiate them from those that improve the safety of nuclear power plants.

Before issuing Management Directive 6.4, NRC had considered an issue to be resolved when its review resulted in the imposition of regulatory requirements or guidance (by rule, standard review plan change, or equivalent), or when it resulted in a documented authoritative decision that no change in requirements was warranted. This policy was changed under Management Directive 6.4, in that generic issues that result in new or revised regulatory requirements are not considered to be resolved until these new or revised requirements have actually been implemented and, where appropriate, verified (e.g., by an inspection program) at the affected plants.

6.2.6 Rulemaking

NRC's regulations, also called rules, impose requirements that licensees must meet to obtain or retain a license or certificate to use nuclear materials or operate a nuclear facility. The process of developing regulations is called rulemaking. NRC's technical staff usually initiates a proposed rule or a change to a rule because of a perceived need to protect the public health and safety. However, any member of the public may petition NRC to develop, change, or rescind a rule. The impetus for a *proposed rule* could be a requirement issued by the Commission, a petition for rulemaking submitted by a member of the public, or research results that indicate a need for a rule change. The proposed rule is published in the U.S. Government's *Federal Register* for public comment. Once the public comment period has closed, the staff analyzes the comments, makes any needed changes, and forwards the *final rule* for approval, signature, and publication in the *Federal Register*. Each final rule that involves significant matters of policy is sent to NRC Commissioners for approval. Once approved, the final rule is published in the *Federal Register* and usually becomes effective 30 days later. For a list of NRC rules published in the past 90 days, including the text of final rules, see Final Rules and Policy Statements at http://ruleforum.llnl.gov/cgi-bin/library?source=*&library=final_lib&file=*. Significant nuclear reactor-related rules issued since the previous *U.S. National Report* are summarized in the Introduction to this report.

In some instances, where the agency wishes to have substantial public input to even develop a proposed rule, the staff may publish an *Advanced Notice of Proposed Rulemaking* in the *Federal Register* or conduct one or more public meetings. The notice solicits comment well in advance of the proposed rulemaking stage. The need for some action is described but only broad concepts are discussed for a proposed action.

NRC welcomes public participation and developed RuleForum to provide an easy means for members of the public to access and comment on NRC rulemaking actions. Accessible through NRC's public Web site, RuleForum contains proposed rulemakings that have been published by NRC in the *Federal Register*, petitions for rulemaking, and other types of documents related to rulemaking proceedings. NRC has been improving its rulemaking process, including allowing the electronic submittals of documents related to rulemaking.

6.2.7 Decommissioning

The decommissioning process consists of a series of integrated activities, beginning with the nuclear facility in transition from "active" to "decommissioning" status and concluding with the termination of the license and release of the site. Once a licensee decides to shut down a nuclear power plant, it must notify NRC in writing within 30 days of the decision. After removing the fuel permanently from the reactor vessel and certifying the event to NRC, the licensee may no longer operate the reactor or place fuel back into the reactor vessel. The licensee must then submit a "Post-Shutdown Decommissioning Activity Report" within 2 years following permanent cessation of operations. This report is then made available to the public and a public meeting is held near the plant. The report must address planned decommissioning activities, and contain a schedule of significant milestones and documentation that demonstrates that environmental impacts have been considered. Subsequently, a licensee must submit a license termination plan no later than two years before the expected license termination, addressing in detail the final radiation survey, site characterization and remediation, and any new information. Before NRC approves the plan, it provides an opportunity for a hearing and holds a public meeting near the plant.

Currently, licensees may choose one of two methods for decommissioning their plants: DECON or SAFSTOR. Under DECON (immediate dismantlement), soon after closing the nuclear plant, the licensee removes equipment, structures, and parts of the plant containing radioactive contaminants or decontaminates them to a level that permits release of the property and termination of NRC license, under the provisions of regulations in 10 CFR Part 20, Subpart E. Under SAFSTOR, the licensee maintains a nuclear plant in a condition that allows the radioactivity to decay, monitors the plant, and later dismantles it. Currently, ENTOMB is not considered a viable option in the U.S. for reactor decommissioning.[d]

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[d] ENTOMB is the practice of encasing radioactive contaminants in a structurally sound material, such as concrete, and maintenance and monitoring until the radioactivity decays to a level permitting release of the property.

During decommissioning, special care is needed in handling reactor and plant components that are contaminated during operations and subsequent decommissioning work. Activated and contaminated materials must be shipped to a low-level radioactive waste disposal site for burial. 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 unrestricted use. NRC reviews and approves license termination plans, conducts inspections, processes license amendments, and monitors the status of activities to ensure that radioactive contamination is reduced or stabilized. This monitoring ensures that safety requirements are being met throughout the process. NRC oversees the decommissioning of nuclear reactors through inspections that emphasize radiological controls, management, compliance with procedures, spent fuel pool operations, and the safety review program. To be acceptable, decommissioning must be completed generally within 60 years of permanent cessation of operations. NRC will consider a period longer than 60 years only when necessary to protect public health and safety.

Decommissioning is addressed in the design criteria for new facility construction in 10 CFR 20.1406, “Minimization of contamination.” Furthermore, the Safety Standards on Decommissioning promulgated by the International Atomic Energy Agency (IAEA) cite the following considerations for future decommissioning in the conceptual design of nuclear facilities, which the United States has supported:

- WS-R-2: Predisposal Management of Radioactive Waste including Decommissioning, 2000
- WS-G-2.1: Decommissioning of Nuclear Power Plants and Research Reactors, 1999
- WS-G-2.2: Decommissioning of Medical, Industrial and Research Facilities, 1999
- WS-G-2.4: Decommissioning of Nuclear Fuel Cycle Facilities, 2001

Spent fuel can remain stored in the spent fuel pool or in dry cask storage facilities until a geologic repository is built and operating. 10 CFR Parts 50 and 72 contain licensing requirements to maintain spent fuel integrity. The Commission, in issuing its Waste Confidence Decision, found that spent fuel can be stored safely in spent fuel pools or in on site independent spent fuel storage installations without significant environmental impacts for at least 30 years beyond the licensed life (which may include the term of a renewed license).

10 CFR Part 60, “Disposal of high-level radioactive wastes in geologic repositories,” specifies the requirements for a geologic repository. Paragraph 60.132, “Additional design criteria for surface facilities in the geologic repository operations area,” includes provisions for designing to facilitate decontamination or dismantlement.

Also, the regulations specify that an independent spent fuel storage installation must be designed for decommissioning. 10 CFR 72.130 specifies the criteria for decommissioning; these criteria are evaluated as part of the licensing process. Provisions must be made to facilitate decontamination of structures and equipment, minimize the quantity of radioactive wastes and contaminated equipment, and facilitate the removal of radioactive wastes and contaminated materials at the time the installation is permanently decommissioned. 10 CFR 72.54 also specifies provisions for financial assurance, final radiation surveys, recordkeeping, waste disposition, the role and effect of short duration decay, and the feasibility of completing decommissioning in a 24-month period.

6.2.8 Research Program

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 identify, evaluate, and resolve safety issues, to ensure that an independent technical basis exists to review licensee submittals, to evaluate operating experience and results of risk assessments for safety implications, and to support the development and use of risk-informed regulatory approaches. In conducting the Reactor Safety Research program, NRC anticipates challenges posed by the introduction of new technologies and changing regulatory demand. NRC continues to seek out opportunities to leverage its resources through domestic and international cooperative programs, and provide enhanced opportunities for stakeholder involvement and feedback on its research program. In addition, the agency will conduct research to address technical issues that it anticipates will arise during its review of advanced reactor designs. The Reactor Safety Research program is directly aligned with NRC's performance goals. Research programs are also discussed under Article 6, (the Accident Precursor Program), and Article 10, (SAPHIRE and ATHEANA).

6.2.9 Programs for Public Participation, Handling Petitions, Allegations, and Differing Professional Opinions

NRC values public participation in its regulatory processes. Toward this end, NRC provides the diverse body of stakeholders (general public, Congress, other Federal, State and local governments, Indian Tribes, industry, technical societies, the international community and citizen groups) clear and accurate information about its role, and opportunities to participate in the agency's regulatory programs. Numerous NRC programs and processes provide the public with accessibility to NRC staff and resources, seek to make communication with stakeholders more clear, accurate, reliable, objective and timely, and help ensure the reporting of the performance of nuclear power plants performance is open and objective. NRC has developed Web pages to disseminate timely, accurate information regarding issues of interest to the public or events at nuclear facilities. NRC elicits public involvement early in the regulatory process so as to address any safety concerns in a timely manner. In addition to the formal petition and hearing processes integrated into the licensing program, the agency uses feedback forms at public meetings to obtain public input.

Fostering an environment in which safety issues can be openly identified without fear of retribution is of paramount interest to NRC. Examples of tools NRC has established for the public and industry and NRC employees to use to raise safety concerns include the petition process under 10 CFR 2.206, the safety conscious work environment policy, the allegation program, and the program for Differing Professional Opinions.

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

NRC encourages workers in the nuclear industry to take their concerns directly to their employers (often the licensee or plant operator) and is particularly vigilant about a safety-conscious work environment that encourages such reporting. It is NRC's expectation that licensees establish and maintain a work environment where there is no fear of retribution of licensee action following such reporting. Additionally, the workers or any member of the public can bring their safety concerns directly to NRC. The agency established a toll-free safety hotline for reporting such safety concerns, and NRC management, staff, and inspectors, including the resident inspectors at plant sites, are trained and available to receive such concerns.

Historically, approximately 500 potential safety or regulatory issues have been reported directly to NRC's allegation program each year by industry workers or members of the public. NRC developed the allegation program to establish a formal process for evaluating and responding to each issue. (See http://www.nrc.gov/reading-rm/doc-collections/nuregs/brochures/br0240/r1/index.html for further details.) The primary purpose of the program is to provide an alternative method for individuals to raise safety or regulatory issues and to have them addressed. About 60 percent of the issues that are reported to NRC are from licensee employees, employees of contractors to licensees, or former employees of licensees or contractors. Given sufficient information, NRC will evaluate each issue to determine whether it can verify the issue and, if so, the effect of the issue on plant safety. The evaluation is either an engineering review, inspection or investigation by NRC staff, or an evaluation by the licensee that is reviewed by NRC staff. Historically, NRC has been able to substantiate 25 to 30 percent of the issues received. If the evaluation reveals a violation of regulatory requirements, the agency takes appropriate enforcement actions. Additionally, NRC informs the individual who raised the issue of the results of its evaluation in writing.

NRC's revised program for Differing Professional Opinions as defined in Management Directive 10.159, allows NRC employees to bring safety concerns or other issues important to the mission of NRC to senior NRC management and, where appropriate, to the Commission. The Differing Professional Opinion program supports the agency policy to maintain a working environment that encourages employees to make known their best professional judgments even though they may differ from the prevailing staff view, disagree with a management decision or policy position, or take issue with a proposed or established agency practice involving technical, legal, or policy issues. The program ensures the full consideration and timely disposition of Differing Professional Opinions by affording an independent, impartial review by knowledgeable personnel who review the concern to determine the need for regulatory action. The program may be used without fear of retaliation, pressure, penalty, or unauthorized divulgence.

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 licences

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

This section explains the legislative and regulatory framework governing the U.S. nuclear industry. It discusses the provisions of that framework for establishing national safety requirements and regulations, and systems for licensing, inspection, and enforcement. The framework and provisions have not changed since the previous *U.S. National Report* was issued.

7.1 Legislative and Regulatory Framework

The Atomic Energy Act of 1954, passed by Congress and signed by the President, provided the framework for all subsequent regulation of nuclear installations. However, as is generally the case with most laws, this Act provided general principles and concepts, and left it to the regulatory body (NRC) to address the details by issuing specific regulations.

7.2 Provisions of the Legislative and Regulatory Framework

7.2.1 National Safety Requirements and Regulations

The regulations on nuclear installations in the United States are governed by the Atomic Energy Act of 1954 (as amended) and the Energy Reorganization Act of 1974. [The Energy Reorganization Act abolished the U.S. Atomic Energy Commission, and created NRC and the U.S. Energy Research and Development Administration (ERDA). ERDA was subsequently incorporated into the U.S. Department of Energy (DOE).] NRC administers these statutes in licensing commercial nuclear installations in the United States. In addition, the following statutes bear substantially on the practices and procedures of the Commission:

- Administrative Procedure Act
- Chief Financial Officers Act of 1990
- Clean Air Act of 1977
- Coastal Zone Management Act
- Comprehensive Environmental Response, Compensation, and Liability Act of 1980

- Endangered Species Act
- Federal Advisory Committee Act
- Federal Water Pollution Control Act (also known as the Clean Water Act of 1977)
- Freedom of Information Act
- Government in the Sunshine Act
- Government Paperwork Elimination Act
- Inspector General Act
- Low-Level Radioactive Waste Policy Amendments Act of 1985
- National Environmental Policy Act of 1969
- National Historic Preservation Act
- Nuclear Waste Policy Act of 1982
- Price Anderson Act of 1957
- Privacy Act
- Resource Conservation and Recovery Act of 1976
- Toxic Substances Control Act of 1976
- Uranium Mill Tailings Radiation Control Act of 1978
- West Valley Demonstration Project Act of 1980
- Wild and Scenic Rivers Act

Commercial nuclear installations in the United States must be licensed by NRC. (Some government facilities that are operated by or for the DOE, are exempt from licensing by the requirements of the Atomic Energy Act and the Energy Reorganization Act.) The following facilities must be licensed:

- Nuclear reactors (power, test, and research)
- Uranium mills
- Solution recovery plants (milling)
- Uranium dioxide (UO_2) and mixed oxide (MOX) fuel fabrication plants
- Spent fuel storage (interim)
- High-level waste and spent fuel geologic repositories
- Low-level waste burial grounds
- Fuel reprocessing plants
- Isotopic separation (enrichment) plants

(This list does not include medical and materials licensees.) Rules and regulations governing the licensing of these facilities are contained in Title 10 of the *U.S. Code of Federal Regulations* (10 CFR). The first 200 parts of 10 CFR (Parts 0 -199) apply to NRC.

Although the licensing process is similar in many respects for reactors, separation facilities, reprocessing plants, and nuclear waste storage and disposal facilities, the following discussion describes the licensing practices for nuclear power plants.

7.2.2 Licensing of Nuclear Installations

Chapter 10, Section 103, “Commercial Licenses,” of the Atomic Energy Act of 1954 grants NRC authority to issue licenses for nuclear reactor facilities. In addition, Section 103 states that such licenses are subject to such conditions as NRC may by rule or regulation establish to effectuate the purposes and provisions of the Atomic Energy Act. Section 103b. states the following:

> The Commission shall issue such licenses on a nonexclusive basis to persons applying therefore (1) whose proposed activities will serve a useful purpose

> proportionate to the quantities of special nuclear material or source material to be utilized; (2) who are equipped to observe and who agree to observe such safety standards to protect health and to minimize danger to life or property as the Commission may by rule establish; and (3) who agree to make available to the Commission such technical information and data concerning activities under such license as the Commission may determine necessary to promote the common defense and security and to protect the health and safety of the public.

Section 182, "License Applications," states the following:

> Each application for a license hereunder shall be in writing and 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, the character of the applicant, the citizenship of the applicant or any other qualifications of the applicant as the Commission may deem appropriate for the license.

Section 189a. of the Atomic Energy Act provides affected parties with hearing rights in proceedings for the granting, suspending, revoking, or amending of a license or construction permit.

Hearings, which are used in licensing proceedings for production and utilization facilities (e.g., nuclear power plants), are held under procedural rules stated in 10 CFR Part 2, "Rules of Practice for Domestic Licensing Proceedings and Issuance of Orders," especially Subpart C, "Rules of General Applicability." The staff participates as a party in all formal hearings, and may also participate as a party in informal hearings. Hearings are usually held before a three-member Atomic Safety and Licensing Board, which is generally composed of one lawyer and two technical members.

A party may appeal the decision of a Licensing Board to the Commission itself. Review by the Commission is discretionary, not mandatory. The Commission ordinarily reviews only novel or complex issues or important legal, policy, or health and safety questions. In deciding whether to grant a review of a particular case, and in writing its appellate decisions, the Commission is assisted by the Office of Commission Appellate Adjudication.

The licensing process is described in greater detail in Article 18. There are two alternative approaches. The traditional approach under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires two steps. NRC reviews a preliminary application and grants a construction permit, and later reviews the final application and grants an operating license. All current operating plants in the United States were licensed according to this process.

In 1989, the Commission established an alternative licensing system, published in 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," that provides for certified standard designs and combined licenses that resolve design issues before construction, and early site permits that resolve siting issues before construction. The basic concept underlying 10 CFR Part 52 is that nuclear reactor designs can be approved through generic rulemaking. Once approved, an applicant can use them in applications for permission to build and operate nuclear power plants without the necessity to relitigate, in individual hearings, the issues resolved in the rulemaking. Moreover, the criteria for determining whether the plant had been built as specified would be determined and approved before construction. Thus completed, the plant could begin operation without a second hearing, provided that it satisfied the acceptance criteria. To the extent possible, issues would be litigated before construction, not after construction is largely complete, when the consequences of delay are much greater. In adopting 10 CFR Part 52, the Commission used the latitude allowed by law to streamline licensing.

The Energy Policy Act of 1992 codified major parts of 10 CFR Part 52 and directed NRC to issue implementing regulations. The legislation provided that the Commission may allow a completed plant to operate during the pendency of any post-construction hearing, provided that certain safety findings can be made. In addition, the legislation made it clear that the Commission could use either formal or informal procedures for such post-construction hearings.

7.2.3 Inspection and Assessment

Under the Atomic Energy Act of 1954, NRC has the authority to inspect nuclear power plants in its role of protecting public heath and safety and the common defense and security. 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, NRC assesses whether activities are properly conducted and equipment is properly maintained to ensure safe operations. NRC integrates inspection results into its overall evaluation of licensee performance, as discussed in Article 6. When NRC discovers a safety problem or failure to comply with requirements, it requires prompt corrective action by the licensee, reinforcing it, if necessary, by enforcement action.

7.2.4 Enforcement

NRC's enforcement jurisdiction is drawn from the Atomic Energy Act of 1954, as amended, and the Energy Reorganization Act of 1974, as amended.

Section 161 of the Atomic Energy Act authorizes NRC to conduct inspections and investigations, and to issue orders as may be necessary or desirable to promote the common defense and security or to protect health or to minimize danger to life or property. Section 186 authorizes NRC to revoke licenses under certain circumstances (e.g., for material false statements, for a change in conditions that would have warranted NRC refusal to grant a license on an original application, for a licensee's failure to build or operate a facility in accordance with the terms of the permit or license, and for violation of an NRC regulation). Section 234 authorizes NRC to impose monetary civil penalties not to exceed $100,000 per violation per day; however, that amount is adjusted every four years by the Federal Civil Penalties Inflation Adjustment Act and is currently $120,000. In addition to the provisions mentioned in Section 234, Sections 84 and 147 authorize the imposition of civil penalties for violations of regulations implementing those provisions. Section 232 authorizes NRC to seek injunctive or other equitable relief for violation of regulatory requirements.

Section 206 of the Energy Reorganization Act authorizes NRC to impose civil penalties for knowing and conscious failures to provide certain safety information to NRC.

Chapter 18 of the Atomic Energy Act provides for varying levels of criminal penalties (i.e., monetary fines and imprisonment) for willful violations of the Act and regulations or orders issued under Sections 65, 161b, 161i, or 161o of the Act. Section 223 provides that criminal penalties may be imposed on certain individuals who are employed by firms constructing or supplying basic components of any utilization facility if the individual knowingly and willfully violates NRC requirements in a manner that could significantly impair a basic component.

Section 235 provides that criminal penalties may be imposed on persons who interfere with nuclear inspectors. Section 236 provides that criminal penalties may be imposed on persons who attempt to or cause sabotage at a nuclear facility or to nuclear fuel. Alleged or suspected criminal violations of the Atomic Energy Act are referred to the U.S. Department of Justice for appropriate action.

The procedures that NRC uses in exercising its enforcement authority are specified in 10 CFR Part 2, "Rules of Practice for Domestic Licensing Proceedings and Issuance of Orders," Subpart B, "Procedure for Imposing Requirements by Order, or for Modification, Suspension, or Revocation of a License, or for Imposing Civil Penalties." The scope of Subpart B includes the procedures described below.

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

10 CFR 2.202, "Orders," states the procedure for issuing orders. In accordance with 10 CFR 2.202, NRC may decide to issue an order to institute a proceeding to modify, suspend, or revoke a license or to take other action against a licensee or other person subject to the jurisdiction of NRC. The licensee or any other person adversely affected by the order may request a hearing. NRC is authorized to make orders immediately effective if required 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 Commission's jurisdiction to determine whether an order should be issued or other enforcement action should be taken. The demand does not provide hearing rights, as only information is being sought. A licensee must answer a demand. An unlicenced person may answer a demand by either providing the requested information or explaining why the demand should not have been issued.

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

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, NRC. It also explains how the functions of NRC are separate from those of bodies responsible for promoting and using nuclear energy (i.e., DOE). This update reports the establishment of a new office, the Office of Nuclear Security and Incident Response. In addition, the section describing NRC's international responsibilities and activities has been expanded.

8.1 The Regulatory Body

This section explains NRC's mandate, authority and responsibilities, status as an independent regulatory agency, structure, and staffing and resources

8.1.1 Mandate

As discussed in Article 7, NRC was created as an independent regulatory agency in January 1975, with the passage of the Energy Reorganization Act of 1974. In giving NRC an exclusively regulatory mandate, the statute reflected (in part) a Congressional judgment that the commercial nuclear power industry (which, at that time, was expanding rapidly and was expected to grow still more) had reached a point at which the full-time attention of an exclusively regulatory agency was warranted. A second major reason for NRC's creation was a developing public concern that regulatory responsibilities were overshadowed by the promotion of nuclear power at the Atomic Energy Commission. By 1975, a large segment of the U.S. population was skeptical about the role of nuclear power in the Nation's energy mix for a variety of reasons. Specifically, these reasons included the safety of nuclear plants, security concerns (including the threat that terrorists would attack nuclear facilities or make weapons from stolen nuclear materials), the proliferation of nuclear weapons-making capacity around the world, and the lack of a solution to the problem of high-level radioactive waste disposal. It was hoped that by limiting NRC responsibilities to regulation, making it an independent agency, and choosing the multi-member bipartisan Commission format, greater public confidence would ensue in governmental decisionmaking in the nuclear area.

8.1.2 Authority and Responsibilities

This section discusses the scope of authority of NRC, its international responsibilities, and status as an independent regulatory body.

8.1.2.1 Scope of Authority

NRC's mission is to ensure that the civilian uses of nuclear energy and materials in the United States are conducted with proper regard for public health and safety, national security, environmental concerns, and (in the case of the initial licensing of nuclear power plants) the antitrust laws. The basic charter for these regulatory responsibilities is the Atomic Energy Act of 1954, through which Congress created a national policy of developing the peaceful uses of atomic energy. That statute has been amended over the years to deal with developing technology and changing perceptions of regulatory needs. For example, antitrust reviews were added in 1970, the same year that the National Environmental Policy Act imposed broad new responsibilities on Federal agencies. Other more specialized statutes prescribe NRC's duties with regard to high-level radioactive waste, low-level radioactive waste, mill tailings, environmental reviews, nonproliferation, anti-terrorism, and import/export.

The Atomic Energy Act of 1954 has been described by the courts as "almost unique" in the degree of discretion that it confers on NRC to make the decisions it believes to be best, using the procedures it considers most suitable. Only very rarely has an NRC decision been overturned on substantive grounds. The language of the Act granting NRC discretionary authority contrasts with the prescriptive language of later statutes governing such agencies as the U.S. Environmental Protection Agency (EPA) and the Occupational Safety and Health Administration.

NRC's licensing authority extends to other Federal agencies (such as the Tennessee Valley Authority, which operates nuclear power plants) and the military's use of radiopharmaceuticals in its hospitals. NRC's responsibilities include both safety and "safeguards" through which the agency ensures the security of commercial nuclear facilities and materials against radiological sabotage and thefts.

In addition, NRC is authorized to relinguish certain regulatory authorities for most nuclear materials (although not nuclear power plants, fuel facilities, or Federal agencies, and not critical mass quantities of special nuclear material), for which authority can be assumed by States (Agreement States) who enter into agreements with NRC. More than half of the States are Agreement States, and together they administer another 16,000 materials licenses.

NRC has no authority to regulate either accelerator-produced radioactive materials, or materials naturally occurring other than uranium and thorium; these other materials are regulated by either the EPA or individual States. Tailings resulting from industrial extraction of metals and minerals of value (such as molybdenum or vanadium) are not routinely considered to be radioactive waste. However, in some cases where tailings have elevated levels of natural radionuclides, the processor may be licensed by NRC. NRC also licenses persons who process ores to extract source material (uranium, thorium).

NRC also has no authority to regulate machine-produced radiation, such as the emissions from X-ray units or linear accelerators; these devices are regulated by either the Food and Drug Administration or individual States.

For the scope of authority over DOE nuclear installations, see Section 8.2 below.

8.1.2.2 International Responsibilities and Activities

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

Several NRC international activities are mandated by U.S. law or international treaties and conventions; other activities are discretionary. NRC's statutorily mandated international activity is as the U.S. licensing authority for exports and imports of nuclear materials and equipment.

NRC supports U.S. foreign policies in the safe and secure use of nuclear materials and in guarding against the spread of nuclear weapons. NRC has actively participated in developing and implementing a variety of legally binding treaties and conventions which create an international framework for the peaceful uses of nuclear energy. NRC provides technical and legal advice and assistance to international organizations and foreign countries to assist in developing effective regulatory organizations and rigorous safety standards. Some activities are carried out within the programs of the IAEA, the Nuclear Energy Agency, or other international organizations. Others are conducted directly with counterpart agencies in other countries pursuant to regulatory and research cooperation agreements.

Export-Import

NRC's key international responsibility is licensing the export and import of nuclear materials and equipment, such as low-enriched uranium fuel for nuclear power plants, high-enriched uranium for research and test reactors, nuclear reactors themselves, certain nuclear reactor components (such as pumps and valves), and radioisotopes used in industrial, medical, agricultural and scientific fields. NRC's principal role is to ensure that such exports and imports are consistent with the goal of the safe and peaceful use of these materials and equipment, limiting the proliferation of nuclear weapons, and promoting the Nation's common defense and security. Standards and procedures for issuing export and import licenses are detailed in the Atomic Energy Act of 1954, as amended, in the Nuclear Non-Proliferation Act of 1978 and in Title 10, Part 110, of the *Code of Federal Regulations*.

As a result of the terrorist attacks in the United States on September 11, 2001, the NRC has undertaken a comprehensive review of nuclear and radioactive material security requirements, with particular focus on high-risk radioactive material. This material has the potential to be used in a radiological dispersal device or a radiological exposure device in the absence of proper security measures. This review took into consideration the changing domestic and international threat environments and related U.S. Government-supported international initiatives in the nuclear security area, particularly activities conducted by the IAEA.

The NRC has also supported U.S. Government efforts to establish common international guidelines governing the export and import of high-risk radioactive materials. This effort has resulted in a major revision to the IAEA Code of Conduct on the Safety and Security of Radioactive Sources (Code of Conduct). The revised Code of Conduct was approved by the IAEA Board of Governors in September 2003. The U.S. Government has formally notified the Director General of the IAEA of its support for the Code of Conduct. Although the Code does not have the stature of an international treaty, and its provisions are non-binding on IAEA Member States, the NRC nevertheless believes it is essential for NRC to update its export/import regulations to incorporate the Code of Conduct recommendations consistent with our responsibilities under the Atomic Energy Act and the NRC's mission of ensuring the common defense and security.

Additionally, the NRC has played a key role in multilateral meetings to develop a document related to the Code of Conduct which provides internationally accepted guidance for export and import activities involving high-risk radioactive material. This document was approved by IAEA Member States in July 2004, and it will be published as an IAEA Information Circular.

The Code of Conduct recommends that IAEA Member States develop specified export/import controls covering sources in Categories 1 and 2 in Table 1 of Annex 1 of the Code. The NRC proposes to require specific licenses for the export and import of high-risk radioactive material. This proposed rule follows the guidance contained in the IAEA's Code of Conduct and is consistent with the recommendations in the Code's section on "Import and Exports of Radioactive Sources" (paragraphs 23-29). A basic principle of the Code of Conduct is that international movements of high-risk radioactive material should take place with the prior notification of the exporting and importing countries. Additionally, international movements of Category 1 quantities of such material require the consent of the importing country. While prior notification by the exporter or importer is required for each export or import shipment, consents must be government-to-government. The Code of Conduct contemplates that, other than in exceptional circumstances, a receiving country should not permit the import of high-risk radioactive material unless it has the technical and administrative capability, resources and regulatory structure needed to ensure that the radioactive material will be managed in a manner consistent with the provisions of the Code. The proposed rule requirements would apply to all identified licensees, both NRC and Agreement State.

Table 1 of the Code of Conduct includes a list of high-risk radionuclides with activities corresponding to thresholds of concern that is essentially identical to the list found in the proposed Appendix P to be added to 10 CFR Part 110. While the radionuclides and threshold quantities are the same, the proposed Part 110 appendix uses the more encompassing term "radioactive material" rather than "sources." Therefore, unlike the Code of Conduct, the proposed rule encompasses the import and export shipments of bulk radioactive material, in addition to sealed sources.

International Treaties

Treaties that legally bind NRC and the U.S. Government's peaceful uses of nuclear energy include the 1978 Nuclear Non-Proliferation Treaty, the 1980 Convention on Physical Protection of Nuclear Material, the1994 Convention on Nuclear Safety, the 1986 Convention on Early Notification of a Nuclear Accident, the 1986 Convention on Assistance in Case of a Nuclear Accident or Radiological Emergency, and the 1997 Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management.

International Organizations and Associations

NRC actively participates in the full scope of programs of the two major international nuclear organizations, the IAEA, and the Nuclear Energy Agency of the Organization for Economic Cooperation and Development. Some examples include the following. Since 1996, the U.S. has or is planning to participate in 27 Operational Safety Assessment Review Team (OSART) missions (with 10 experts from NRC, the rest from industry). Since 1999, NRC has participated in 11 International Regulatory Review Team (IRRT) missions. IAEA is planning an OSART mission to Brunswick Nuclear Power Plant and NRC intends to coordinate with IAEA and industry to plan for OSART missions in the United States on three year cycles. NRC also participates in the Commission on Safety Standards and safety standards committees. Other IAEA activities include the extra-budgetary program for long term operation.

The NRC participates in the NEA Steering Committee as well as the Committee on the Safety of Nuclear Installations (CSNI) and the Committee on Nuclear Regulatory Activities (CNRA). NRC is also represented on many of the CNRA- and CSNI-chartered working groups.

In addition, the Chairman of NRC routinely participates in the IAEA General Conference and biannual meetings of the International Nuclear Regulators Association.

Bilateral Relations

NRC has close working relations with nuclear safety agencies in more than thirty-five countries. NRC and its foreign counterparts exchange operational safety data and other regulatory information. Subject to outside funding, NRC provides safety and safeguards advice, training, and other assistance to countries that seek U.S. help to improve their regulatory programs. At its inception in 1975, NRC primarily cooperated with major nuclear power countries. In the 1980s, responding to an increase in requests, NRC expanded its efforts to assist countries with small or incipient nuclear power programs.

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

In a typical year, the staff conducts more than 40 bilateral meetings with up to 20 countries.

Research Programs

NRC conducts confirmatory regulatory research in partnership with nuclear safety agencies and institutes in more than 20 countries. This research supports regulatory decisions on emerging technologies, aging equipment and facilities, and various other safety issues. NRC and other nuclear regulatory and safety organizations carry out cooperative research projects to achieve mutual research needs with greater efficiency. This research is currently conducted under the following programs:

- CSARP: The Cooperative Severe Accident Research Program. This cooperation facilitates joint information exchange, periodic analysis, and data for additional code validation and verification.

- CAMP: NRC and international partners have formed the Code Applications and Maintenance Program to exchange information on thermal-hydraulic safety issues related to reactor and plant systems.

- COOPRA: The Cooperative Probabilistic Risk Assessment Program facilitates information exchange in the areas of reliability and component failure data, probabilistic risk assessment (PRA) methods, and analysis.

- SGTIP: The cooperative Steam Generator Tube Integrity Program provides data and analysis for predicting the ability of degraded steam generator tubes to withstand normal operating and accident conditions.

8.1.2.3 NRC as an Independent Regulatory Agency

The Commission's status as an independent regulatory agency within the Executive Branch of the Federal Government means that its regulatory decisions cannot ordinarily be directed by the President. (By law, however, the U.S. Office of Management and Budget reviews the proposed NRC budget.) Likewise, the Congress cannot override the Commission's decisions, except by duly enacted legislation. Executive Order 12866, "Regulatory Planning and Review," which provides for review of agency regulations by the U.S. Office of Management and Budget, exempts independent regulatory agencies from its coverage. The subject of NRC's relationship to other governmental bodies is discussed in greater detail below.

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

8.1.3 Structure of the Regulatory Body

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

8.1.3.1 The Commission

The Energy Reorganization Act of 1974 requires that the five Commissioners be U.S. citizens, and that no more than three belong to the same political party. Commissioners serve for fixed five-year terms, and are removable only for cause. The Chairman is designated by the President, and if the Chairman is relieved of that position by the President, he or she remains a Commissioner for the duration of their term. The statute further provides that each member of the Commission shall have full access to information that is necessary to fulfill the member's duties and "equal responsibility and authority in all decisions and actions of the Commission."

However, the Chairman has responsibilities, as spokesman and Chief Executive Officer of the Commission, that significantly differentiate the Chairman's role from the roles of the other Commissioners. The Chairman initiates the appointment, subject to the approval of the Commission, of the Executive Director for Operations, the General Counsel, the Secretary, the members of the Commission's adjudicatory panel, the Chief Financial Officer, and (in consultation with the Executive Director for Operations) the Directors of the Commission's major program offices and the Regional Administrators. The Chairman or a Commissioner can initiate removal of a holder of one of these offices, subject to the approval of the Commission.

To transact business, a quorum of Commissioners (i.e., a minimum of three) is required to be "present." Decision is by a majority of those participating Commissioners may vote, abstain (in which case they are regarded as participating), or decline to participate. To approve final rules and issue adjudicatory decisions, the Commissioners must meet in person; other types of business are handled by written vote sheets.

The results of Commission deliberations are set forth in rules, orders, and "staff requirements memoranda," which record how Commissioners voted, describe the outcome, and typically direct some unit of the staff to take a particular action. Most Commission actions are taken in response to NRC staff papers ("SECY papers"), which are generally made public after the Commission has made its decision.

In adjudicatory proceedings, the Commission is the ultimate decisionmaking body. For these purposes, the Commissioners occupy the role of judges, with all of the constraints that implies on their freedom to discuss or gather information about a pending case. Like other judges, they must decide cases solely on the basis of the record before them, setting aside any knowledge of the issues gained through extra-judicial means, such as the media.

8.1.3.2 Component Offices of the Commission

Since the previous *U.S. National Report* was written, NRC has reorganized to establish a new Office of Nuclear Security and Incident Response.

Office of the Executive Director for Operations

The position of the Executive Director for Operations (EDO) is established by statute. As the head of NRC staff, the EDO reports to the Chairman and is subject to the Chairman's supervision and direction. The EDO is the chief operational and administrative officer of the Commission, and is authorized and directed to discharge such licensing, regulatory, and administrative functions of NRC and to take such actions as necessary for day-to-day operations of the agency. The Executive Director supervises and coordinates policy development and operational activities of NRC's staff, program and regional offices. The Executive Director implements Commission policy directives pertaining to these offices.

There are four Deputy Executive Directors that report directly to the EDO. The four Deputy Executive Directors are: the Deputy Executive Director for Homeland Protection and Preparedness, the Deputy Executive Director for Reactor Programs, the Deputy Executive Director for Materials, Research and State Programs and the Deputy Executive Director for Management Services.

Office of the Chief Financial Officer

The Office of the Chief Financial Officer is responsible for NRC's Planning and Budgeting, and Performance Management Process, and for all of NRC's financial management activities. This office establishes planning, budgeting, and financial management policy for the agency and provides advice to the Chairman and the Commission on these matters. This office develops and maintains an integrated agency accounting and financial management system; establishes policy and directs oversight of agency financial management personnel, activities, and operations; and prepares and transmits an annual report which includes the agency's audited financial statement to the Chairman and the Director of the Office of Management and Budget. This office also monitors the financial execution of NRC's budget in relation to actual expenditures; controls the use of agency funds to ensure that they are expended in accordance with applicable laws and standards; prepares and submits to the Chairman timely cost and performance reports; and reviews, on a periodic basis, fees and other charges imposed by NRC for services provided and makes recommendations for revising those charges as appropriate. In addition, the Office of the Chief Financial Officer provides an agencywide management control program for financial and program managers to comply with the Federal Managers' Financial Integrity Act of 1982, and is responsible for implementing the Chief Financial Officers Act and the Government Performance and Results Act at NRC.

Office of the General Counsel

The Office of the General Counsel provides legal advice, opinions, and assistance to NRC's officials on all of NRC's activities. It advises and assists NRC's regional offices, although each region also has its own Regional Counsel, who is appointed by and reports to the Regional Administrator. The Office of the General Counsel interprets laws, regulations, and other sources of authority, and it advises on the legal form and content of proposed official actions. It represents the staff in licensing and enforcement proceedings; provides legal advice on enforcement actions; prepares, reviews, and interprets all of NRC's contractual documents, interagency agreements, delegations of authority, regulations, orders, licenses, and other legal documents; reviews and gives opinions on issues of intellectual property, including patent, trademark, copyright and proprietary matters; and gives NRC legal opinions and advice about the administration of the Freedom of Information, Privacy, and Government in the Sunshine Acts. It also advises the Commission on legislative matters and represents NRC in the Federal Courts.

Office of the Chief Information Officer

The Office of the Chief Information Officer plans, directs, and oversees the delivery of centralized information technology infrastructure, applications, and information management services, and the development and implementation of information technology and information management plans, architecture, and policies to support the mission, goals, and priorities of the agency. This Office advances the achievement of NRC's mission by assisting management in recognizing where information technology can add value while transforming or supporting agency operations. This Office provides advice and oversight to ensure the NRC complies with Best Practices and applicable Federal Laws and regulation. These laws and regulations include the Clinger-Cohen Act, the Government Paperwork Reduction Act, and the Federal Information Security Act.

Office of the Inspector General

NRC's Inspector General reports to the Congress and NRC Chairman; he or she is not subject to supervision by any other member of NRC. The Inspector General provides leadership and policy direction in conducting audits and investigations to promote economy, efficiency, and effectiveness within NRC, and to prevent and detect fraud, waste, abuse and mismanagement in agency programs and operations. The Inspector General recommends corrective actions to be taken, reports on progress made in implementing those actions, and reports criminal matters to the Department of Justice. The Inspector General analyzes and comments on the impact of existing and proposed legislation and regulations on the economy and efficiency of NRC programs and operations, and the prevention and detection of fraud, waste, abuse and mismanagement.

Office of Nuclear Reactor Regulation

The Office of Nuclear Reactor Regulation is responsible for the licensing, inspection, and regulation of nuclear power and other reactors. Its reviews encompass the safety, environmental, and antitrust aspects of these facilities, and it guides regional offices on facility inspections and licensing activities. Coordinating with NRC's four regional offices, the staff of this office help to ensure appropriate responses to degraded conditions or licensee performance that may adversely affect public health and safety, and the environment. The Office's responsibilities also include the technical review, certification, and licensing of advanced nuclear reactor facilities and the renewal of current power reactor operating licenses. This office is also responsible for licensing issues and regulatory policy concerning reactor operators, including the initial licensing examination and requalification examinations.

Office of Nuclear Material Safety and Safeguards

The Office of Nuclear Material Safety and Safeguards ensures the public health and safety through licensing, inspection, and environmental reviews for activities regulated by NRC, except operating power and all non-power reactors and the safeguards technical review of all licensing activities. The office develops and implements NRC policy regulating activities involving the use and handling of radioactive materials, such as uranium milling activities; fuel fabrication and development; medical, industrial, academic, and commercial uses of radioactive materials; safeguards activities; transportation of nuclear materials, including certification of transport containers, and reactor spent fuel storage; safe management and disposal of low-level and high-level radioactive waste; and management of related decommissioning. In addition, this office coordinates with the IAEA and provides technical support to improve international safeguards on nuclear materials.

Office of Nuclear Security and Incident Response

NRC established this office in 2002. The office develops overall agency policy and provides management direction for evaluation and assessment of technical issues involving security at nuclear facilities, and is the agency safeguards and security interface with the Department of Homeland Security, the intelligence and law enforcement communities, DOE, and other agencies. It also develops and directs NRC programs for response to incidents, and is the agency incident response interface with the Department of Homeland Security, Federal Emergency Management Agency (FEMA) and other Federal agencies.

Under this office, the Emergency Preparedness Directorate (EPD) is responsible for developing emergency preparedness policies, regulations, programs, and guidelines for both currently licensed nuclear reactors and potential new nuclear reactors. EPD also provides technical expertise regarding emergency preparedness issues and interpretations and coordinates with the Office of Nuclear Materials Safety and Safeguards, the Office of Nuclear Reactor Regulation, and other NRC organizations on emergency preparedness matters. Additionally, EPD coordinates and manages emergency preparedness communications with internal and external stakeholders including the public, industry, the international nuclear community, and federal, state, and local government agencies. EPD also provides oversight and technical direction for the emergency preparedness cornerstone of the reactor oversight process.

Office of Nuclear Regulatory Research

The Office of Nuclear Regulatory Research plans, recommends, and conducts research programs to identify, lead, and sponsor reviews that support the resolution of ongoing and future safety issues. These reviews include providing independent information and expertise needed to support NRC's decisionmaking process by improving the agency's knowledge in areas in which uncertainty exists, safety margins are not well characterized, or regulatory decisions are needed. The reviews also anticipate, identify, and characterize technical challenges posed by the introduction of new technologies, emerging regulatory issues, and the evolving state of domestic and international knowledge and operating experience that may become important safety issues in the future. In addition, this office is responsible for regulatory research into reactor safety, safeguards, waste management, radiological protection, and environmental protection.

Regional Offices

NRC's four regional offices are located in the Philadelphia (Region I), Atlanta (Region II), Chicago (Region III), and Dallas (Region IV) areas. About 31 percent of NRC's personnel are stationed in the regions. Each regional office is headed by a Regional Administrator. This administrator is responsible for executing established NRC policies and programs on inspection, enforcement, licensing, State agreements reviews, State liaison, and emergency response within the Region's boundaries.

The responsibilities of NRC's regional offices include inspection and evaluation of engineering, construction, and operational activities of power reactors; operator licensing functions for power reactors; implementation of nuclear material safety licensing and inspection, emergency preparedness, and safeguards licensing functions assigned to the region; coordination of NRC's Incident Response Program for activities within the region; issuance of notices of violation and proposed civil penalties (subject to further approval of Headquarters staff, depending on severity); review of Agreement State regulatory programs; and provision of technical assistance to Agreement States in carrying out their regulatory programs.

The core of the NRC inspection program for nuclear power plants is carried out by resident (on-site) inspectors; at least two inspectors are assigned to each operating site. Resident inspectors monitor licensee activities in accordance with the baseline inspection program.

Office of Enforcement

The Office of Enforcement oversees, manages, and directs the development and implementation of policies and programs for enforcing NRC requirements. The office also develops NRC policy regarding the management of allegations from sources external to NRC. It oversees the agency's allegations management programs and the allegations review process. The office is responsible for external safety culture policy matters and is responsible for the Agency's Differing Professional Opinions Program. It coordinates with the Office of Investigations on issues involving discrimination and wrongdoing associated with allegations from sources external to NRC. In addition, this office approves orders and civil penalties, and reviews other enforcement actions for consistency with the Commission's enforcement policy.

Office of Investigations

This office has the authority to initiate investigations. This office develops policy, procedures, and quality control standards for investigations of licensees, applicants, their contractors or vendors, including the investigations of all allegations of wrongdoing by other than NRC employees and contractors. It refers substantiated criminal cases to the Department of Justice. It also coordinates liaison with other agencies and organizations to ensure the timely exchange of information of mutual interest. This office does not investigate allegations against NRC employees and contractors; that is done by the Office of the Inspector General.

Office of State and Tribal Programs

The Office of State and Tribal Programs is responsible for establishing and maintaining effective communications and working relationships between NRC and other governmental entities, including the States, local governments, Native American tribal governments, and other Federal agencies. This office administers the Agreement State Program, through which NRC provides technical assistance and reviews the adequacy and compatibility of the Agreement States' radiation control programs. (Agreement States now have responsibility for three-fourths of all materials licenses in this country).

Office of International Programs

In consultation with the Commission, the Office of International Programs formulates and implements policies and programs on licensing of nuclear exports and imports, non-proliferation, and bilateral and multilateral nuclear safety international information exchanges. The office also maintains relationships related to international activities with other Federal agencies. In addition, the Office staff assists technical program office staff in implementing IAEA safeguards in the United States and interacting with the IAEA in developing safeguards policies, international physical security of facilities and materials in situ and in transit, and international cooperation and assistance in nuclear safety and radiation protection.

Director of Communications

Established in August 2003 to improve external and internal communications, this position reports directly to the Chairman. The Director is responsible for enhancing the effectiveness of NRC''s communications with the public, the media, and the Congress. The Director directs and supervises the activities and functions of the Office of Public Affairs and the Office of Congressional Affairs, coordinates communication matters with the Office of the Executive Director for Operations and with Commission and staff offices, and provides policy, guidance, and oversight for communication activities across the agency. This includes defining roles and responsibilities and developing processes for overseeing and coordinating interactions with the media and the Congress and communication with stakeholders and the general public.

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

Office of Congressional Affairs The Office of Congressional Affairs is the primary contact point for all NRC communications with Congress. This office monitors legislative proposals, bills, and hearings, and informs NRC of the views of Congress on NRC policies, plans, and activities.

The Office of Commission Appellate Adjudication

The Office of Commission Appellate Adjudication provides the Commission with an analysis of any adjudicatory matter requiring a Commission decision (e.g., petitions for review of initial licensing board decisions, certified questions, interlocutory referrals, stay requests), including available options. This office also drafts necessary decisions pursuant to the Commission's guidance after presentation of options. When necessary, this office consults with the Office of the General Counsel to identify options to be presented to the Commission and to draft the final decision to be presented to the Commission.

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 advises and assists the Commission and staff on the planning, scheduling, and conduct of Commission business; maintains historical paper files of official Commission records; and administers NRC's Historical Program. The Secretariat maintains the Commission's official adjudicatory and rulemaking dockets, including management of the Commission's Electronic Hearing Docket.

Support Offices

Supporting the EDO are the Offices of Administration, Human Resources, and Small Business and Civil Rights.

8.1.3.3 Advisory Committees

The Commission has four advisory committees chartered under the Federal Advisory Committee Act and composed of subject matter experts from outside NRC. This statute imposes certain constraints on advisory committees, primarily that they give advance notice of their meetings and, unless certain exemptions apply, hold them open to the public. The Commission also has licensing panels.

- Advisory Committee on Reactor Safeguards: This Committee has statutory responsibilities as described in the Atomic Energy Act of 1954, as amended. The Committee reviews and advises the Commission with regard to the licensing and operation of production and utilization facilities and related safety issues, the adequacy of proposed reactor safety standards, technical and policy issues related to the licensing of evolutionary and passive plant designs, and other matters referred to it by the Commission. Upon request, the Committee reviews and advises with regard to the hazards of the DOE nuclear activities and facilities and provides technical advice to the DOE Nuclear Facilities Safety Board. On its own initiative, the Committee may conduct reviews of specific safety-related items. It submits an annual report to the Commission commenting on the NRC Safety Research Program.

- Advisory Committee on Nuclear Waste: This Committee reports to and advises the Commission on all aspects of nuclear waste management. The Committee will undertake studies and activities related to the transportation, storage, and disposal of radioactive waste, including the interim storage of spent nuclear fuel; materials safety; decommissioning; application of risk-informed, performance-based regulations; control of the disposition of radioactive materials; and evaluation of licensing documents, rules, regulatory guidance, and other issues as requested by the Commission.

- Advisory Committee on the Medical Uses of Isotopes: This Committee advises NRC on policy and technical issues that arise in the regulation of the medical uses of radioactive material in diagnosis and therapy. Members include health care professionals from various disciplines who comment on changes to NRC regulations and guidance; evaluate certain non-routine uses of radioactive material; provide technical assistance in licensing, inspection, and enforcement cases; and bring key issues to the attention of the Commission for appropriate action.

- The Licensing Support Network Advisory Panel: Established in 1989, this panel's purpose is to advise on the design and development of an electronic information management system to store and retrieve documents on the licensing of a geologic repository for disposing of high-level radioactive waste, and on the operation and maintenance of the system. Membership on the Panel is drawn from those interests that will be affected by the use of the network, including the DOE, NRC, the State of Nevada, the National Congress of American Indians, affected units of local governments in Nevada, the Nevada Nuclear Waste Task Force, and a coalition of nuclear industry groups. Federal agencies with expertise and experience in electronic information management systems may also participate on the Panel.

8.1.3.4 Significant Changes in NRC Organization Since the 2001 U.S. National Report

NRC, recognizing, that it is a safety, security, and preparedness agency, created a new Deputy Executive Director for Homeland Protection and Preparedness. This Director oversees activities of the newly established Office of Nuclear Security and Incident Response and is responsible for ensuring the effectiveness of NRC's security and emergency preparedness programs. The integration of safety, security, and preparedness has become increasingly more important since September 11, 2001. To ensure that security and safety programs do not become isolated from each other, but remain closely coupled, the Office of Nuclear Reactor Regulation, Office of Nuclear Materials Safety and Safeguards, and Office of Nuclear Security and Incident Response work closely together.

Reinforcing its commitment to improve external communications, NRC created a new position, a Director of Communications who reports directly to the Chairman. This Director oversees the Offices of Public Affairs and Congressional Affairs and enhances the effectiveness of NRC's communications with the public, the media, and the Congress to support the agency's strategic goals.

To increase communication of its emergency preparedness activities with the public; industry; the international nuclear community; and Federal, State, and local Government agencies, the agency has formed an Emergency Preparedness Project Office, which is located in the Office of Nuclear Security and Incident Response. This office develops emergency preparedness policies, regulations, programs, and guidelines for both currently licensed nuclear reactors and potential new nuclear reactors; provides technical expertise on emergency preparedness issues and coordinates with other parts of NRC and stakeholders; and oversees and provides technical direction for the emergency preparedness cornerstone of the Reactor Oversight Process.

8.1.4 Financial and Human Resources

This section discusses the budget and funding of NRC, its human resources, and financial management.

For fiscal years 1991–2000, NRC was required by law to recover about 100 percent of its budget by assessing fees. However, NRC raised concerns regarding the fairness and equity of charging licensees for budgeted agency costs that do not directly benefit licensees. As a result, legislation was amended in 2001 to decrease the amount of NRC's budget recovered by fees by 2 percent per year, beginning in FY 2001, until 90 percent is recovered in FY 2005. For FY 2004, Congress appropriated $626.1 million to NRC, of which NRC is required to recover about $545.6 million in fees.

NRC assesses two types of fees. One type is license and inspection fees, which recover NRC's costs of providing special benefits to particular/known applicants and licensees. Examples of special benefits are the review of applications for new licenses and for license renewal, amendment requests, and inspections. The second type of fee is an annual fee to recover generic and other regulatory costs not otherwise recovered through license and inspection fees.

Tables 8.1 and 8. 2 below show the financial and human resources to support NRC's programs.

Table 8. 1: Budget Authority by Appropriation

NRC Appropriation	**FY 2003 Enacted**	**FY 2004 * Enacted**	**FY 2005 Full Cost** Request**
Salaries and Expenses (S&E) ($K)			
Budget Authority	577,806	618,800	662,777
Offsetting Fees	519,884	538,844	534,355
Net Appropriated—S&E	57,922	79,956	128,422
Office of the Inspector General (OIG) ($K)			
Budget Authority	6,797	7,300	7,518
Offsetting Fees	6,389	6,716	6,766
Net Appropriated—OIG	408	584	752
Total NRC ($K)			
Budget Authority	**584,603**	**626,100**	670,295
Offsetting Fees	526,273	545,560	541,121
Total Net Appropriated	58,330	80,540	129,174
Net Appropriated			
Nuclear Waste Fund	24,738	33,100	69,050
General Fund (Percent Off Fee Base)	33,592	47,440	60,124
Total Net Appropriated	58,330	80,540	129,174

* Does not include recision estimated at $.5 million from the Consolidated Appropriations Act, 2004 (Public Law 108-199)
** In accordance with the requirement defined in Section 220 (b) of Office of Management and Budget (OMB) Circular A-11, NRC is providing the full cost associated with its programs for the FY 2005 budget request. Full cost includes an allocation of the agency's infrastructure and support costs to each of NRC's programs. The allocation methodology is consistent with the methodology used for preparing the agency's financial statements.

Table 8.2: Budget Authority and Staffing by Strategic Arena

Summary	FY 2003 Enacted	FY 2004 Enacted *	FY 2004 Full Cost Estimate	FY 2005 Full Cost** Request
Budget Authority by Strategic Arena ($K)				
Nuclear Reactor Safety	276,395	306,982	**423,486**	**435,149**
Nuclear Materials Safety	59,979	65,803	**95,824**	**100,337**
Nuclear Waste Safety	70,416	72,279	**90,809**	**118,096**
International Nuclear Safety Support	5,237	5,856	**8,681**	**9,195**
Management and Support	165,779	167,880	**0**	**0**
Subtotal	**577,806**	**618,800**	**618,800**	**662,777**
Inspector General	6,797	7,300	**7,300**	**7,518**
Total NRC	**584,603**	**626,100**	**626,100**	**670,295**
Staffing Full-Time Equivalents (FTE) by Strategic Arena				
Nuclear Reactor Safety	1,573	1,662	**2,086**	**2,102**
Nuclear Materials Safety	380	406	**516**	**518**
Nuclear Waste Safety	270	271	**338**	**375**
International Nuclear Safety Support	38	43	**53**	**53**
Management and Support	601	611	**0**	**0**
Subtotal	**2,862**	**2,993**	**2,993**	**3,048**
Inspector General	44	47	**47**	**47**
Total NRC	**2,906**	**3,040**	**3,040**	**3,095**
Reimbursable Business-Like FTE	13	19	**19**	**14**
Total NRC	**2,919**	**3,059**	**3,059**	**3,109**

*** Does not include recision of approximately $.5 million from the Consolidated Appropriations Act, 2004 (Public Law 108-199)**

**** In accordance with the requirement defined in Section 220 (b) of Office of Management and Budget (OMB) Circular A-11, NRC is providing the full cost associated with its programs for the FY 2005 budget request. Full cost includes an allocation of the agency's infrastructure and support costs to each of NRC's programs. The allocation methodology is consistent with the methodology used for preparing the agency's financial statements.**

8.1.5 Position of NRC in the Governmental Structure

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

8.1.5.1 Executive Branch

The components of the Executive Branch with which NRC has the most frequent contacts and interactions are the White House, Office of Management and Budget, Department of State, DOE, EPA, FEMA, Department of Homeland Security, Department of Labor, Department of Transportation, and Department of Justice. NRC's relationship to the DOE is discussed in Section 8.2.

The White House

As noted above, NRC's status as an independent regulatory agency means that the White House cannot directly set NRC policy. It can, however, influence NRC policy by (1) appointing Commissioners and Chairmen in whose outlook and judgment it has confidence, and (2) making its views known on nonadjudicatory matters. In certain areas, such as national security policy, the Commission has declared its intent to give great weight to the views of the Executive Branch. In informal policy matters, such as rulemaking, White House and Executive Branch officials may properly try to influence NRC decisions, either publicly or privately; ultimately, however, NRC must make the decision and accept responsibility for it.

NRC also works with the National Security Council and the White House to help the United States develop policies for cooperating with and assisting the former Soviet Union, and the agency works with the Department of Homeland Security and the White House in defending against terrorism.

Office of Management and Budget

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

U.S. Department of State

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

NRC also works with the U.S. Department of State in such matters as negotiating international agreements in the nuclear field and interacting with the IAEA and other international organizations of the United Nations, as well as the Nuclear Energy Agency of the Organization for Economic Cooperation and Development. In general, the purposes of these interactions are to develop policy on nuclear issues that are under NRC's purview; and to plan and coordinate programs of nuclear safety and safeguards assistance to other countries, such as the former Soviet Union and Central and Eastern Europe.

EPA

The responsibilities of NRC and EPA intersect or overlap in a number of areas in which the EPA issues generally applicable environmental standards for activities that are also subject to NRC licensing. Examples include standards for high-level waste repositories, decommissioning standards, and standards for public and worker protection. Under Reorganization Plan No. 3 of 1970, EPA has the ultimate authority to establish generally applicable environmental standards to protect the environment from radioactive material.

FEMA

In 1979, after the accident at Three Mile Island (TMI), the President assigned FEMA the lead responsibility for offsite emergency planning and response at nuclear power plants. NRC remained responsible for evaluating onsite planning, and for making the overall finding regarding whether a plant can operate "without undue risk to public health and safety." A 1980 Memorandum of Understanding between the two agencies, since updated, lays out the relationship between FEMA and NRC, as it relates to emergency planning. Among other things, FEMA assists NRC's licensing process by preparing reviews and evaluations, as well as presenting witnesses to testify at licensing hearings. FEMA also participates with NRC in observing and evaluating emergency exercises at nuclear plants. FEMA's findings are not binding on NRC, but they are presumed to be valid unless controverted by more persuasive evidence. FEMA is now part of the Department of Homeland Security.

U.S. Department of Homeland Security

Since its founding, the NRC has had a statutory obligation to regulate radioactive materials in a way that not only protects public health and safety but also promotes the common defense and security of the United States. In fulfilling its obligation to the common defense and security, the agency regulates security at nuclear facilities and the protection of radioactive materials. The new U.S. Department of Homeland Security was established in 2003 to lead a unified national effort to prevent terrorist attacks, reduce vulnerability to terrorism, and coordinate the Federal government's response to terrorist attacks and natural disasters. The NRC coordinates much of its safeguards and security work with Department of Homeland Security, the intelligence and law enforcement communities, the U.S. Department of Energy, and other agencies.

Department of Labor

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

U.S. Department of Transportation

By virtue of the NRC's broad licensing and regulatory control of the possession, use, and transfer of nuclear materials, the NRC is responsible for ensuring that standards, rules, and regulations provide for adequate protection of the health and safety of the public and maintenance of the national defense and security during the transport of radioactive materials. However, under the Hazardous Material Transportation Act and other statutes, the Department of Transportation also has responsibilities for the safe transportation of radioactive materials. To avoid duplication of effort and to clarify agency roles, the Department of Transportation and NRC have entered into a memorandum of understanding. Generally speaking, the NRC is responsible for setting safety standards for package design and performance, and for considering applications for package design approval. Department of Transportation regulations cover all aspects of transportation -- including packaging, shipper and carrier responsibilities, and documentation.

U.S. Department of Justice

NRC litigation almost always requires coordinating with the 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. Department of Justice attorneys represent the United States. However, the Hobbs Act also provides for independent representation of NRC by the agency's own attorneys. In practice this means that NRC attorneys, under the supervision of the NRC Solicitor and the General Counsel, write the briefs in Hobbs Act cases and argue the cases in the courts of appeals. The Department of Justice joins in NRC's briefs, except in the extremely rare circumstance in which the Department of Justice views an NRC position as inconsistent with general government interests.

In cases other than those involving the Hobbs Act (i.e., those cases [usually in district court]) that do not involve NRC licensing or regulatory action, such as tort, subpoena enforcement, personnel, licensee bankruptcy, or Freedom of Information Act cases), Department of Justice attorneys normally take the lead role, with backup support from NRC attorneys.

NRC's investigatory arms frequently work with the Department of Justice staff. NRC has authority to revoke or suspend licenses, impose civil penalties, and take other civil actions for willful wrongdoing.

The Office of Investigations, which investigates allegations of wrongdoing by NRC's applicants and licensees, as well as by their contractors, normally deals with the General Litigation Section of the Criminal Division at the Department's headquarters and with U.S. Attorneys in the field.

Pursuant to the Inspector General Act, the Office of the Inspector General reports 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.

8.1.5.2 The States

At NRC, the Office of State and Tribal Programs is responsible for establishing and maintaining effective communications and working relationships between the NRC and the States and serves as the primary contact for policy matters, keeping the States informed on NRC activities and keeping the NRC appraised of State activities and views as they may affect NRC policies, plans, and activities. While program direction is primarily provided through the Office of State and Tribal Programs, other NRC Offices and the NRC Regional Offices provide major support to implement State relations program policy and guidance, for example, through Regional State Liaison and State Agreements Officers.

As explained above, the Atomic Energy Act of 1954 confers on NRC preemptive authority over health and safety regulation of nuclear energy and Atomic Energy Act materials. As a result, the general rule is that nuclear power plant safety, like airline safety, is the exclusive province of the Federal Government and cannot be regulated by the States. A State law that attempted to set nuclear safety standards would thus be voided by the courts. However, the courts will not overturn a State law that regulates nuclear energy for purposes, such as economics, that are other than health and safety, unless it conflicts with an NRC requirement. Similarly, the courts will not ordinarily question a State's declared purpose in enacting legislation.

However, the Atomic Energy Act of 1954, as amended, did not entirely exclude States from the regulation of nuclear matters. Section 274 of the Act created the Agreement State Program, under which NRC may relinquish its authority over most nuclear materials, and States may assume that authority. NRC may not relinquish authority over such facilities as reactors, fuel reprocessing and enrichment plants, imports and exports, critical mass quantities of special nuclear material, high-level-waste disposal, or certain other excepted areas.

As of February 2004, thirty-three States have signed formal agreements with NRC, by which those States have assumed regulatory responsibility over certain byproduct, source, and small quantities of special nuclear material. Agreement States receive no Federal funding to support their regulatory programs. NRC conducts performance-based reviews of Agreement State programs to ensure that they remain adequate to protect the public health and safety and compatible with NRC's materials program.

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

The Nuclear Waste Policy Act of 1982 also affords a major role to affected States and Native American tribes in decisions on siting high-level waste repositories and monitored retrievable storage facilities.

Through the Federal, State and Tribal Liaison Program, NRC works in cooperation with Federal, State, and local governments, interstate organizations and Native American Tribal Governments to ensure that NRC maintains effective relations and communications with these organizations and promotes 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.

8.1.5.3 Congress

This section explains the relationship of NRC to the U.S. Congress. Components of Congress discussed are NRC's oversight committees in the Senate and House and subcommittees with jurisdiction over aspects of NRC's activities.

Senate Oversight

Unlike most Federal agencies, the relations of NRC with its oversight committees are (in part) governed by statute. NRC is required to keep these committees "fully and currently informed" of matters in NRC's jurisdiction.

In the Senate, the Committee on Environment and Public Works exercises jurisdiction over domestic nuclear regulatory activities. Within the Committee, the Subcommittee on Clean Air, Climate Change, and Nuclear Safety has responsibility for legislation and oversight of NRC. It considers nominations of Commissioners and the Inspector General.

The Senate Energy and Natural Resources Committee shares jurisdiction over nuclear waste issues with the Environment and Public Works Committee.

In the area of international conventions, the Senate Foreign Relations Committee has purview in providing the advice and consent to the President for ratification.

House Oversight

In the U.S. House of Representatives, jurisdiction over domestic nuclear regulatory activities resides in the Committee on Energy and Commerce. Within the Committee, the Subcommittee on Energy and Air Quality has responsibility for regulation and oversight of NRC.

Other Relevant Committees

In addition to these subcommittees, 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 Congressional subcommittees on appropriations, international affairs, research, and general government operations have jurisdiction over some aspect of NRC activities.

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

Although the NRC and the DOE have responsibilities relating to the management of nuclear facilities and materials, they maintain separate and independent functions. The partitioning of the U.S. Atomic Energy Commission in the mid 1970's was strongly motivated to provide distinct entities for the U. S. Government's regulatory and promotional responsibilities in nuclear applications.

Specifically, the Energy Reorganization Act of 1974 redistributed the functions performed by the U.S. Atomic Energy Commission to two new agencies. The Energy Reorganization Act of 1974 created the NRC to regulate the commercial nuclear power sector and the Energy Research and Development Administration (ERDA) to promote energy and nuclear power development and to develop defense applications. NRC was established as an independent authority governed by a five-member Commission to regulate the possession and use of nuclear materials as well as the siting, construction, and operation of nuclear facilities. The ERDA was established to ensure the development of all energy sources, increase efficiency and reliability of energy resource use, and carry out the other functions, including but not limited to the U.S. Atomic Energy Commission military and production activities and general basic research activities. Supporters and critics of nuclear power agreed that the promotional and regulatory duties of the U.S. Atomic Energy Commission for commercial activities should be assigned to different agencies.

The NRC performed its regulatory mission by issuing regulations, licensing commercial nuclear reactor construction and operation, licensing the possession of and or use of nuclear materials and wastes, safeguarding nuclear materials and facilities from theft and radiological sabotage, inspecting nuclear facilities, and enforcing regulations. NRC regulates the commercial nuclear fuel cycle materials and facilities. Regarding the regulatory control of commercial spent nuclear fuel and radioactive waste, NRC is responsible for licensing commercial nuclear waste management facilities, independent spent fuel management facilities, and DOE's proposed Yucca Mountain site for the disposal of high-level waste and spent fuel.

The DOE addresses the need by the U.S. Government to unify energy organization and planning. The DOE Organization Act brought a number of the Federal government's agencies and programs, including ERDA, into a single agency with responsibilities for nuclear energy technology and nuclear weapons programs. Over the past decade, the DOE has added new nuclear-related activities directed to environmental clean up of contaminated sites and surplus facilities and nonproliferation. The DOE retains authority under the Atomic Energy Act of 1954 for regulating its nuclear activities. Activities and sites under the DOE's regulatory control include research and weapons-related nuclear facilities, such as Rocky Flats in Colorado, Fernald in Ohio, and the Hanford Reservation in Washington. The DOE also retains

responsibility to regulate the disposal of the its own low-level radioactive waste (other than commercial 'greater than Class C" waste).

A separate statute, the Waste Isolation Pilot Plant Land Withdrawal Act (WIPP LWA) of 1992 provides the EPA authority to periodically certify that WIPP meets EPA generally applicable standards (40 CFR Part 191). The WIPP is a geologic repository licensed to safely and permanently dispose of transuranic radioactive waste left from the research and production of nuclear weapons. NRC does not license the DOE's activities at the WIPP; however, NRC does have a role in certificating WIPP shipping containers. The primary regulatory control remains with the EPA, which enforces its generally applicable radiation standards and provides oversight of the DOE WIPP disposal facility for transuranic radioactive waste.

NRC and the DOE have mutual assistance agreements for responding to radiological emergencies through the Federal Response Plan and the Federal Radiological Emergency Response Plan coordinated by FEMA.

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.

This section explains how NRC ensures that the prime responsibility for the safety of a nuclear installation rests with the licensee through the Atomic Energy Act. Steps the NRC takes to ensure that each licensee meets its primary responsibility include the licensing process, discussed in Articles 18 and 19, the Reactor Oversight Process, discussed in Article 6, and the Enforcement Program, discussed below.

This section was updated to incorporate recent experience and examples.

9.1 Introduction

The basis for NRC's regulatory programs continues to be that the safety of commercial nuclear power reactor operations is the responsibility of NRC licensees. The responsibility of NRC is regulatory oversight of licensee activities to ensure that safety is maintained. NRC comprehensively 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 requirements of the *Code of Federal Regulations*, NRC then issues a license to operate the facility. Such licenses specify the terms and conditions of operation, to which a licensee must conform. Failure to conform would subject the licensee to enforcement action, which can include modifying, suspending, or revoking the license. NRC can also order particular corrective actions or issue civil penalties. These enforcement mechanisms are discussed in greater detail below.

9.2 The Licensee's Prime Responsibility for Safety

As discussed in Article 7 of this report, the Atomic Energy Act of 1954, Chapter 10, Section 103, "Commercial Licenses" grants NRC authority to issue licenses for nuclear reactor facilities. Moreover, Section 103 states that such licenses are subject to such conditions as NRC may by rule or regulation establish to effectuate the purposes and provisions of the Atomic Energy Act. Consistent with the Act, before issuing a license, the Commission determines that the applicant is 1) equipped to observe and agrees to observe such safety standards to protect health and to minimize danger to life or property as the Commission may by rule establish; 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 the health and safety of the public.

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

When the Commission or licensee determines that the licensee is not complying with the Commission's rules or regulations, action is taken to ensure the facility is returned to a condition compliant with its license.

9.3 NRC Enforcement Program

This section explains NRC's enforcement program, and provides experience and examples.

9.3.1 Description of Program

As discussed in Article 7, NRC has enforcement powers. As discussed in Article 6 (section 6.2.2.5), the enforcement process has been integrated into the Reactor Oversight Process. 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.

NRC identifies violations through inspections and investigations. All violations are subject to civil enforcement action and also may be subject to criminal prosecution. Unlike the burden of proof standard for criminal actions (beyond a reasonable doubt), NRC uses the Administrative Procedure Act standard in enforcement proceedings [preponderance of evidence]. After an apparent violation is identified, it is assessed in accordance with the Commission's Enforcement Policy. The policy is published as NUREG-1600, "General Statement of Policy and Procedure for NRC Enforcement Actions," to foster its widespread dissemination to NRC licensees and members of the public. Revisions of the policy are noticed in the *Federal Register*. NRC's Office of Enforcement maintains the current policy statement on NRC's public Web site at http://www.nrc.gov/what-we-do/regulatory/enforcement/enforce-pol.html. Because it is a policy statement and not a regulation, the Commission may deviate from it, as appropriate, under the circumstances of a particular case.

NRC has three primary enforcement sanctions available. Specifically, those sanctions include notices of violation, civil penalties, and orders.[e] A notice of violation identifies a requirement and how it was violated, formalizes a violation pursuant to 10 CFR 2.201, "Notice of Violation," requires corrective action, and normally requires a written response. A civil penalty is a monetary fine issued under authority of Section 234 of the Atomic Energy Act or Section 206 of the Energy Reorganization Act. Section 234 of the Atomic Energy Act provides for penalties of up to $100,000 per violation per day; however, that amount is adjusted every four years by the Debt Collection Improvement Act of 1996, and is currently $120,000. The Commission's order-issuing authority under Section 161 of the Atomic Energy Act is broad, and extends to any area of licensed activity that affects public health and safety or the common defense and security. Orders modify, suspend, or revoke licenses, or require specific actions by licensees or persons. Notices of violations and civil penalties are issued on the basis of violations. Orders may be issued for violations or, in the absence of a violation, because of a public health or safety or common defense and security issue.

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

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

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[e] 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.

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

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

- Associate the violation with findings assessed through the Reactor Oversight Process Significance Determination Process assigned a color code of green, white, yellow, or red based on increasing risk significance. (The Significance Determination Process is described in Article 6.)

The Commission recognizes that there are violations of minor safety or environmental concern that are below Severity Level IV violations, as well as below violations associated with green findings. These minor violations are not assigned a severity level category nor a color assessment.

The way a violation is dispositioned depends on the seriousness of the violation and the circumstances. Of limited risk significance, minor violations are not subject to enforcement action and are not normally described in inspection reports. However, minor violations, like all violations, must be corrected. Severity Level IV violations and violations associated with green findings comprise most of the other violations identified in the nuclear industry. Provided certain criteria in Section VI.A of the Enforcement Policy are met, NRC will normally handle these Severity Level IV violations and violations associated with green findings as non-cited violations. Non-cited violations are documented in inspection reports (or inspection records for some materials licensees) to establish public records of the violations, but are not cited in notices of violation which normally require written responses from licensees. Dispositioning violations this way does not eliminate NRC's emphasis on compliance with requirements nor the importance of maintaining safety. Licensees are still responsible for maintaining safety and compliance and for taking corrective actions. This approach for violations that have low risk significance is consistent with the agency's performance goals. More significant violations are candidates for escalated enforcement. Escalated enforcement action is defined as action involving Severity Level I, II, or III violations; violations associated with white, yellow, or red findings; civil penalties; or orders.

NRC may hold a predecisional enforcement conference or a regulatory conference with a licensee before making an enforcement decision if escalated enforcement action appears warranted, if NRC decides a conference is necessary, or if the licensee requests it.
The purpose of the conference is to obtain information to assist NRC in determining the appropriate enforcement action, such as a common understanding of: facts, root causes, and missed opportunities associated with the apparent violations; corrective actions taken or planned; and the significance of issues and need for lasting, comprehensive corrective actions.

The decision to hold a conference does not mean that the agency has determined that a violation has occurred or that enforcement action will be taken. In accordance with the Enforcement Policy, conferences are normally open to public observation unless individual wrongdoing is involved. If NRC concludes that a conference is not necessary, it may give a licensee an opportunity to respond to the apparent violations in writing before it makes an enforcement decision.

Civil penalties are normally assessed for Severity Level I and II violations, as well as knowing and conscious violations of the reporting requirements of Section 206 of the Energy

Reorganization Act. Civil penalties are considered for Severity Level III violations. Although not normally used for violations associated with the Reactor Oversight Process, civil penalties (and the use of severity levels) are considered for issues that are willful, that have the potential to affect the regulatory process, or that have actual consequences.

NRC imposes different levels of civil penalties based on several factors:

- type of licensed activity
- type of licensee
- severity level of the violation
- whether the licensee has had any previous escalated enforcement action (regardless of the activity area) during the past 2 years or past 2 inspections, whichever is longer
- whether the violation was willful or very significant
- whether the licensee should be given credit for actions related to identification
- whether the licensee's corrective actions are prompt and comprehensive
- whether, in view of all the circumstances, the matter in question requires the exercise of discretion

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

If a civil penalty is proposed, a written notice of violation and proposed imposition of civil penalty is issued and the licensee has 30 days to respond in writing. It can do so by either paying the penalty or contesting it. NRC considers the response, and if the penalty is contested, may either mitigate it or impose it by order. The licensee may then pay the civil penalty or request a hearing.

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 such other action as may be proper. It may issue orders in lieu of, or in addition to civil penalties. Additionally, NRC may issue an order to impose a civil penalty where 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 (30 days). However, orders can be made immediately effective without prior opportunity for a hearing when it is determined the public health, safety, or interest so requires. Subsequent to the hearing process, a licensee or individual may appeal the administrative hearing decision to the Commission and appeal the Commission's decision to the court of appeals.

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

9.3.2 Experience and Examples

During FY 2003 (October 1, 2002, through September 30, 2003), NRC issued a variety of significant enforcement actions to operating power reactors. Specifically, these actions included 19 escalated notices of violation without civil penalties, 2 civil penalties, 1 confirmatory order, and 1 order imposing a civil monetary penalty.

To provide accurate and timely information to all interested stakeholders and enhance the public's understanding of the enforcement program, NRC's Office of Enforcement publishes related information on NRC's public Web site at http://www.nrc.gov/what-we-do/regulatory/enforcement.html. This information includes copies of significant enforcement actions issued to reactor and materials licensees since 1996, and actions issued to individuals for a period of one year from the date of the action or for the duration of the enforcement-related restriction.

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.

This section discusses probabilistic risk assessment (PRA) and safety culture. The applications of PRA discussed are (1) severe accident issues, (2) evaluating new and existing regulatory requirements and programs, (3) the implementation plan for risk-informed regulation, (4) activities that improve data and methods of risk analysis, (5) industry activities and pilot PRA applications, and (6) activities that apply risk assessment to plant-specific changes to the licensing basis. Finally, this section discusses safety culture.

Other articles (for example, Articles 6, 14, 18, and 19) also discuss activities undertaken to achieve nuclear safety at nuclear installations. In particular, see the discussion of the Reactor Oversight Process in Article 6.

This section was updated to discuss new regulations, new developments in PRA, and safety culture.

10.1 Background

The United States has made much progress in developing and using the results of PRAs for all operating reactor facilities, and NRC has developed extensive guidance regarding the role of PRA in regulatory programs in the United States. Specifically, NRC uses insights derived from PRA, together with safety goals, to prioritize resources, establish regulatory coherence, and develop policies that give due priority to nuclear safety. NRC believes that a PRA for a plant can yield important information about plant safety since it analyzes plant safety holistically. NRC has extensively applied information gained from PRA to complement other engineering analyses in improving issue-specific safety regulation, and in changing the current licensing bases for individual plants. The move toward risk-informing the current regulations and processes continues to mark perhaps the most significant changes taking place at NRC, as illustrated by the following recent examples:

(1) 10 CFR 50.44 “Standards for Combustible Gas Control in Light-Water-Cooled Power Reactors,” published in September 2003, eliminated the need for design basis combustible gas controls and realigned the regulatory treatment of oxygen and hydrogen monitoring systems.

(2) 10 CFR 50.69 “Risk-Informed Categorization and Treatment of Structures, Systems, and Components,” was published for comment in May 2003. This new, voluntary rule would modify the scope of the “special treatment” regulations in 10 CFR Part 50. It would do so by creating an alternative regulatory framework that enables licensees to use a risk-informed approach to categorize structures, systems, and components, and their associated design and protection, according to their safety significance.

NRC is also continuing a program to develop additional changes to the specific technical requirements in the body of 10 CFR Part 50, including the general design criteria. This program provides a framework for risk-informing deterministic requirements. NRC has used this framework to evaluate the technical feasibility of risk-informing parts of 10 CFR 50.46

“Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors.” NRC is proceeding with rulemaking, closely cooperating with stakeholders.

10.2 Probabilistic Risk Assessment Policy

Four policy statements form the basis for NRC’s current treatment of PRA and the related regulatory safety goals and objectives. These are the “Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants,” issued August 8, 1985; the “Safety Goals for the Operations of Nuclear Power Plants Policy Statement,” issued August 21, 1986; the “Policy Statement on Use of PRA Methods in Nuclear Activities,“ issued August 16, 1995; and the "Commission Policy on Regulatory Decisionmaking and PRA Quality," issued on December 18, 2003. These policies are described in more detail in the previous *U.S. National Report*.

10.3 Applications of Probabilistic Risk Assessment

NRC applies PRA technology to resolve severe accident issues, evaluate new and existing requirements and programs, implement risk-informed regulation, and improve data and methods of risk analysis. NRC also engages in cooperative activities with industry (such as pilot programs for 10 CFR 50.69 and Regulatory Guide 1.200) and in activities that assess risk in determining plant-specific changes to the licensing basis.

10.3.1 Severe Accident Issues

The main focus of the severe accident policy statement was on the criteria and procedures to be used to certify new designs for nuclear power plants. NRC expected new plants to achieve a higher standard of safety performance in severe accidents than plants of earlier designs. Demonstrating its commitment to NRC’s severe accident policy, the industry established a goal of designing future reactors to a core damage frequency of less than 1×10^{-5} per year of reactor operation. The “Probabilistic Safety Assessment Applications Guide” for advanced light-water reactors, published in August 1995, by the Electric Power Research Institute (EPRI), describes a method to meet this goal.

As for addressing the risk of severe accidents at existing plants, NRC established the “Integration Plan for Closure of Severe Accident Issues” (SECY 88-147), confirming the view that it saw no need for immediate regulatory action. As part of the Integration Plan, the Individual Plant Examination Program examined operating plants for vulnerabilities to severe accidents attributable to internally initiated *events within the plant* during full power operation. By contrast, the Individual Plant Examination of External Events program examined vulnerabilities to severe accidents caused by *external events*, such as earthquakes, fires, and high wind. NRC’s review of these programs for all of the existing plants showed core damage frequencies that are generally in the range of 1×10^{-4} to 1×10^{-6} per reactor year, and revealed that loss-of-offsite power and station blackout are the significant contributors to core damage frequency. As a result of its review, NRC published NUREG/CR-1560, “Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance,” in December 1997. In April 2002, NRC also published NUREG-1742, “Perspectives Gained from the Individual Plant Examination of External Events Program.”

10.3.2 Using the Safety Goals to Evaluate New and Existing Regulatory Requirements and Programs

The aim of the safety goal policy statement (described in Section 10.2) was to establish goals that broadly define an acceptable level of radiological risk that might be imposed on the public as a result of nuclear power plant operation. Radiological risk means the risk associated with the release of radioactive material from the reactor to the environment from normal operations as well as from accidents.

The safety goals, which are qualitative, are as follows:

(1) Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.

(2) Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risk of generating electricity by viable competing technologies, and should not be a significant addition to other societal risks.

To define "risk to life and health," the NRC approved the following quantitative health objectives:

(1) The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.

> (The average individual in the vicinity of the plant is defined as the average individual biologically, in terms of age and other risk factors, who resides within 1 mile from the plant site boundary. This means that the risk to the average individual is found by accumulating the individual risks and dividing them by the number of individuals residing in the vicinity of the plant.)

(2) The risk of fatalities from cancer to the population in the area near a nuclear power plant that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

> The population considered "near" a nuclear power plant is taken as the population within a 10-mile radius of the plant site.

Underlying the safety goals is the premise that the current regulatory practice of requiring compliance with NRC's regulations ensures the basic statutory standard of adequate protection. NRC believes, however, that the current practices can be improved to better test the adequacy and necessity of current requirements, as well as the possible need for additional requirements. NRC sees the safety goals policy as a vehicle for achieving these objectives. As a result, the safety goals have become the bases, in part, for evaluating the need for new and revised regulations and regulatory practices. In fact, NRC has applied this ongoing evaluation and refinement to simplify the implementation of the safety goals themselves. In doing so, NRC recognized that, although the safety goals are quite straightforward, they are somewhat difficult to implement. As a result, NRC established a subsidiary objective of a core damage frequency of 1×10^{-4} per reactor-year. In addition, NRC approved a conditional containment failure probability of 0.1 (one-tenth) for evolutionary light water reactor designs. These values have evolved into the “benchmark” values of 1 x 10^{-4} for core damage frequency and 1 x 10^{-5} for large early release frequency, as discussed in Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” for use in risk-informed regulatory decisionmaking.

Over the past several years, the agency has used these subsidiary objectives in developing new regulations. For example, it developed new regulations on anticipated transients without scram, station blackout, and pressurized thermal shock, in part, using the estimated changes to the collective core damage frequency provided by the rules, and by applying the subsidiary objectives. Other policies and regulations that have been based, in part, on PRA methods and insights are the “Final Policy Statement on Technical Specification Improvement for Nuclear Power Reactors,” the alternative to 10 CFR Part 50 Appendix J, “Containment Leakage,” and the reactor siting criteria in the 1996 revision to 10 CFR Part 100, “Reactor Site Criteria.”

NRC is committed to ensuring that future regulatory initiatives will conform to the quantitative objectives of the safety goals. Additionally, in reviewing the need to apply new requirements to existing plants, NRC will consider the safety goals when evaluating the potential benefit expected from the new requirements. The process is discussed in NRC’s Backfit Rule, 10 CFR 50.109, “Backfitting.”

NRC also applies PRA and safety goal principles to enhance existing programs. For example, NRC uses PRA techniques in the Accident Sequence Precursor Program to evaluate conditional core damage frequencies. It applies the safety goals when setting priorities for resolving generic issues under the Program for Resolving Generic Issues.

10.3.3 Risk-Informed Regulation Implementation Plan

The Risk-Informed Regulation Implementation Plan discusses NRC’s actions to risk-inform its regulatory activities and specifically describes each of the activities identified as supporting the goals and objectives of the agency’s Strategic Plan and the Probabilistic Risk Analysis Policy Statement.

The plan has two parts. Part 1 provides a general discussion of the document’s relationship to the PRA Policy Statement and the Strategic Plan. It also discusses factors to consider in the process of risk-informing an agency requirement or practice, and provides guidance for selecting candidate requirements, practices and processes. Part 2 describes the staff’s ongoing risk-informed regulation activities in the Reactor Safety arena and the Materials and Waste Safety arenas. The latest version of the plan is available at the NRC Website: http://www.nrc.gov/what-we-do/regulatory/rulemaking/risk-inform.

10.3.4 Activities that Improve Data and Methods of Risk Analysis

NRC’s research activities consist of many issue-oriented projects, as well as more general work, such as developing and demonstrating risk analysis methods and risk-related training and guidance for NRC staff. Progress has been made in (1) analysis of low-power and shutdown accident risks, (2) computer tools for running the Systems Analysis Programs for Hands-On Integrated Reliability Evaluation (SAPHIRE), (3) analysis of uncertainties of the effects of severe accidents on the population offsite, (4) human reliability analysis, (5) containment response to high-pressure melt ejection (direct containment heating), hydrogen combustion, core melt-concrete interactions, debris coolability, and fuel-coolant interactions, (6) source terms,

(7) reactor vessel integrity under severe accident conditions, and (8) analytical codes for core melt progression.

SAPHIRE is an example of an improved risk analysis code to perform PRAs. Analysts use it to quantify accident risk measures for nuclear plants. Significant improvements have been made in its capabilities to support PRA applications. Work continues to enhance SAPHIRE to meet future needs.

NRC has improved methods and tools for human reliability analysis, such as “A Technique for Human Event Analysis” (ATHEANA), to identify and assess the likelihood of human failure events. NRC has applied this approach in PRAs for pressurized thermal shock. Applying lessons learned from PRAs and human reliability analysis and from developing and applying ATHEANA, NRC is developing guidance on human reliability analysis to provide “good practices” and evaluate various methods concerning “good practices.” This guidance will help address many aspects of human reliability analysis, including the ability of an individual method to support different regulatory applications, the need for consistency among practitioners in implementing the methods, and the necessary rigor needed for quantifying human reliability. In addition, NRC is developing a database entitled Human Event Repository and Analyses for use in both human factors and human reliability analysis. This activity includes developing a structure for collecting information suitable for the needs of human reliability analysis and quantitative approaches using Bayesian frameworks to quantify human failure events.

The significance of cooperation to improve regulatory priority to safety is exemplified by the efforts of NRC and stakeholders to establish a database concerning equipment reliability and availability to support the Maintenance Rule and other performance-based regulation. NRC will continue to work with industry representatives and other stakeholders to identify areas of mutual interest for the use of PRA methods and insights to encourage the use of plant-specific failure data.

10.3.5 Industry Activities and Pilot PRA Applications

NRC and industry representatives have cooperated in a number of activities and pilot programs to develop and apply risk-informed methodologies for specific regulatory applications. Lessons learned from these activities are used to enhance the effectiveness of developed guidance. The activities described in this section are inservice testing, inservice inspection, technical specification changes, standards development, and plant-specific changes to the licensing basis.

10.3.5.1 Risk-Informed Inservice Testing

In August 1998, NRC issued Regulatory Guide 1.175, “An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing,” to guide licensees on changes to the risk-informed inservice testing program. The agency subsequently completed a pilot application of risk-informed inservice testing in 1998, and has approved or is reviewing several other applications, generally of limited scope. For example, in August 2001, the staff granted a risk-informed exemption request from the licensee of the South Texas Project regarding special treatment requirements of low-risk and non-risk-significant safety-related nuclear components (including an exemption from prescriptive inservice testing requirements). Having successfully implemented this exemption, the staff is now developing a new rule, 10 CFR 50.69, discussed in Section 10.1 of this report, to allow risk insights to be applied to reduce the special treatment requirements in 10 CFR Part 50 for structures, systems, and components that are categorized as being of low risk significance.

As another example, the American Society of Mechanical Engineers (ASME) is updating the *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code) and applicable Boiler and Pressure Vessel (BPV) Code Cases to allow further use of risk insights in the inservice testing of pump and valves, and NRC amended 10 CFR 50.55a by publishing a final rule, entitled *Incorporation by Reference of ASME BPV and OM Code Cases*. This rulemaking incorporated by reference specific revisions of NRC Regulatory Guides 1.84, 1.147, and 1.192, which list ASME Code Cases that NRC accepts as alternatives to ASME Code requirements. Since Regulatory Guide 1.192 lists acceptable (and conditionally acceptable) OM Code Cases, including risk-informed categorization and component-specific code cases, licensees can now implement risk-informed inservice testing programs without following Regulatory Guide 1.175 and without prior NRC approval.

10.3.5.2 Risk-Informed Inservice Inspection

In September 2003, NRC issued Revision 1 to Regulatory Guide 1.178, "An Approach for Plant Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping" and the corresponding Standard Review Plan Chapter 3.9.8. The agency has also approved two industry methodologies, one developed by Westinghouse Owners Group and the other by EPRI, to develop alternatives to the ASME XI Inservice Inspection Program. The agency has approved programs for about 70 plants based on these two methodologies.

NRC participates in the ASME code development process. In this capacity, the staff has been involved in reviewing risk-informed inservice inspection Code Cases N-560, N-577, and N-578 and Appendix X. Staff activities also include continuing meetings with the industry to resolve issues such as the minimum ASME Class 1 sample size and extension of the risk-informed inservice inspection methodology to augmented inspection programs.

According to the information provided by Nuclear Energy Institute (NEI), 86 plants (units) are expected to implement risk-informed inservice inspection programs. The Institute also indicated that of the 86 submittals on risk-informed inservice inspection, 61 would be based on the methodology promulgated by EPRI, and 25 would be based on the methodology espoused by the Westinghouse Owners Group. As of the end of December 2003, 77 plants have submitted their programs. The staff has approved 67 programs and the remaining 10 programs are currently under review.

10.3.5.3 Risk-Informed Technical Specification Changes

Since the mid-1980s, NRC has been reviewing and granting improvements to technical specifications that are based, at least in part, on PRA insights. In its final policy statement on technical specification improvements of July 22, 1993, the Commission stated that it expects that licensees will use any plant-specific PRA or risk survey in preparing submittals related to technical specifications. The Commission reiterated this point when it revised 10 CFR 50.36, "Technical Specifications," in July 1995.

In August 1998, NRC issued Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," and a companion Standard Review Plan chapter. These documents guide licensees on making risk-informed changes to plant technical specifications. The agency is using this regulatory guide as well as Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," (issued in July 1998) to improve plant technical specifications. These improvements are intended to maintain or improve safety while reducing unnecessary burden and to bring technical specification requirements into congruence with NRC's other risk-informed regulatory activities

The industry and NRC have been pursuing increased use of PRA in developing improvements to technical specifications. The industry and the staff have identified the following eight initiatives to date for risk-informed improvements to the standard technical specifications:

(1) Define the preferred end state for technical specification actions (usually hot shutdown for PWRs).

(2) Increase the time allowed to delay entering required actions when a surveillance is missed.

(3) Modify existing mode restraint logic to allow greater flexibility (i.e., use risk assessments for entry into higher mode limiting conditions for operation based on low risk).

(4) Replace the current system of fixed completion times with reliance on a configuration risk management program.

(5) Optimize surveillance frequencies.

(6) Modify limiting conditions for operation 3.0.3 actions to allow for a risk-informed evaluation to determine whether it is better to shut down or to continue to operate.

(7) Define actions to be taken when equipment is not operable but is still functional.

(8) Risk-inform the scope of the Technical Specifications Rule.

To date, Initiatives 2 and 3 have been approved by the staff and the majority of licensees have adopted them. There is great interest in the other initiatives and they are in various stages of review. All of the initiatives involve, to some prescribed degree, assessing and managing plant risk using a configuration risk management program consistent with and in some cases exceeding the requirements of the Maintenance Rule in 10 CFR 50.65.

10.3.5.4 Development of Standards

The NRC worked with ASME to develop a national consensus standard for PRA covering internal initiating events. Staff members worked actively with industry and ASME participants to resolve comments on draft revisions of the standard. The standard was issued by ASME in April 2002 as ASME-RA-5-2002. The first addendum was published in December 2003 as ASME-RA-5a-2003.

In parallel, the staff worked with the American Nuclear Society to develop companion standards covering PRAs for external events, low power, and shutdown operations. The PRA standard for external events was issued for public comment, revised, and issued as a final document in December 2003. Work on the low power and shutdown PRA standard is progressing.

In addition, the industry, represented by the Nuclear Energy Institute, issued NEI-00-02, "PRA Peer Review Process Guidelines," for a licensee to use in assessing whether its PRA is adequate to support different classes of risk informed application, and submitted it for NRC review.

In December 2003, the NRC published Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," for trial use. This regulatory guide describes one acceptable approach for determining that the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results such that the PRA can be used in regulatory decision making. It endorses, with comments, the ASME PRA standard and the NEI peer review guidance. Several licensees have volunteered to pilot risk-informed applications that rely on the regulatory guide. The staff plans to update the regulatory guide as it gains experience with the pilot applications. Further revisions to the regulatory guide are planned to incorporate the American Nuclear Society standards as they are published.

10.3.6 Activities that Apply Risk Assessment to Plant-Specific Changes to the Licensing Basis

In its approval of the policy statement on the use of PRA methods in nuclear regulatory activities, the Commission expected that "the use of PRA technology should be increased in all regulatory matters . . . in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy." Risk-informed decisions are expected to be reached in an integrated fashion, considering both traditional engineering, such as defense-in-depth and safety margins, and risk information, and may be based on qualitative factors as well as quantitative analyses and information. The degree to which the risk insights play a role in the decision depends on the specific application, but being *risk-informed* (as opposed to *risk-based*), decisions are not to be driven solely by numerical results from a PRA. PRA results are but one input into the decisionmaking and help in building an overall picture of the implications of the proposed change on risk. The risk-informed guidance documents described above are used by licensees to prepare proposals for plant-specific changes to their licensing bases. In addition, Regulatory Guide 1.174 and Standard Review Plan Chapter 19, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance," are notable for their guidance on using PRA to support licensees' requests for changes to a plant's licensing bases (such as license amendments and changes to technical specifications). NRC has been approving many requests for license amendments on the basis of the guidance of Regulatory Guide 1.174. As a result of lessons learned from reviewing these requests, the staff issued Revision 1 to Regulatory Guide 1.174 in November 2002. For details concerning this guidance, see the NRC's public Web site at http://www.nrc.gov/reading-rm/doc-collections/reg-guides/power-reactors/active/01-174/.

10.4 Safety Culture

This section has been added as a result of the interest shown by the contracting parties at the Second Review Meeting. Discussions of both NRC's safety culture and the licensee's safety culture are presented to provide a complete picture of safety culture issues in the United States.

10.4.1 NRC Safety Culture

In response to questions about NRC's safety culture raised during the peer review of the 2001 *U.S. National Report*, NRC explained that the Inspector General was conducting a safety culture survey at NRC. Subsequently, the completed "2002 Survey of NRC's Safety Culture and Climate" (1) measured NRC's safety culture and climate, (2) compared results against NRC's 1998 Safety Culture and Climate Survey, and (3) compared results to Government and national benchmarks. The survey included a qualitative design phase, consisting of interviewing a random sample of NRC employees and managers, and a quantitative component consisting of a survey offered to all NRC employees. Survey questions were grouped into 18 categories representing the major topics of NRC's safety culture and climate. The categories singled out for particular comment are summarized below.

The survey concluded that NRC safety culture and climate continued to show improvement since the 1998 survey. Specifically, the workforce views itself as effective and dedicated to NRC's safety mission. Comparison with the 1998 survey results also showed improvement in virtually every category or topical area. Further, the survey found that most scores exceed the established national benchmarks for Government research and technical composites.

The *Future of the NRC* category measured employee concerns over reductions-in-force, changes in management, technology, regulatory methodology, the federal government, the future of one's work unit, NRC, and the industry, as well as fears of one's skills becoming obsolete. This category was singled out for particular comment as its scores have shown dramatic positive improvement (18 percentage points) between 1998 and 2002. The 2002 results demonstrate a 30-point advantage over other Government agencies, a 20-point advantage over research and development organizations, and an 11-point advantage over the broader U.S. workforce.

Another particularly positive finding was the significant increases within the *Organizational Change* category. This category rates how the following have changed from the past and will change in the future: the way people are managed day to day, communication, quality of work produced, productivity, public image of the agency, and NRC as a whole. This category is critical as it is typically correlated strongly with employee perceptions of stability, and ultimately their desires to remain with an organization.

The study also identified one area of concern. The *Continuous Improvement Commitment* category, which assesses employee views on NRC's commitment to public safety, and whether employees are encouraged to communicate ideas to improve safety/regulations/operations, showed only minimal improvement (3 percentage points) over the 1998 survey. This score is well below the norm for other U.S. Government research and technology agencies, U.S. research and development organizations, and the broader U.S. labor force. However, the positive news in considering the scores below the norm for this category is that dramatic improvement was demonstrated among the *Future of NRC* category that tends to focus on items that evaluate employees' views on how NRC's regulation of its licensees has changed in the past year. Improvement in topics under this category can often positively impact issues gauged in a category such as *Continuous Improvement Commitment.*

Overall, the study concluded that NRC had a variety of strengths to build from, in that the results are very positive in relation to a wide variety of norms and show significant improvement in comparison with the 1998 survey. The entire study is available on NRC's public Web site at http://www.nrc.gov/reading-rm/doc-collections/insp-gen/2003/03a-03.pdf.

In response to the Inspector General's Safety Culture and Climate Survey, NRC made a number of improvements, which include the following. It created a Communications Council to improve internal communications The Council is composed of high-level staff from a variety of program offices who meet monthly to plan, coordinate and implement internal communication strategies, and share best practices that add value across the agency. NRC also developed a strategy for improving leadership training and maintaining leadership skills at each managerial level across the agency. In addition, it created an internal web page that points employees to the available resources across the agency where they can discuss professional, personal or supervisory issues - even when they wish to remain confidential. And finally, NRC has encouraged senior managers to continually emphasize the primary importance of NRC's safety mission in their communications to employees.

10.4.2 Licensee Safety Culture

This section covers the policies, programs, and practices that apply to safety culture and the Davis Besse case. The Commission is considering revisons to its policies on the safety conscious work environment and safety culture, and the final Commission decisions in this area will be available on the NRC web site at http://www.nrc.gov.

Governing Documents and Process

NRC recognizes that licensee "management has a duty and an obligation to foster the development of a safety culture at each facility and to provide a professional working environment, in the control room and throughout the facility, that assures safe operations." However, this "Policy Statement on the Conduct of Nuclear Power Plant Operations," issued January 24, 1989, does not limit NRC's authority to act on matters affecting the safe operation of the plants. The implementation section of the Commission's Policy Statement states that licensee management "should routinely monitor the conduct of operations...and review their procedures and policies of operations...to assure they support an environment for professional conduct."

NRC's current regulations address several important attributes of safety culture. For example, Appendix B to 10 CFR Part 50 requires the licensees to establish a quality assurance program. Quality assurance means "all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service." Criterion XVI of Appendix B, "Corrective Actions," states, "Measures shall be established to ensure that conditions that are adverse to quality (such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances) are promptly identified and corrected." Conditions that will promote quality as envisaged in Appendix B include adherence to procedures and an effective corrective action program. All of these attributes are components of safety culture.

The Reactor Oversight Process accounts for safety culture through its three "cross-cutting" issues: Human Performance, Safety-Conscious Work Environment, and Problem Identification and Resolution. A sampling of letters from NRC regional offices to plant managers presenting findings of inspections carried out under the Reactor Oversight Process shows that the staff focuses considerable attention on aspects of safety culture. Findings such as "plant personnel focused on replacement rather than understanding causes of wear" and "industry experience was not incorporated so as to minimize wear" could be said to reflect two aspects of safety culture that are commonly cited, namely, a "questioning attitude" of personnel and the plant's "organizational learning." These findings are based on observations related to specific incidents (i.e., they are based on actual licensee performance).

The NRC Inspection Manual provides adequate guidance to ensure that licensees are detecting and correcting problems. Inspection Procedure 71152, "Identification and Resolution of Problems," requires that every two years, the inspectors select a sample of conditions that are adverse to quality that the licensee has processed through its corrective action program.
The purpose is to focus on problem identification, resolution, and the effectiveness of corrective

actions. Appendix 1 to this inspection procedure lists a number of questions that are intended to help the inspectors assess whether there are impediments to the establishment of a safety-conscious work environment. Appendix 1 states, “It is not intended that these questions be asked verbatim, but rather that they form the basis for gathering insights regarding whether there are impediments to the formation of a safety-conscious work environment.”

Concerning a safety-conscious work environment, NRC has a regulation, 10 CFR 50.7, “Employee protection,” that prohibits licensees from firing or taking adverse actions against employees who raise safety issues. NRC also evaluates allegations from plant workers regarding safety culture issues. (See Section 6.2.8 of this report for a discussion of the allegation program. To add to that discussion, an allegation can be treated confidentially, if the alleger wishes; confidentiality encourages allegers to come forward.) In addition, the Policy Statement on “Freedom of Employees in the Nuclear Industry to Raise Safety Concerns without Fear of Retaliation” sets forth expectations that licensees will establish and maintain safety-conscious environments in which employees feel free to raise safety concerns, both to their management and to NRC, without fear of retaliation.

In addition, NRC performs supplemental or special inspections when significant issues are identified. These inspections focus resources in the area of problem identification and resolution so as to ensure that licensees are effectively identifying, assessing, and correcting performance deficiencies. Thus, through their focus on the licensee’s efforts to determine and correct the root causes of problems, these inspections reveal certain aspects of safety culture. Both the baseline and supplemental inspection programs encourage inspectors to identify issues related to cross-cutting areas that are directly related to safety culture, such as the adequacy of human performance, the establishment of a safety-conscious work environment, and the robustness of the problem identification and resolution program.

Licensees with plants in an extended shutdown may be subject to increased NRC inspection effort under the process specified in Inspection Manual Chapter 0350 before restart. The increased inspection includes evaluating the three cross-cutting areas and focuses on those that contributed to the plant’s shutdown, and other associated risk-significant issues. Inspection resources focus on determining the root causes of the shutdown, identifying and resolving risk-significant issues, and ensuring an appropriate response. Safety culture issues are addressed to the extent that they are associated with the problem being examined.

In keeping with the Commission guidance, the staff does not conduct direct evaluations or inspections of safety culture as a routine part of assessing licensee performance. For nuclear power plants, the staff relies on the Reactor Oversight Program to monitor some underlying elements (or aspects) of safety culture. In the process of monitoring these elements, NRC will intervene should the need arise. If the NRC determines that there may be potential issues with a licensee's safety conscious work environment or safety culture, then this concern is elevated to NRC management for further discussion, evaluation, and possible action.

For example, concerning the Salem and Hope Creek Nuclear Power Plants, in a January 28, 2004, letter, the NRC Region I staff notified PSEG (the licensee for the Salem and Hope Creek plants) of interim results of an ongoing NRC special review of the work environment at the facilities. The letter outlined areas of NRC concerns, particularly as they relate to the handling of emergent equipment issues and associated operational decision making. On March 18, 2004, the NRC held a management meeting with PSEG, which was open to the public for observation, at which PSEG summarized its plans to assess the work environment at Salem and Hope Creek.

The NRC questioned PSEG on certain aspects of its plan, including how specific events and discrete issues would be reviewed.

The licensee formed an independent assessment team that interviewed plant staff and extensively reviewed documents. That team's findings are consistent with other recent surveys. PSEG is now developing an action plan to address the team's findings and improve the work environment at Salem and Hope Creek. NRC Region I staff has been following these activities closely and will be scheduling a meeting with PSEG in the near future to discuss the results of PSEG's assessment. NRC Region I staff will continue to maintain a focus on PSEG's progress in implementing planned actions to address issues that might arise regarding the work environment, as well as actions taken or planned to strengthen the sites' corrective action programs. Additionally, NRC management will continue to make visits on-site and NRC staff will continue to conduct inspections to oversee selected aspects of PSEG actions to improve the work environment.

Davis-Besse

In the wake of the discovery of the reactor vessel head degradation at the Davis-Besse Nuclear Power Plant, weaknesses in the licensee's safety culture were identified as a key contributor in not identifying the corrosion of the reactor vessel head in a more timely manner. Therefore, on the basis of Appendix B Criterion XVII of Part 10 CFR 50, the NRC performed special inspections to evaluate the processes used by Davis-Besse to assess their safety culture and to evaluate their corrective action plans. The inspection teams included experts in human performance and organizational effectiveness, as well as former industry executives with a track record of improving safety culture at problem nuclear power plants. The inspection reports are available on NRC's public Web site at http://www.nrc.gov/reactors/operating/ops-experience/vessel-head-degradation.html.

NRC's Management and Human Performance Inspection Team evaluated the licensee's tools for monitoring the effectiveness of the corrective actions taken in response to the vessel head degradation condition. The team also evaluated the licensee's actions to improve and protect the site's safety-conscious work environment, and the tools the licensee put in place to monitor the effectiveness of those actions. The team evaluated the internal and external safety culture assessment tools, the current safety-conscious work environment at the site, the activities of the safety-conscious work environment review team, and the current status of the employee concern program. The team reviewed documents, interviewed individuals, observed management activities, and evaluated licensee survey results. The team's most significant conclusions included the following examples:

- The licensee's internal safety culture monitoring tools, including employee concern program surveys, Nuclear Quality Assurance surveys, and the restart readiness review business practice, when taken together, provide an appropriate examination of the site's safety culture. The internal tools generally follow the concepts in internationally recognized guidance from the IAEA, the International Nuclear Safety Group, and the Nuclear Energy Agency.

- The licensee's external, independent safety culture assessment was appropriately designed and implemented. The process provided a comprehensive review of safety culture traits using methods, concepts, and focus areas accepted by the international nuclear community.

- Actions to improve the safety-conscious work environment at Davis-Besse since the event have been effective. However, it can be further improved.

- To ensure sustained licensee performance, NRC, in addition to its approval for restart, required, by a confirmatory order, annual assessments of organizational safety culture, including the safety conscious work environment, for five years.

NRC's Oversight Panel, which has been coordinating the agency's activities at Davis-Besse after the identification of reactor vessel head degradation, used the results of the special inspections, combined with results from other inspections, to evaluate the effectiveness of the utility's management and human performance corrective actions. This aspect of plant performance was necessary for the approval of the safe restart and operation of the facility, which occurred on March 8, 2004.

NRC's Response to Davis-Besse

An evaluation by the Advisory Committee on Reactor Safety, an independent advisory committee to the Commission, discussed shortcomings identified by NRC's Lessons-Learned Task Force concerning NRC's oversight of the safety culture aspects of Davis-Besse:

> The NRC staff's Lessons-Learned Task Force concluded that: (1) NRC failed to adequately review, assess, and followup on relevant operating experience, and (2) NRC failed to integrate known or available information into its assessments of Davis-Besse's safety performance.

The Davis-Besse Lessons Learned Task Force made numerous recommendations about improving NRC's processes, some of which relate directly to safety culture. For example, one recommendation addresses the issue of "maintaining a questioning attitude in the conduct of inspection activities." The Committee agreed with this recommendation, but believes that the agency's safety culture is fundamentally sound; that NRC is focused on safety, and safety issues receive the attention warranted by their significance. The Committee in its evaluation also emphasized the need to keep safety culture in perspective, saying that the industry and NRC have mature programs to monitor reliability of equipment and simulator testing of control room crews to monitor human reliability; and that awareness of safety culture adds to understanding and management of the deeper causes that shape human performance.

The agency is currently assessing its programs and policies. The Commission has been reviewing recommendations of the Lessons-Learned Task Force and is considering potential changes to NRC's regulatory program to respond to the identified shortcomings in NRC's oversight of Davis-Besse.

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 regarding the financial resources that licensees must have to support the nuclear installation throughout its life, including the financial resources needed for safety improvements that are made during a plant's operation, decommissioning, as well as handling claims and damages associated with accidents. This section also explains the regulatory requirements for qualifying, training, and retraining personnel.

This section was updated to incorporate changes in the dollar amounts for liability under the Price Anderson Act, decommissioning, and experience and examples.

11.1 Financial Resources

Adequate funds for safe construction, operation, and decommissioning are necessary for the protection of public health and safety. Although there does not appear to be a direct correlation between a licensee's current finances and the current operational safety at a specific facility, there is some evidence that, financial pressures have limited the resources that are devoted to corrective actions, plant improvements, upgrades, and other safety-related expenditures. Further, because a power reactor must operate to provide revenues for eventual plant decommissioning, any shutdown of a plant before its owner has accumulated sufficient funds for decommissioning could potentially hinder the safe decommissioning of that plant.

Additionally, many States have initiated or completed actions to economically deregulate their nuclear power plants. Traditionally, nuclear power plant owners in many States have been large, vertically integrated companies with substantial assets in generation, transmission, and distribution. In exchange for having exclusive franchises to provide electric power in defined geographical areas, nuclear plant owners have had the rates they charge to their customers regulated by State governmental bodies. This system of rate regulation has ensured a source of funds for construction, operation, and decommissioning of nuclear power plants. Nonetheless, this model of rate regulation has been changing and, as discussed below, the United States has adjusted its processes.

NRC distinguishes between financial qualifications for construction, operation, as well as decommissioning of nuclear power plants, and has separate regulations and programs that apply to each. NRC also implements programs to ensure that the public has financial protection for bodily injury and property damage losses in the event of an accident. Finally, the agency has implemented requirements to ensure that licensees have insurance to help pay onsite recovery costs resulting from accidents to provide funds for post-accident restart or decommissioning. As part of its initial licensing reviews, NRC must also determine whether activities that are conducted under a license will create or maintain a situation that is inconsistent with the antitrust laws of the United States. However, because these reviews do not directly apply to the protection of public health and safety, they are not addressed further in this article.

11.1.1 Financial Qualifications Program for Construction and Operations

This section explains the financial qualifications program for construction and operations. It covers the governing documents and process used to implement requirements. It explains NRC reviews for construction permits, operating licenses, combined licenses, post-operating non-transferred licenses, and transfers of licenses.

11.1.1.1 Governing Documents and Process

Section 182.a of the Atomic Energy Act of 1954, as amended, provides that "Each application for a license… shall specifically state such information as the Commission, by rule or regulation, may determine to be necessary to decide such of the technical and financial qualifications of the applicant… as the Commission may deem appropriate for the license." To implement this provision, NRC has developed the following regulations and guidance:

Construction Permit Reviews: 10 CFR 50.33(f)(1) requires applicants for construction permits to submit information that "demonstrates that the applicant possesses or has reasonable assurance of obtaining the funds necessary to cover estimated construction costs and related fuel cycle costs." Appendix C to 10 CFR Part 50 provides more specific directions for evaluating the financial qualifications of applicants.

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 for the total annual operating costs for each of the first 5 years of operation of their facilities, and must also indicate the sources of funds to cover operating costs.

Combined License Application Reviews: As authorized in 10 CFR Part 52, "Early Site Permits, Standard Design Certifications, and Combined Licenses for Nuclear Power Plants," applicants may apply for a combined construction permit and operating license. Under 10 CFR 52.77, "Contents of applications; technical information," all such applications must contain all of the information required under 10 CFR 50.33, including information regarding financial qualifications. The review procedures described above are used to review any future combined license applications that NRC receives.

Post-Operating License Non-Transfer Reviews. NRC does not systematically review the financial qualifications of power reactor licensees once it has issued an operating license, other than for license transfers as described below. However, as provided in 10 CFR 50.33(f)(4), NRC can seek additional information on licensees' financial resources if the agency considers such information appropriate.

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

The overall process for NRC review of applicants’ and licensees’ financial qualifications for nuclear power plant construction and operation is described in the “Standard Review Plan on Power Reactor Licensee Financial Qualifications and Decommissioning Funding Assurance,” NUREG-1577, Rev. 1.

11.1.1.2 Experience and Examples

NRC has not received a construction permit application for a nuclear power plant for more than 25 years. Thus, the agency has not performed any financial qualifications reviews of applications for construction permits since the 1970s. The last power reactor unit to receive an operating license was Watts Bar Unit 1, in 1996. Because Watts Bar is owned and operated by the Tennessee Valley Authority, which meets NRC’s definition of an “electric utility,” the facility is exempt from an operating license financial qualifications review.

However, NRC has conducted several post-licensing reviews under the license transfer provisions of 10 CFR 50.80. To foster free market competition, Federal and State governments have continued to deregulate the wholesale and retail sale of electricity under their jurisdictions; therefore, several NRC power reactor licensees have restructured themselves in various ways to meet increased competition. Since 1999, more than 15 plants have been sold. Additionally, power plant licensees have merged with other companies or have formed parent holding companies. NRC has viewed each of these actions as a direct or indirect transfer of a license under the provisions of 10 CFR 50.80.

11.1.2 Financial Qualifications Program for Decommissioning

This section covers the governing documents and process to implement requirements, experience, and examples.

11.1.2.1 Governing Documents and Process

Among other sections of the Atomic Energy Act, Section 182.a establishes the basis for NRC’s decommissioning funding assurance regulations and guidance, as follows:

(1) NRC’s regulations governing decommissioning funding assurance for plants under construction and operating nuclear power plants are contained in 10 CFR 50.75, “Reporting and Record Keeping for Decommissioning Planning.”

(2) NRC’s regulations for nuclear power plants that have permanently ceased, or within five years of permanently ceasing operations are provided in 10 CFR 50.82, “Termination of License.”

(3) The overall process for NRC review of applicants and licensees with regard to decommissioning funding assurance is described in the “Standard Review Plan on Power Reactor Licensee Financial Qualifications and Decommissioning Funding Assurance,” NUREG-1577, Rev. 1.

(4) Guidance on decommissioning funding assurance provisions and mechanisms is contained in “Assuring the Availability of Funds for Decommissioning Nuclear Reactors,” Regulatory Guide 1.159.

(5) Guidance on preparing the cost estimates required by NRC’s regulations is contained in “Standard Review Plan for Decommissioning Cost Estimates for Nuclear Reactors,” NUREG-1713, Draft Report for Comment, November 2001.

(6) Guidance on applying the escalation factors for the generic formula in 10 CFR 50.75 is contained in NUREG-1307, “Report on Waste Burial Charges,” Rev. 10, dated October 2002.

11.1.2.2 Experience and Examples

Several U.S. power reactors have permanently terminated operations. Some have completed decommissioning and terminated their NRC licenses, and others are in various stages of decommissioning. The licensees of many shutdown plants have decided to defer final dismantlement. This is generally the case at multi-unit sites where other reactors continue to operate. The Shippingport plant, which was the first commercially operated reactor, rated at 65 MW(e), cost $98 million to decommission. The Trojan Plant, an 1,130-MW(e) pressurized-water reactor is currently undergoing dismantlement. The licensee estimates that radiological decommissioning will cost $239 million (1997 dollars); as of the end of 2002, it had spent approximately $175 million of the estimated total.

11.1.3 Financial Protection Program for Liability Claims Arising From Accidents

This section explains the financial protection program for liability claims arising from accidents. It covers the governing documents, primarily the Price-Anderson Act, and process to implement requirements. It also discusses relevant experience and gives examples.

11.1.3.1 Governing Documents and Process

The Price-Anderson Act, enacted in 1957, became Section 170 of the Atomic Energy Act of 1954, as amended, and governs the U.S. financial protection program. Section 170 (with related definitions in Section 11) provides the financial and the legal framework to compensate those who suffer bodily injury or property damage as a result of accidents at covered nuclear facilities. NRC’s regulations implementing the provisions of Section 170 for NRC licensees are codified in 10 CFR Part 140, “Financial Protection Requirements and Indemnity Agreements.”

The Price-Anderson Act was enacted to meet two basic objectives:

(1) Remove the deterrent to private sector participation in atomic energy presented by the threat of potentially enormous liability claims in the event of a catastrophic nuclear accident.

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

In enacting the Price-Anderson Act, the U.S. Congress sought an equitable balance of industry’s needs with those of the public. Specifically, Congress required that all reactor licensees (sometimes termed “operators”), among other licensees, purchase specified amounts of liability insurance (then at a maximum level of $60 million) or possess other equal financial protection against the risk of a nuclear accident. Beyond the financial protection, the U.S. Government agreed to indemnify the licensees for each accident for an additional $500 million of damages in order to achieve reasonable compensation for the public; it also agreed to limit the total liability for an accident. The limit was set at the sum of the required financial protection and the government indemnity (thus a maximum limit of $560 million which was applicable largely to commercial power reactors.)

With regard to commercial nuclear reactors, Price-Anderson Act has been revised several times since 1957, the most recent being when Congress renewed the Commission's authority to cover new facilities until December 2003. As revised over the years, the means for providing financial protection for power reactors over 100 MWe have changed significantly. Under current law, those reactors must contribute to a pool that replaces the government as the second provider of funds if the first layer of financial protection (liability insurance--now $300 million) is exhausted.

The reactor operators are required after an accident to pay into a "retrospective premium pool," in maximum annual installments not exceeding $10 million, up to a total of $95.8 million each. But payment is called for only if the accident exhausts the first layer of financial protection, and only if and to the extent that additional funds are needed to pay the damages. With 104 reactors currently participating in the system, the total financial protection available under the Price-Anderson Act for any one accident is $9.96 billion [$300 million primary coverage + (95.8 million per reactor x 104 reactors)]; $9.96 billion is also the limit on liability. As reactors leave the retrospective premium system as a result of permanent closure or join as the result of construction of new reactors, this coverage limit may fall or rise. A change in the limit may also occur when the $95.8 million contribution is adjusted for inflation, as must be done every five years. In any event, Congress will address any damages exceeding the total sum that reactors must contribute to the pool and will decide upon the next steps needed for compensation.

The public is significantly benefitted by another feature of the Act. Claimants need only prove that the accident caused their injury in order to receive compensation for damages from any accident with significant offsite releases of radiation (i.e., an "extraordinary nuclear occurrence"). No proof of fault is necessary, nor is proof of what caused the accident.

11.1.3.2 Experience and Examples

Claims for more than 130 alleged incidents involving nuclear material have been filed under various liability policies since the inception of the Price-Anderson Act in 1957. Earlier claims tended to be property damage claims arising from alleged radiation from leakage or from other accidents involving containers of nuclear materials in transit. More recent claims have emphasized bodily injury arising from alleged radiation exposure, especially by contractor employees working at the sites of operating nuclear power plants. The insured losses and expenses paid so far total more than $100 million. Of this amount, most payments arose out of the accident at TMI Unit 2.

11.1.4 Insurance Program for Onsite Property Damages Arising From Accidents

Among other sections of the Atomic Energy Act, Section 182.a provides the basis for NRC's onsite property damage insurance requirements for operating nuclear power reactors contained in Paragraph (w) of 10 CFR 50.54, "Conditions of Licenses." Power reactor licensees are required to obtain onsite property damage insurance, or an equivalent source of protection, that would be used, before any other purpose, to ensure that the reactor is maintained in, or is returned to, a safe and stable condition, and that radioactive contamination is removed or controlled so that personnel exposures are consistent with the occupational exposure limits in 10 CFR Part 20, “Standards for Protection Against Radiation.” NRC specified that this coverage had to be at least $1.06 B. Most licensees purchase the maximum onsite insurance currently available (about $3 B). In the event of an accident, licensees would be able to use the proceeds from this insurance above NRC-required amount, for example, to pay for the costs to replace equipment that is damaged in the accident.

The U.S. nuclear industry has not experienced an accident of significant radioactive contamination since TMI Unit 2.

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, process for implementing requirements, experience, and examples. It also discusses the Institute for Nuclear Power Operations (INPO) accreditation activities.

11.2.1 Governing Documents and Process

10 CFR Part 55, “Operator Licensing,” regulates the training requirements for licensed operators and licensed senior operators, while allowing facility licensees to have operator requalification program content that is derived using a systems approach to training, or that meets the requirements outlined in paragraph (c)(1) of 10 CFR 55.59, “Requalification.” Subpart D, “Applications,” of 10 CFR Part 55 requires that operator license applications must contain information about an individual’s education and experience. The operator licensing process at power reactors includes a generic fundamentals examination covering the theoretical knowledge that is required to operate a nuclear power plant, as well as a site-specific examination, which consists of a written examination and an operating test that includes a plant walkthrough and a dynamic performance demonstration on a simulation facility. License applicants must pass the generic fundamentals examination before they can take the site-specific examination.

NRC staff has worked to redefine the operator licensing function to transfer additional responsibility for developing examinations to licensees. In 1999, NRC amended 10 CFR Part 55, “Operators’ Licenses,” to allow nuclear power reactor licensees to prepare the written examinations and operating tests that NRC uses to evaluate the competence of applicants for operators’ licenses at those facilities. Licensees that elect to prepare their own examinations are required to establish procedures to control examination security and integrity and to prepare and submit proposed examinations and operating tests to NRC according to the guidance in NUREG-1021, “Operator Licensing Examiner Standards,” Rev. 8, Supplement 1 issued in April 2001. NRC reviews facility-prepared examinations, administers all operating tests, and makes the final licensing decisions.

10 CFR 50.120, “Training and Qualification of Nuclear Power Plant Workers,” requires that training programs be established, implemented, and maintained using a systems approach to training for eight categories of nonlicensed workers at nuclear power plants and the shift supervisor who is licensed. 10 CFR 50.120 complements the requirements for training based on a systems approach for the requalification of licensed operators.

Regulatory Guide 1.8, “Personnel Selection and Training” contains guidance to implement the regulations.

NRC monitors industry performance in implementing the training requirements of 10 CFR Parts 50 and 55 by (1) reviewing licensee event reports and inspection reports for training issues, (2) observing the accreditation process, and (3) reviewing the results of operator licensing activities. Guidance for inspecting the aspects of the operator training programs that are unique to requalification is given in Inspection Procedure 71111.11, “Licensed Operator Requalification Program.” In addition, NRC verifies compliance with the requirements for training based on a systems approach through its inspection program when appropriate for cause, using Inspection Procedure 41500, “Training and Qualification Effectiveness,” which references the guidance in NUREG-1220, Rev. 1 “Training Review Criteria and Procedures.”

NRC has endorsed the training accreditation process managed by INPO and 12 accredited training programs. In issuing 10 CFR 50.120 and revising 10 CFR Part 55, the Commission reaffirmed its conclusion that currently accredited training programs can meet the requirements of 10 CFR Part 55 and 10 CFR Part 50.120. The staff recognizes that training programs developed in accordance with INPO guidelines and accredited by the National Nuclear Accrediting Board (the Board) are based on a systems approach to training; therefore, accredited programs are considered to be consistent with the regulations. NRC also recognizes INPO managed accreditation and associated training evaluation activities as 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, NRC monitors INPO accreditation activities as part of its assessment of the effectiveness of the industry’s training programs. (NRC also monitors the selected performance areas of its licensees as part of its assessment.) NRC monitors INPO activities by observing accreditation team visits and the monthly National Nuclear Accrediting Board meetings. These visits are intended to monitor the implementation of programmatic aspects of the accreditation process.

Placing a training program on probation or withdrawing accreditation indicates a Board concern. It does not necessarily place a training program in noncompliance with either 10 CFR 50.55 or 10 CFR.120, since training programs are accredited to a “standard of excellence” rather than a minimum level of regulatory compliance. However, NRC does review the circumstances leading to the withdrawal or probation to ensure safe operations and continued compliance with regulations.

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

11.2.2 Experience and Examples

NRC reviewed training issues contained in licensee event reports and inspection reports at the end of 2002 using data from the Human Factors Information System. (This system is also described in Article 12.) This review revealed that, over the three-year review period, the contribution of training related deficiencies to overall human performance has decreased from about 8 percent in 1999 to 4.5 percent in 2002 for the industry as a whole. The identified training issues continue to be concentrated in two distinct areas — “training less than adequate” and “individual knowledge less than adequate.”

NRC has concluded that INPO managed accreditation continues to be an acceptable means of ensuring that the training requirements in 10 CFR Parts 50 and 55 are being met. Although NRC monitoring of training gives some indication of limited specific weaknesses in training programs, all indicators suggest that the industry is successfully implementing training programs in accordance with the regulations. Monitoring of selected performance areas will continue with emphasis placed on identifying training process problems and ensure that they are appropriately resolved.

An example of this monitoring process is a supplemental inspection conducted at Indian Point 2 in April 2002. That inspection was conducted to ensure that the root causes and contributing causes of the crew high failure rate during facility-administered annual licensed operator requalification examinations were understood, to independently assess the extent of the condition, and to ensure that the corrective actions to risk significant performance issues were sufficient to address the causes, and to prevent recurrence. The inspectors noted that the licensee’s evaluation identified a fundamental underlying weakness: the station has yet to overcome cultural weaknesses that include an unwillingness to confront poor performance, over reliance on procedures to change behavior, and compartmentalization. The licensee identified three root causes, including (1) operations training had not focused on the basic building blocks that ensure a healthy program, (2) the station had not maintained a core of career-oriented, plant-knowledgeable instructors and operators, and (3) Operations Department involvement with operations training had often been ineffective. The inspectors concluded that the licensee’s root cause evaluation was reasonable.

ARTICLE 12. HUMAN FACTORS

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

This section explains NRC's program on human performance. The seven major areas under this program are (1) human factors engineering issues, (2) emergency operating procedures and plant procedures, (3) working hours and staffing, (4) fitness-for-duty, (5) human factors information system, (6) support to event investigations and for-cause inspections, and (7) training. This section also discusses research activities.

This section was updated to incorporate new experience and examples; new documents; and effects of the terrorist attacks on September 11, 2001, as they relate to human factors issues.

12.1 NRC Program on Human Performance

This section discusses NRC Program on Human Performance and significant activities performed under this program.

12.1.1 Goals and Mission of the Program

NRC has a comprehensive program for ensuring that human performance is properly addressed in a risk-informed regulatory framework for maintaining reactor safety. The agency developed the program based on reviewing risk information, as well as information from sources such as activities in the domestic and international nuclear industry. SECY-00-0053, "NRC Program on Human Performance in Nuclear Power Plant Safety," dated February 29, 2000, describes the program in detail.

12.1.2 Program Elements

Human performance is incorporated in the Reactor Oversight Process, plant licensing and monitoring, the Risk-Informed Regulation Implementation Plan, and emerging technology and emerging issues.

The Reactor Oversight Process (discussed in Article 6) focuses on cornerstones of safety which are assessed through a combination of performance indicators and risk-informed inspections. The inspections focus on risk-significant activities and systems related to the cornerstones. The three elements that are considered cross-cutting to the cornerstones are human performance, safety-conscious work environment, and corrective actions. The Human Performance Program has contributed directly to the development of a supplemental inspection procedure related to the human performance cross-cutting element. This program also depends on the other two elements, since a safety-conscious work environment and many of the actions involved in corrective action programs result from human performance problems.

Activities that apply to plant licensing and monitoring include reviewing licensing actions and monitoring plant and program performance. As part of regulatory initiatives, the staff reviews Commission policies related to human performance that address identified problems. The staff also supports rulemaking and the development of regulatory guidance.

Support to the Risk-Informed Regulation Implementation Plan includes generating, collecting, and evaluating data on human performance for use in human reliability analysis models. NRC staff evaluates information to gain insights to support risk-informed regulation and to provide human performance data for human reliability analysis.

Activities that apply to emerging technology and emerging issues are intended to prepare NRC for the future. The two activities applying to these categories are developing regulatory guidance for reviewing designs of control stations and processing requests related to deregulation. Licensees are replacing aging analog controls and displays with digital components, and the agency needs to be prepared to review safety issues that arise from human-system interfaces resulting from such new designs and technologies. With regard to deregulation, NRC has been processing numerous industry requests to transfer operating licenses, which may involve changes in organizational structure affecting human performance as discussed further below.

12.1.3 Significant Regulatory Activities

NRC performs significant regulatory activities in the following seven areas to address human performance under the Human Factors Program:

(1) Human Factors Engineering Issues

(2) Emergency Operating Procedures and Plant Procedures

(3) Working Hours and Staffing

(4) Fitness-for-Duty

(5) Human Factors Information System

(6) Support to Event Investigations and For-Cause Inspections

(7) Training

The first six are described below, while Training is described under Article 11, “Financial and Human Resources.”

12.1.3.1 Human Factors Engineering Issues

This section discusses human factors activities related to engineering issues, covering the governing documents and process to carry out requirements, and experience and examples.

Governing Documents and Process

NRC staff evaluates the human factors engineering design of the main control room and control centers outside of the main control room using NUREG-0800, Rev. 1, Chapter 18, “Human Factors Engineering.” It also uses Revision 2 to NUREG-0700, “Human System Interface Design Review Guideline,” and Revision 2 to NUREG-0711, “Human Factors Engineering Program Review Model.” These documents provide guidance on human factors engineering to NRC staff for its reviews of submittals on human-system interface design related to licenses or design certification of nuclear installations, and reviews of the human-system interface that could be performed as part of an inspection. The staff also uses NUREG-1764, “Guidance for the Review of Changes to Human Actions,” to support its reviews of license amendment requests that credit the use of manual actions.

Experience and Examples

NRC reviews licensees' requests that involve aspects of human factors engineering. Examples include crediting manual actions in amendments to plant technical specifications, license transfers, and increasing the licensed reactor power level ("power uprates"). As an example, the staff recently evaluated requests for license amendments from Perry Nuclear Power Plant and Browns Ferry Units 1, 2, and 3. For Perry, the licensee proposed crediting manual actions as part of allowing the Inclined Fuel Transfer System to be open during certain, at power conditions. For Browns Ferry, the licensee requested crediting certain manual actions related to using the Standby Liquid Control System as part of its proposed application of Alternate Source Term methodology. NRC considered information in Information Notice 97-78, "Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times," in reviewing such requests. Supplemental information the staff considered in these reviews included NUREG-0711, "Human Factors Engineering Program Review Model," Rev. 1, and NUREG-1764, "Guidance for the Review of Human Actions, Draft Report for Comment," which was issued in 2002. Additionally, there have been recent trends in the U. S. nuclear industry to credit manual actions in place of certain physical separation barriers prescribed by regulation to protect redundant trains that are located in the same fire area from fire damage in order to maintain safe shutdown capability. As a result, the NRC has undertaken new rulemaking, to include standard acceptance criteria for assuring that manual actions credited by licensees are feasible and reliable. The rulemaking includes preparing guidance for NRC inspectors and an accompanying Regulatory Guide.

NRC has also evaluated a number of requests to transfer operating licenses, paying special attention to management and organization, staffing, and technical qualifications. The staff uses NUREG-0800, Standard Review Plan, Chapter 13, "Conduct of Operations," as principal guidance for these reviews.

NRC also approves requests for power uprates of currently licensed plants. For such requests, the staff examines the effect of the power uprate on plant procedures, controls, displays and alarms, and required operator actions.

12.1.3.2 Emergency Operating and Plant Procedures

Licensees must have programs for developing, implementing and maintaining such procedures. (Emergency preparedness is discussed in Article 16; the discussion in Article 12 is limited to the human factors aspect of emergency procedures.)

Governing Documents and Process

Generic Letter 82-33, "Requirements for Emergency Response Capability" (which transmitted Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability"), requires each licensee to submit a set of documents for developing emergency operating procedures.

NRC staff conducts for-cause inspections of plant procedures and emergency operating procedures using Inspection Procedures 42700, "Plant Procedures," and 42001, "Emergency Operating Procedures," respectively. The staff uses Inspection Procedure 42700 to focus inspections on identified procedural problems. In particular, this procedure guides inspectors on inspecting the usability of a licensee's procedures by assessing the degree to which accepted human factors principles have been incorporated into them. Inspection Procedure 42001 gives guidance for inspecting emergency operating procedures -- their development, implementation, revisions, and maintenance. Inspections include onsite human factors specialists and systems experts.

Experience and Examples

No significant examples applying to emergency operating and plant procedures were identified since 2001.

12.1.3.3 Working Hours and Shift Staffing

This section discusses activities related to working hours and shift staffing. It covers the governing documents and process to implement requirements and gives experience and examples.

Governing Documents and Process

Working Hours. NRC's policy on working hours is stated in the "Policy on Factors Causing Fatigue of Operating Personnel at Nuclear Reactors," which was issued on February 18, 1982, and revised in June 1982. The objective of the policy is to ensure, to the extent practicable, that personnel are not assigned to shift duties while in a fatigued condition that can significantly reduce their mental alertness or decisionmaking ability. The policy applies to "unit staff who perform safety-related functions (e.g., senior reactor operators, reactor operators, auxiliary operators, health physicists, and key maintenance personnel)," and it gives specific guidance to be used if unforeseen problems require substantial amounts of overtime. The policy also allows deviations from the guidelines "for very unusual circumstances," provided that (1) the plant manager, his deputy, or higher levels of management authorize the deviations, and (2) it would be "highly unlikely" that such deviations would cause significant reductions in the effectiveness of operating personnel. Issued in March 1983, Generic Letter 83-14, "Definition of Key Maintenance Personnel (Clarification of Generic Letter 82-12)," clarified the applicability of the policy to maintenance personnel.

In the months following the terrorist attacks on September 11, 2001, increased security demands at U.S. nuclear facilities caused a significant increase in the work hours of security personnel at U.S. nuclear facilities. In response to the concern about excessive fatigue of security personnel, NRC issued orders in April 2003 to all licensees of nuclear power plants. The orders require licensees to maintain the work hours of security force personnel below specified levels, monitor for individual fatigue, and establish a process to be followed if an individual declares that he or she is unfit for duty as a result of fatigue.

Shift Staffing. Paragraph (m) of 10 CFR 50.54, "Conditions of Licenses," specifies the minimum number of licensed operators that are required for nuclear power reactor sites. In addition NRC has other requirements with staffing implications. These include the personnel requirements for fire brigades and emergency response personnel contained in Appendix R, "Fire Protection Programs for Nuclear Power Facilities Operating Prior to January 1, 1979," and Appendix E, "Emergency Planning and Preparedness for Protection and Utilization Facilities," to 10 CFR Part 50, respectively.

In the September 2002, NRC began work on a process to evaluate exemption requests from 10 CFR 50.54(m) due to the changing demands and new technologies proposed by advanced reactor control room designs and significant light water reactor control room upgrades.
At present, the process for submitting an exemption request is included in a draft guidance

document that will be published for public comment in the near future. The justification for the recommended process is explained in NUREG/CR-6838, "Technical Basis for Assessing Exemptions from Nuclear Power Plant Licensed Operator Staffing Requirements 10 CFR 50.54(m)."

Experience and Examples

Working Hours. In June 2001, the staff summarized the staff's assessment of the policy and its implementation in SECY-01-0113, "Fatigue of Workers at Nuclear Power Plants." Citing weaknesses in NRC's regulatory framework concerning worker fatigue and inconsistency in the industry's implementation of the policy guidelines the staff proposed granting, in part, a 1999 petition requesting enforceable work hour requirements. In January 2002, the Commission approved the staff's rulemaking plan. From February 2002 thru August 2003 the staff held a series of public meetings to inform the staff's development of the proposed rule. The staff is currently in the process of developing the cost-benefit analysis for this rulemaking.

Shift Staffing. No significant examples applying to shift staffing were identified for 2000–2003.

12.1.3.4 Fitness-for-Duty

NRC first published a rule, entitled "Fitness-for-Duty Programs," in 1989. The rule required each licensee authorized to operate or construct a nuclear power reactor to implement a Fitness-for-Duty Program for all personnel having unescorted access to the protected area of its plant. The rule specified, as a performance objective, that licensees must provide reasonable assurance that nuclear power plant personnel perform their tasks in a reliable and trustworthy manner and are not under the influence of any substance, legal or illegal, or mentally or physically impaired from any cause. In addition, the rule required that licensee policy should also address factors that could affect fitness for duty such as mental stress, fatigue, and illness. Currently, NRC staff is preparing a comprehensive revision to the rule to update and enhance it.

Experience and Examples

NRC issues annual reports on statistical data and lessons learned by licensees from their fitness-for-duty program performance reports. The most recent of these is NRC Information Notice 2003-04, "Summary of Fitness-for-Duty Program Performance Reports for Calendar Year 2000," dated February 6, 2003. The NRC staff is currently completing a similar report that will provide data for the years since 2000.

12.1.3.5 Human Factors Information System

The Human Factors Information System is designed to be a convenient means of storing, retrieving, sorting, and analyzing human performance information. An automated management information system, it contains human performance information extracted from inspection reports, and licensee event reports. The system also contains information regarding the status of the accredited training programs at each site. The system can generate a variety of specialized reports, and has built-in system maintenance functions. Initiated in 1990, the system compiles human factors information that is not readily available from other NRC sources. Sources include NRC inspections and audits at plant sites. The agency maintains aWeb page (http://www.nrc.gov/reactors/operating/ops-experience/human-factors.html) to disseminate information on human performance issues at individual nuclear power plant sites.

The staff uses information from the system to gain insights about human performance and to monitor the frequency of human performance occurrences related to staffing, training, overtime, procedures, and human-system interface. The system is used in preparing for plant performance assessments. In such instances, human performance data are provided for the most recent 12-month period for each region. The total number of human performance factors contributing to each plant's licensee event report during the reporting period is compared to the national average. Written analyses of the data, including the types of personnel and the performance issues, are prepared for plants that meet or exceed twice the national average (a threshold value). In addition, the data for previous years are presented in tabular form for trending.

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

NRC staff with human factors technical expertise are included in special inspections; incident investigation team inspections; augmented inspection team inspections; event investigations; and projects, programs, and policy activities. NRC staff assesses management effectiveness, procedures, training issues, staffing issues, and human-machine interfaces. For training issues, inspectors use Inspection Procedure 41500, "Training and Qualification Effectiveness." For procedure issues, inspectors use Inspection Procedure 42001, "Emergency Operating Procedures," and Inspection Procedure 42700, "Plant Procedures." For baseline inspections under the Reactor Oversight Process, inspectors use Inspection Procedure 71152, "Identification and Resolution of Problems." This procedure is intended to establish confidence that each licensee is detecting and correcting problems in a manner that limits the risk to the public. A key premise of the Reactor Oversight Process is that weaknesses in licensees' problem identification and resolution programs will manifest themselves as performance issues that can be identified during the baseline inspection program or by crossing predetermined indicator thresholds.

12.2 Significant Research Activities

In addition to its regulatory activities, NRC researches human performance issues. This research has recently resulted in the publication of NUREG-1764, "Guidance for the Review of Changes to Human Actions," and revisions to NUREG-0711, "Human Factors Engineering Program Review Model" (Rev. 2) in 2004 and NUREG-0700, "Human-System Interface Design Review Guideline" (Rev. 2) in 2002.

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 explains quality assurance (QA) policy and requirements, and guidance for design and construction, operational activities, and staff licensing reviews. It also describes QA programs, including QA under the Reactor Oversight Process, augmented QA, and graded QA.

This section was updated to discuss adopting international QA standards and the staff-approved revision to the licensee/AmerGen Quality Assurance Topical Report. The section on graded QA was deleted. (Graded QA did not prove of interest to the U.S. nuclear industry.)

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 the health and safety of the public, and (2) the mitigation of adverse consequences of such accidents if they should occur. A primary means for achieving these objectives is by establishing and effectively implementing a nuclear QA program. Although a licensee may delegate aspects of the establishment or execution of the QA program to others, the licensee remains ultimately responsible for its overall effectiveness of the program. Licensees perform a variety of self-assessments to validate the effectiveness of their QA program implementation. NRC reviews descriptions of QA programs, and performs onsite inspections to verify aspects of the program implementation.

13.2 Regulatory Policy and Requirements

Each applicant for a construction permit for a nuclear power plant is required by paragraph (a)(7) of 10 CFR 50.34 “Contents of Applications; Technical Information,” to describe its QA program in its preliminary safety analysis report. This program applies to the design, fabrication, construction, and testing of safety-related plant equipment. Each applicant for a license to operate a nuclear power plant, is required by paragraph (b)(6)(ii) of 10 CFR 50.34 to provide a final safety analysis report that details its managerial and administrative controls to ensure safe operation. In both reports, the applicant must describe how it will satisfy the applicable requirements of 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.”

If a licensee wants to make changes in its QA program, it is required by paragraph (a)(3) of 10 CFR 50.54, “Conditions of Licenses,” to inform NRC of the changes. A licensee can make changes without prior NRC approval if the changes do not reduce the commitments in the program description as accepted by NRC. In April 1999, NRC revised 10 CFR 50.54(a) to define six categories of QA Program changes that are not “reductions in commitments.” Changes that do reduce commitments related to the QA Program must receive NRC approval before implementation.

Nuclear quality assurance criteria apply to all activities that affect the safety-related functions of structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. High-level criteria for determining which plant structures, systems, and components are safety related are provided in 10 CFR 50.2, “Definitions.” The definition is consistently applied in Appendix A to 10 CFR Part 100, “Seismic and Geologic Siting Criteria for Nuclear Power Plants,” 10 CFR 50.49, “Environmental Qualification of Electrical Equipment Important to Safety of Nuclear Power Plants,” and 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.” Based upon these criteria, licensees’ engineering organizations develop plant-specific listings of safety-related structures, systems, and components.

13.2.1 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants”

Appendix A states the general requirements for establishing QA controls. General Design Criterion I contains certain requirements applying to the QA of items important to safety. The scope of items that are “important to safety” includes a subset of plant equipment that is classified as “safety-related.” QA program requirements for safety-related structures, systems, and components are contained in Appendix B to 10 CFR Part 50 (discussed below). Safety-related structures, systems, and components are defined in 10 CFR 50.2, “Definitions.” QA program controls that are appropriate for some types of non-safety-related equipment are contained in other regulatory guidance.

13.2.2 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants”

Appendix B states the QA requirements that apply to activities that affect the safety-related functions of structures, systems, and components that prevent or mitigate the consequences of postulated accidents. The appendix defines quality assurance as all planned and systematic actions that are necessary to provide adequate confidence that structures, systems, and components will perform satisfactorily in service. Toward that end, Appendix B specifies 18 criteria that must be satisfied by the commitments in a licensee’s QA program. These criteria cover such aspects as organizational independence, design control, procurement, document control, test control, corrective action, and audits. Appendix B also stipulates that licensees establish measures to ensure that applicable regulatory requirements, design bases, and other requirements that are necessary to ensure adequate quality are suitably included or referenced in the documents for procurement of safety-related materials, equipment, and services whether purchased by the licensee or its contractors or subcontractors. Consistent with the importance and complexity of the products or services to be provided, licensees (or their designees) are responsible for periodically verifying that contractor’s QA programs comply, as appropriate, with the applicable criteria in Appendix B and that they are effectively implemented.

The requirements of Appendix B are written at a high level, such that it was necessary for NRC and the industry to develop consensus standards that give acceptable ways to conform to these requirements. NRC then issued companion regulatory guides, which endorsed (with conditions, if warranted) QA codes and standards.

13.2.3 Approaches for Adopting More Widely Accepted International Quality Standards

During the past 15 years, some suppliers of safety-related components have dropped their Appendix B programs to focus on larger commercial markets. Consequently, the number of suppliers from which licensees can procure safety-related parts and services has declined. Some licensees, in concert with EPRI, believe that in order to maintain a large supplier base in support of current operating nuclear plants, it may be necessary to evaluate the acceptability of procuring from suppliers with quality assurance programs other than Appendix B programs.

The staff reviewed options for adopting more widely accepted international quality standards like International Organization for Standardization (ISO) Standard 9001, considering how international standards compare with the existing Appendix B framework. The staff looked at quality standards, including widely adopted international standards such as ISO 9001-2000, “Quality Management System (QMS) — Requirements”; ASME NQA-1, “Quality Assurance Requirements for Nuclear Facility Applications”; and IAEA 50-C-QA, “Code on the Safety of Nuclear Power Plants: Quality Assurance.” The staff compared Appendix B requirements for quality assurance to the ISO 9001-2000 and interviewed suppliers having experience with both Appendix B and ISO quality programs. Finally, the staff met with industry representatives and discussed the feasibility of adopting international standards at a number of meetings.

Based on this review, the staff concluded that supplemental quality requirements would need to be applied when implementing ISO 9001 within the existing regulatory framework. The staff developed four potential approaches for licensee implementation of ISO 9001, two of which are considered more suitable for further development. These two were (1) licensees’ imposing specific controls for ISO 9001 certified suppliers during procurement and (2) using ISO 9001 certified suppliers for procuring commercial-grade items.

Experience and Examples

In early 2003, Virginia Electric Power Company, the licensee, replaced the North Anna, Unit 2, reactor vessel head with one manufactured to the French Nuclear Construction Code (RCC-M) following the reconciliation process of Section XI of the ASME Boiler and Pressure Vessel Code (Code). This effort was unique, representing a first-of-a-kind activity for the staff and for the U.S. commercial nuclear power industry. In overseeing this effort, the staff interacted with French and Belgian regulators and the reactor vessel head fabricator (Framatome ANP), conducted onsite inspections, and reviewed the licensee’s design and reconciliation documentation. The staff determined there were no regulatory barriers that precluded procurement of a replacement reactor vessel head manufactured to the RCC-M Code and believes other licensees, with licensing bases similar to North Anna, Unit 2, may adopt a similar process.

One of the more important technical issues identified by the staff involved the use of friction welds to join the flange and tube portions of the vessel head penetration nozzles. The staff determined that the use of a friction weld in the fabrication of these nozzles produced high quality, repeatable welds when the requirements of the RCC-M Code are followed and that such welds can be acceptable for use in commercial U.S. PWRs. On the basis of its oversight activities, the staff concluded that reactor vessel heads fabricated to RCC-M Code are of sufficient quality to ensure safety is maintained and that the replacement of the head enhanced the overall safe operation of North Anna, Unit 2.

13.3 QA Regulatory Guidance

NRC has QA guidance for design and construction, operational activities, and licensing.

13.3.1 Guidance for Design and Construction Activities

In 1971, the American National Standards Institute (ANSI) issued the consensus QA Standard N45.2, "Quality Assurance Program Requirements for Nuclear Power Plants." NRC endorsed ANSI N45.2 as providing a generally acceptable way to comply with the requirements of Appendix B to 10 CFR Part 50 during the design and construction phases of nuclear facilities. In later years, further ANSI standards promulgated in the N45.2 series, provided additional guidance on both programmatic aspects (such as records controls) and application-specific QA controls (such as those that apply to structures). NRC conditionally endorsed these ANSI standards through its regulatory guides.

In 1979, ASME published NQA-1, "Quality Assurance Requirements for Nuclear Facilities," which consolidated eight of the ANSI QA standards regarding the programmatic aspects of Appendix B. NRC endorsed the 1983 version of NQA-1 and the 1983-1a Addenda through Regulatory Guide 1.28, Rev. 3, "Quality Assurance Program Requirements (Design and Construction)." Licensees with approved QA programs could voluntarily choose to adapt their programs to NQA-1. Two licensees amended their QA programs to adopt NQA-1; the remainder chose to remain committed to the ANSI series standards to meet Appendix B requirements.

In December 2002, the staff approved a revision to the Exelon/AmerGen Quality Assurance Topical Report. This revision updated the licensee's commitment to quality standards by adopting the guidance in ASME NQA-1-1994. The staff concluded that adopting NQA-1-1994 was acceptable because the topical report continued to conform to the regulatory positions of Revision 3 to Regulatory Guide 1.28, as well as Revision 2 to Regulatory Guide 1.33, and because the licensee's common quality assurance program satisfies the requirements of Appendix B to 10 CFR Part 50 in accordance with the review criteria contained in Standard Review Plan 17.2.

13.3.2 Guidance for Operational Activities

In 1972, the American Nuclear Society issued N18.7, "Administrative Controls for Nuclear Power Plants," which focused on activities of importance for operating facilities, including reviews and audits, maintenance, tests, plant records management, and procedural controls. NRC endorsed N18.7-1972 and N45.2-1971, in combination, as providing an acceptable way to satisfy the requirements of Appendix B to 10 CFR Part 50 for an operating facility. The dual endorsement caused some confusion in the industry. ANSI N18.7-1976 was then developed to better integrate the QA provisions from the various ANSI standards into a more cohesive framework. Specifically, ANSI N18.7-1976 referenced pertinent standards from the N45.2 series, and also extracted information from them to provide guidance on how operational phase activities of a similar nature should be carried out. For example, for design control of plant modifications, N18.7 states that the provisions of N45.2.11 shall be used. NRC later conditionally endorsed the revised ANSI N18.7 standard in its Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operations)," as complying with the requirements of Appendix B.

13.3.3 Guidance for Staff Reviews for Licensing

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," guides the staff review of applications. The staff uses the plan to initially review QA program descriptions in licensees' preliminary and final safety analysis reports, as well as program revisions during facility operation. The Plan has specific review guidance correlated with the 18 criteria of Appendix B to 10 CFR Part 50, and integrates a review of licensee commitments to adopt NRC's QA-related regulatory guides and the industry's QA codes and standards.

13.4 QA Programs

NRC inspects licensees' QA under the Reactor Oversight Process and conducts augmented inspection programs.

In April 2000, NRC implemented its Reactor Oversight Process for operating reactors, of which baseline inspections are a component. (See Article 6.) Under the baseline inspection program, there is one primary procedure related to QA issues, which is known as Inspection Procedure 71152, "Identification and Resolution of Problems." Inspectors use this procedure to assess the effectiveness of licensees' programs to identify and resolve problems according to a performance-based review of specific issues. In particular, inspectors look for cases in which a licensee may have missed generic implications of specific problems, and for the risk significance of combinations of problems that individually may not have significance. They also verify that licensees are properly capturing issues that could affect the availability of equipment that is tracked under 10 CFR 50.65, "the Maintenance Rule," or by performance indicators. They do not inspect other aspects of QA program implementation in the baseline inspection program, but may, through supplemental inspections.

Some equipment in the nuclear facility may be classified as non-safety-related, and yet still be important to safety for some unique reason. In specific cases, NRC has specified that QA controls are warranted for equipment determined to be more important than commercial-grade equipment. However, the QA controls would not have to meet Appendix B requirements, which apply only to activities affecting safety-related functions. Typically, applying QA controls to this important-to-safety, yet non-safety-related, equipment is called "augmented quality control." For example, for equipment that applies to fire protection, Branch Technical Position CMEB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants," gives 10 quality program elements that should be applied to the design, procurement, installation, and testing of fire protection systems in safety-related areas of the plant.

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 continues to be in assurance with its design, applicable national safety requirements, and operational limits and conditions.

This section explains the governing documents and process for ensuring that systematic safety assessments are carried out during the life of the nuclear installation, including for the period of extended operation. It focuses on assessments performed to maintain the licensing basis of a nuclear installation. Finally, this section explains verification of the physical state and operation of the nuclear installation by analysis, surveillance, testing and inspection.

Other articles, for example, Articles 6, 10, 13, 18, and 19, also discuss activities undertaken to achieve nuclear safety at nuclear installations.

This section was updated to discuss the restart of Browns Ferry Unit 1 and the current status of license renewal. It also contains a new subsection on periodic safety reviews (PSRs).

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, which are reviewed and approved by NRC. These assessments and reviews are discussed in Article 18. This section focuses on the assessments that are 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. Further, renewal of a license is predicated on the requirement that a licensee will continue to meet its current licensing basis, and this section explains how this requirement is accounted for in license renewal.

14.1.1 Maintaining the Licensing Basis

NRC carries out regulatory programs to give reasonable assurance that plants continue to conform to the licensing basis. (Article 6 discusses these programs.) This section explains the governing documents and process used to maintain the licensing basis. The main governing documents are 10 CFR 50.90, “Application for Amendment of License or Construction Permit,” 10 CFR 50.59, “Changes, Tests, and Experiments,” and 10 CFR 50.71, “Requirements for Updating of Final Safety Analysis Reports.”

14.1.1.1 Governing Documents and Process

A licensee is to operate its facility in accordance with the license, and as described in its final safety analysis report. To change its license or reactor facility, a licensee must follow the review and approval processes established in the regulations. For license amendments, including changes to technical specifications, the licensee must request NRC approval in accordance with 10 CFR 50.90, “Application for Amendment of License or Construction Permit.” However, licensees can make certain changes without prior NRC approval if they perform specified reviews and meet certain conditions. Such changes are provided for in 10 CFR 50.59, “Changes, Tests, and Experiments,” as described below.

10 CFR 50.59, “Changes, Tests, and Experiments.” 10 CFR 50.59 describes the circumstances under which a licensee may make changes to its facility or procedures as described in its final safety analysis report and may conduct tests and experiments that are not described in its report without prior NRC approval. Licensees are required to periodically submit information about changes made in accordance with 10 CFR 50.59. NRC monitors each licensee’s processes for implementing the requirements in 10 CFR 50.59.

NRC approval is required in circumstances when the change, test, or experiment would require a change to the Technical Specifications or if any of eight evaluation criteria in the rule are met. These criteria pertain to important aspects of the plant design and analysis, such as the likelihood or consequences of accidents or of malfunctions of equipment. The rule requires review if the change, test or experiment would result in more than a minimal increase in any of these characteristics. Criteria are also included for requiring review if a change would create a new type of accident, or a malfunction with a different result from those previously evaluated. Two other criteria are also included: one for design basis limits for the fission product barriers (e.g., parameters for fuel or containment) and one on the methods of evaluation used in the safety analyses. The licensee must apply to amend the license pursuant to 10 CFR 50.90 under such circumstances. NRC performs and documents a safety evaluation in these instances before it authorizes the change.

10 CFR 50.71, “Requirements for Updating of Final Safety Analysis Reports.” Another process for making changes is set forth in paragraph (e) of 10 CFR 50.71, “Requirements for Updating of Final Safety Analysis Reports,” which requires licensees to update their final safety analysis reports periodically to incorporate the information and analyses that they submitted to the Commission or prepared pursuant to Commission requirements. Revisions to the updated final safety analysis reports are to include the effects of changes that occur in the vicinity of the plant, changes made in the facility or procedures described in the report, safety evaluations for approved license amendments and for changes made under 10 CFR 50.59, and safety analyses conducted at the request of the Commission to address new safety issues.

14.1.1.2 Regulatory Framework for the Restart of Browns Ferry Unit 1

The regulatory framework for the restart of Browns Ferry Unit 1 consists of two major elements: inspection and licensing activities. Inspection is performed in accordance with a dedicated Inspection Manual Chapter and the licensing activities are conducted in accordance with a Regulatory Framework letter as discussed below. The following paragraph provides background information for the Browns Ferry site.

The Browns Ferry site, located near Decatur, Alabama, has three essentially identical boiling water reactors (GE, BWR-4, Mark-1 Containment). After resolving management and regulatory issues that caused all three units to shut down in 1985, TVA successfully restarted Units 2 and 3 in the 1990s. In May of 2002, TVA decided to initiate a restart effort for Unit 1, planned for completion in 2007 (a 5-year program). TVA has kept Unit 1 in a de-fueled layup condition since 1985. TVA is implementing programs for Unit 1 that are similar to those used to restart Units 2 and 3, incorporating improvements and lessons learned, and dedicating resources, including personnel with experience on restarting Units 2 and 3. Restarting Unit 1 differs from the restarting Units 2 and 3 in that TVA is applying for license renewal and an extended power uprate in parallel.

As of May 2004, Browns Ferry Unit 1 restart project is on schedule. TVA is minimizing the impacts of the restart project on the operating units, and is completing major activities, including modifying, installing, and welding piping, installing cable trays and supports, and re-tubing condensers.

Regarding regulatory activities, NRC issued Inspection Manual Chapter 2509, "Browns Ferry Unit 1 Restart Project Inspection Program," in August 2003. This chapter provides the policies and requirements for the restart inspection program, establishes a Restart Oversight Panel in the final year of the project, and establishes documentation expectations for the major regulatory and licensee actions associated with the restart. This chapter also provides guidance for assessing the licensee's readiness for restart and for the eventual return of the plant to the Reactor Oversight Process. In addition, on August 14, 2003, the staff issued a Regulatory Framework letter which identified the licensing issues that require resolution before restart. Some of these issues are generic communications (i.e., bulletins, generic letters, and Three-Mile Island Action Items), 27 special programs (e.g., environmental qualification of electrical equipment, individual plant examination, long-term torus integrity program), and numerous technical specifications changes (e.g., power range neutron monitor upgrade, oscillation power range monitor, and 24-month fuel cycle).

TVA will gradually transition the unit into the Reactor Oversight Process as it verifies each cornerstone or inspectable area to be ready for monitoring. NRC expects that Emergency Preparedness, Security, Radiation Protection, and the cross-cutting area of problem identification and resolution will be ready for transition in Fiscal Year 2004. Final transition of Unit 1 to the Reactor Oversight Process, wherein its Action Matrix will determine NRC response, will not occur until after startup when all cornerstones can be monitored using baseline inspection and performance indicators.

As for inspection activities, NRC's Office in Region II (Atlanta, Georgia) is performing inspections and oversight activities. Inspection activities to date have included health physics, safe-end welding, drywell steel platforms, large-bore pipe supports, and cable trays and supports. The first dedicated Unit 1 quarterly inspection report was issued on August 15, 2003.

Upon restarting Unit 1, the three Browns Ferry units will be essentially identical in design and licensing basis. To accomplish this, TVA will submit a significant quantity and variety of licensing actions. The Regulatory Framework letter provides a detailed listing of generic communications and other licensing actions that need to be addressed by NRC's Office of Nuclear Reactor Regulation (NRR) and followed-up by Region II inspections. To complement the dedicated staffing by Region II, NRR has recently assigned a dedicated project manager for Unit 1 restart. To facilitate communications with key stakeholders, NRC has held periodic public meetings at the site, and has developed a public outreach Web page which is similar to the Reactor Oversight Process Web page. The Web page address is http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/bf1-recovery.html

The primary challenges for the Unit 1 restart are considered to be the parallel pursuit of license renewal and the extended power uprate, and the length of time that the unit has spent in layup (18 years as of 2003). TVA had submitted an application for license renewal for all three units in December 2003. TVA is developing a restart test program similar to that for the Unit 3 restart. NRC staff will be carefully evaluating the restart test program for Unit 1, focusing particularly on the plans for power ascension. For both restart and license renewal, it is possible that specific challenges may arise owing to the length of time the unit has spent in layup.

14.1.2 License Renewal

This section explains license renewal. It covers the governing documents and regulatory process, recent experience and examples.

14.1.2.1 Governing Documents and Process

Background: The U.S. Atomic Energy Act and NRC regulations limit commercial power reactor licenses to 40 years, but also permit such licenses to be renewed. The original 40-year term was selected on the basis of economic and antitrust considerations, not technical limitations.

NRC has established a license renewal process that can be completed in a reasonable period of time with clear requirements to ensure safe plant operation for up to an additional 20 years of plant life. NRC's current schedule is to complete renewal reviews within 30 months of receipt if a hearing is conducted, and within 22 months if not. The decision whether to seek license renewal rests entirely with nuclear power plant owners, and typically is based on the plant's economic situation and whether it can meet NRC requirements.

Research results have concluded that aging phenomena are readily manageable and do not pose technical issues that would preclude life extension for nuclear power plants. It was also found that many aging effects are dealt with adequately during the initial license period and credit should be given for these existing programs, particularly those under NRC's Maintenance Rule (10 CFR 50.65), which helps manage plant aging.

The license renewal process proceeds along two tracks — one for review of safety issues and another for environmental issues. An applicant must provide NRC with an evaluation that addresses the technical aspects of plant aging and describes the ways those effects will be managed. It must also prepare an evaluation of the potential impact on the environment if the plant operates for another 20 years. NRC reviews the application and verifies the safety evaluations through inspections.

Public participation is an important part of the license renewal process. There are opportunities for members of the public to question how aging will be managed during the period of extended operation and all information related to the review and approval of a renewal application is made available to the public. Significant concerns may also be litigated in an adjudicatory hearing if any party who would be adversely affected requests a hearing.

10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."
10 CFR Part 54, known as the "License Renewal Rule," establishes the technical and procedural requirements for renewing operating licenses. License renewal requirements for power reactors are based on two key principles:

(1) The regulatory process, which assesses and verifies safety, continued into the extended period of operation, is adequate to ensure that the licensing basis of all currently operating plants provides an acceptable level of safety. The possible exception is detrimental effects of aging on certain systems, structures, and components, and possibly a few other issues applying to safety only during the period of extended operation, and

(2) Each plant's licensing basis is required to be maintained throughout the renewal term.

The foundation of license renewal rests on the determination that currently operating plants continue to maintain an adequate level of safety. Over the plant's life, this level has been enhanced by maintaining the licensing basis, properly adjusted to incorporate new information that is derived from operating experience. Moreover, NRC activities have continually ensured that the licensing basis will continue to provide an acceptable level of safety.

Guidance that applies to license renewal includes Regulatory Guide 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," to guide applicants in applying to renew a license and the "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," NUREG-1800, to guide the staff in reviewing applications. The standard review plan incorporates by reference the "Generic Aging Lessons Learned (GALL) Report," NUREG-1801, which generically documents the basis for determining when existing programs are adequate, and when they should be augmented for license renewal. As lessons are learned from the review of renewal applications or generic technical issues are resolved, improved guidance is issued in the interim for use by applicants until the guidance is incorporated into the next formal update of the guidance documents.

10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions." NRC's environmental protection regulation, 10 CFR Part 51, also applies to license renewal and this regulation was amended to facilitate the agency's environmental review process for license renewal. Specifically, the review requirements for 10 CFR Part 51 are founded on the conclusion that certain environmental issues can be resolved generically and need not be evaluated in each plant-specific application. These issues are described in NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants." NRC performs plant-specific reviews of the environmental impacts of license renewal to determine whether the impacts are so great that they should preclude license renewal as an option for energy-planning decision makers.

Guidance to applicants preparing environmental reports for license renewal is provided in Supplement 1 to NRC Regulatory Guide 4.2, "Preparation of Supplemental Environmental Reports for Applications to Renew Nuclear Power Plant Operating Licenses." The "Standard Review Plan for Environmental Reviews for Nuclear Power Plants, Operating License Renewal," NUREG-1555, Supplement 1, guides NRC staff's review of the environmental issues associated with a renewal application.

14.1.2.2 Experience

NRC issued the first renewed licenses for the Calvert Cliffs Nuclear Power Plant and the Oconee Nuclear Station in 2000. As of April 2004, NRC renewed licenses for 25 reactors at 14 sites, and is currently reviewing applications to renew the licenses of an additional 17 reactors at 9 sites. If it approves all of these applications, it will have extended the operating licenses of about

40 percent of U.S. plants. On the basis of industry statements, NRC expects that essentially all plants will apply for license renewal.

14.1.3 The United States and Periodic Safety Reviews

This section on periodic safety reviews (PSRs) has been added to this report as a result of widespread interest in this topic at the Convention's Second Review Meeting.

This section explains how the U.S. regulatory approach provides a continuum of assessment and review that ensures public health and safety throughout the period of plant operation. As discussed below, plant safety is improved by a combination of the ongoing NRC regulatory process, oversight of the current licensing basis, backfitting, broad-based evaluations, license renewal, and licensee initiatives that go beyond the regulations.

NRC's Robust and Ongoing Regulatory Process and The Current Licensing Basis

Before issuing an operating license, NRC comprehensively determines that the design, construction, and proposed operation of the nuclear power plant satisfy NRC's requirements and reasonably ensures the adequate protection of the 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 continuing regulatory activities of NRC, as well as the activities of the licensee.

NRC engages in a large number of regulatory activities which, when considered together, constitute a regulatory process that provides ongoing assurance that the licensing bases of nuclear power plants provide an acceptable level of safety. This process includes regulatory research, inspections (both periodic regional inspections as well as daily oversight by the resident inspector), audits, investigations, evaluations of operating experience, and regulatory actions to resolve identified issues. NRC's activities may result in changes to the licensing basis for nuclear power plants through promulgation of new or revised regulations, acceptance of licensee commitments for the modification to nuclear power plant designs and procedures, and the issuance of orders or confirmatory action letters. NRC also publishes the results of operating experience analysis, research, or other appropriate analyses through generic communication documents such as bulletins and generic letters. Licensee commitments in response to these documents also change the plant's licensing basis. In this way, NRC's consideration of new information provides ongoing assurance that the licensing basis for the design and operation of all nuclear power plants provides an acceptable level of safety. This process continues for plants that receive a renewed license to operate for 20 years beyond the original operating license.

In addition to NRC-required changes in the licensing basis, a licensee may also voluntarily seek changes to the current licensing basis for its plant. However, these changes are subject to NRC's formal regulatory controls with respect to the changes (such as 10 CFR 50.54, 50.59, 50.90, and 50.92). These regulatory controls ensure a documented basis for licensee-initiated changes to the licensing basis for a plant and that the licensee obtains NRC review and approval prior to implementation of changes to the licensing basis that meet the review thresholds in 10 CFR 50.59. Any changes or modifications to the licensing basis that a licensee makes without prior NRC review must be reported to NRC at least every two years. Region-based NRC inspectors perform a sampling inspection of those changes in accordance with the Reactor Oversight Process to help ensure that the changes or modifications were properly characterized.

The Backfitting Process: Timely Imposition of New Requirements

NRC recognized the need to systematically consider new requirements rather than depend on the license renewal process or other regulatory processes to decide on plant upgrades. In the late 1970s and early 1980s, NRC recognized the need for a process to determine when to address generic issues for all plants. As a result, NRC developed the "backfitting" process and initiated the Committee To Review Generic Requirements (CRGR) to review staff-proposed backfits on licensees.

The Backfit Rule (10 CFR 50.109) applies to both generic and plant-specific backfits for power reactors. It defines a backfit as any modification of or addition to plant systems, structures, components, procedures, organizations, design approvals, or manufacturing licenses that may result from the imposition of a new or amended rule or regulatory staff position. Except for backfits that are imposed to bring a licensee back into compliance with its license or to ensure adequate protection of the public health and safety or common defense and security a cost-benefit backfit analysis is required. The NRC must determine though a backfit analysis that the proposed backfit will provide a substantial increase in overall protection of the public health and safety (or common defense and security) and that the direct and indirect costs for the facility are justified in view of the increased protection.

Compliance and adequate protection backfits are justified differently. A documented evaluation is required, which provides the basis and states the objectives and purpose of the proposed backfit.

In 1988, NRC issued an amended Backfit Rule, which clearly states that economic costs will not be considered in cases of ensuring, defining, or redefining adequate protection of the public health and safety, or in cases of ensuring compliance with NRC requirements or written licensee commitments.

Backfitting is expected to occur and is an inherent part of the regulatory process. However, it is to be done only after a formal, systematic review to ensure that changes are properly justified and suitably defined. The requirements of this process are intended to ensure order, discipline, and predictability and to enhance optimal use of NRC staff and licensee resources.

The controls on generic backfitting include review by the CRGR. Established in November 1981, this committee is made up of senior managers from various NRC offices. The CRGR's objectives include eliminating unnecessary burdens on licensees, reducing radiation exposure to workers while implementing requirements, and optimizing use of NRC and licensee resources to ensure safe operation. Following its review of a proposed generic communication, the CRGR recommends approval, revision, or disapproval to NRC's Executive Director for Operations. If the office proposing the communication does not agree with the CRGR recommendation it may refer the issue to the Executive Director for Operations for decision. The CRGR operates under a charter that specifically identifies the documents to be reviewed and the analyses, justifications, and findings to be provided. Thus, although the primary responsibility for proper backfit considerations belongs to the initiating organization, the CRGR charter is a key implementing procedure for generic backfitting.

NRC's Extensive Experience with Broad-Based Evaluations

In the mid 1970s, NRC recognized the importance of assessing the adequacy of the design and operation of currently licensed nuclear power plants, understanding the safety significance of deviations from applicable current safety standards that may have been approved after those plants were licensed, and providing the capability to make integrated and balanced decisions about whether backfit modifications were required at those plants.

Consequently, in 1977, NRC initiated the Systematic Evaluation Program (SEP). From a list of approximately 800 potential issues and topics related to nuclear safety, the SEP found that the regulatory requirements for 137 issues had changed sufficiently to warrant evaluation. The staff then compared the designs of 10 of the older plants to the licensing criteria delineated in the then recently issued Standard Review Plan[f]. After further review, the staff determined that 27 issues required some corrective action at one or more plants and resolution of those issues could lead to safety improvements for other operating plants built at about the same time. These 27 issues became known as the 27 "SEP lessons learned."

In 1984, NRC staff presented the 27 SEP lessons learned to the Commission as part of a proposal for an Integrated Safety Assessment Program (ISAP). The staff developed this program to review safety issues for a specific plant in an integrated manner instead of continuing the SEP at other older operating reactors. In November 1984, the Commission published the "Commission Policy Statement on the Systematic Evaluation of Operating Nuclear Power Reactors." In this policy statement, the Commission articulated its view that issues relating to the safety of operating nuclear power plants can be more effectively and efficiently implemented in an integrated, plant-specific review. Probabilistic safety analysis was discussed, for the first time, as a method to obtain consistent and comparable results which could be used to enhance a safety assessment. The SEP process was transformed into the ISAP pilot program.

In May 1985, NRC initiated the ISAP pilot program at two plants, Millstone Unit 1 and Haddam Neck (Connecticut Yankee). The ISAP pilot program identified some benefits, however, the Commission deferred extending it beyond the pilot phase until the staff provided an integrated package of options that clarified the relationship between the proposed follow-on program to the ISAP pilot, (ISAP II), and the newly proposed individual plant examination (IPE) process. (See Section 10.3 for an explanation of the IPE process.)

The Commission determined that since the ISAP II program would be voluntary and the IPE program, through NRC's generic letter process, would require a licensee response, priority should be given to the IPE program. Many of the same benefits that may have been derived through the proposed ISAP II were derived instead through the IPE (e.g., probabilistic safety analysis) process.

In the late 1980's and throughout the 1990's, NRC continued its efforts to strengthen its regulatory infrastructure and ensure continued safe operation of commercial nuclear power plants through inspection, broad-based assessment, and where appropriate, establishment of new generic requirements. For example, the Commission determined that licensees should assess the accessibility and adequacy of their design bases information and determine whether a design basis reconstitution program was needed for their plant. The Commission captured its expectations in a Commission Policy Statement on Availability and Adequacy of Design Bases Information at Nuclear Power Plants. The Commission also expanded the IPE program to consider external events and, recognizing the relationship between maintenance, equipment

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[f] Standard Review Plans help ensure the quality and uniformity of staff reviews and provide a well-defined base from which to evaluate a licensee or applicant submittal. The Standard Review Plans are also intended to make information about regulatory matters widely available, to enhance communication with interested members of the public and the nuclear power industry, and to improve their understanding of the staff review process.

reliability, plant risk and safety, the Commission promulgated the Maintenance Rule, 10 CFR 50.65.

License Renewal Confirms Safety of Plants

As late as 1991, the 27 SEP lessons learned from the SEP had not been definitively resolved at some plants. As the staff considered a process to renew the operating licenses for the operating nuclear power plants, it assessed how to address these 27 issues.

Four of the 27 issues had been completely resolved for all plants. One issue was of such low safety significance that it required no additional action. The staff determined that none of the remaining 22 issues required immediate action to protect public health and safety. These 22 issues were placed into the established regulatory process for determining the safety significance of generic issues[g].

In developing the License Renewal Rule (10 CFR Part 54), the Commission concluded that issues material to the renewal of a nuclear power plant operating license are to be limited to those issues that the Commission determines are uniquely relevant to protecting the public health and safety and preserving common defense and security during the period of extended operation. Other issues would, by definition, have a relevance to the safety and security of the public during current plant operation. Given the Commission's ongoing obligation to oversee the safety and security of operating reactors, issues related to current plant operation would be addressed by the existing regulatory process within the present 40-year license term rather than deferred until the time of license renewal. (See Section 14.1.2.1 for a description of license renewal.) To add to that discussion, license renewal applicants are required to complete an integrated plant assessment (IPA)[h] and evaluate time-limited aging analyses. An additional benefit of the license renewal program is that licensees applying for license renewal are willing to spend significant resources on equipment upgrades which improve the overall safety of the plant.

Risk-Informed Regulation and the Reactor Oversight Process

NRC is actively increasing the use of risk insights and information in its regulatory decision-making. In the reactor area, risk-informed activities occur in the five broad categories of (1) applicable regulations; (2) licensing process; (3) revised oversight process; (4) regulatory guidance; and (5) risk analysis tools, methods, and data. Activities within these categories include revisions to technical requirements in the regulations; risk-informed technical

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[g] NRC documents and tracks resolution of these "generic safety issues." The generic safety issue program provides for (1) identifying generic issues; (2) assigning them priorities; (3) developing detailed action plans for their resolution; (4) overseeing resolution progress by senior managers; and (5) disseminating to the public the status of resolution progress. The resolution of these issues may involve new or revised rules, new or revised guidance, or revised interpretation of rules or guidance that affect nuclear power plant licensees or nuclear material certificate holders. Congress requires that NRC maintain this program.

[h] An Integrated Pant Assessment identifies and lists structures and components subject to an aging management review (AMR). These include "passive" structures and components that perform their intended function without moving parts or without a change in configuration or properties. These include such components as the reactor vessel, the steam generators, piping, component supports, seismic Category I structures, etc. 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.

specifications; a new framework for inspection, assessment, and enforcement actions; guidance on risk-informed inservice inspections; and improved standardized plant analysis risk models.

In 2000, NRC implemented a revised Reactor Oversight Process (see discussion under Article 6) using risk insights and lessons learned from more than 40 years of regulating nuclear power plants. The previous oversight process evolved over a period of time when the nuclear power industry was less mature and there was much less operational experience on which to base rules and regulations. Very conservative judgments governed the rules and regulations. Significant plant operating events occurred with some frequency and the oversight process tended to be reactive and prescriptive, closely observing plant performance for adherence to the regulations, and responding to operational problems as they occurred.

After nearly 4 decades of operational experience and, generally speaking, steadily improving plant performance, the Reactor Oversight Process now focuses more of the agency's resources on the relatively small number of plants which evidence performance problems. The Reactor Oversight Process is more effective in correcting performance or equipment problems today because the agency's response to problems is more timely and predictable.

The Reactor Oversight Process makes greater use of objective performance indicators. Together, the indicators and inspection findings provide the information needed to support reviews of plant performance, which are conducted on a quarterly basis. In addition, the Reactor Oversight Process features expanded reviews on a semiannual basis to include inspection planning and a performance report, (all of which are posted on NRC's public Web site.)

Licensee Responsibilities for Safety: Regulations and Initiatives Above Regulations

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

Some of these reviews are not specifically mandated by NRC regulations. Rather, they are self-imposed initiatives over and above regulations, motivated by their self-described pursuit of excellence and by the recognition that, in the U.S. free-market competitive energy industry, safety and economics are directly linked. Licensees have, for example, voluntarily replaced analog instrumentation and control systems with digital instrumentation and control systems, upgraded their plants to increase production of electricity, and managed their plants to performance levels above NRC's performance indicator thresholds.

Under the U.S. regulatory structure, 10 CFR Part 50 Appendix B requires that all nuclear power plant licensees maintain a quality assurance program. Quality assurance comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component 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 the quality of the material, structure, component, or system to predetermined requirements.

Licensees carry out a comprehensive system of planned and periodic audits to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program. The audits are performed in accordance with the written procedures or check lists by appropriately trained personnel not having direct responsibilities in the areas being audited.

Audit results are documented and reviewed by management having responsibility in the area audited and appropriate follow-up initiated.

Summary

The IAEA and the Western European Nuclear Regulators' Association (WENRA) have developed guidance[i] and objectives for conducting PSRs that share broad commonality. Specifically, consistent with the guidance of both organizations, PSRs are comprehensive assessments with the following purposes:

(i) Determine, at the time of the review, whether the plant complies with its licensing basis.
(ii) Identify the extent to which the current licensing basis remains valid, in-part, by determining the extent to which the plant meets current safety standards and practices.
(iii) Provide a basis for implementing appropriate safety improvements, corrective actions, or process improvements.
(iv) Provide confidence that the plant can continue to be operated safely.

For the reasons discussed above and summarized below, the shared objectives associated with the IAEA and WENRA PSR guidance are substantively accomplished in the United States on an ongoing basis.

First, NRC's regulatory process provides a robust foundation for ongoing assessments, evaluations, and when appropriate, imposition of new requirements. NRC and the U.S. nuclear industry consider new information, in a more risk-informed manner, as it becomes available; adjust the regulatory oversight and plant safety priority, respectively; and provide ongoing assurance that the licensing basis for the design and operation of all nuclear power plants provides an acceptable level of safety.

Second, NRC and the U.S. nuclear industry have a 30-year history of implementing broad-based plant assessments. The regulatory history of implementing broad-based assessments is a direct result of adaptive, probing, and independent regulatory process. These assessments have included the SEP, the ISAP, the IPE, and the reactor license renewal process and provide additional confidence that plant safety continues to be the highest priority and that NRC and industry continue to pursue enhancements that improve safety. The time line in Figure 4 demonstrates that, over a period of almost 25 years, broad-based NRC assessments and regulatory initiatives have provided a continuum of assessment, improvement, and oversight, which ensures that licensed plants continue to operate safely.

NRC's approach for continuing to ensure plant safety differs from the historically deterministic focus of PSRs. The transition to a more risk-informed regulatory framework and the Reactor Oversight Process provides an ongoing approach and basis for implementing appropriate safety improvements, corrective actions, or process improvements and provides confidence that the plant can continue to be operated safely. The NRC's more risk-informed approach helps ensure resources are most effectively and efficiently focused on those issues most important to safety.

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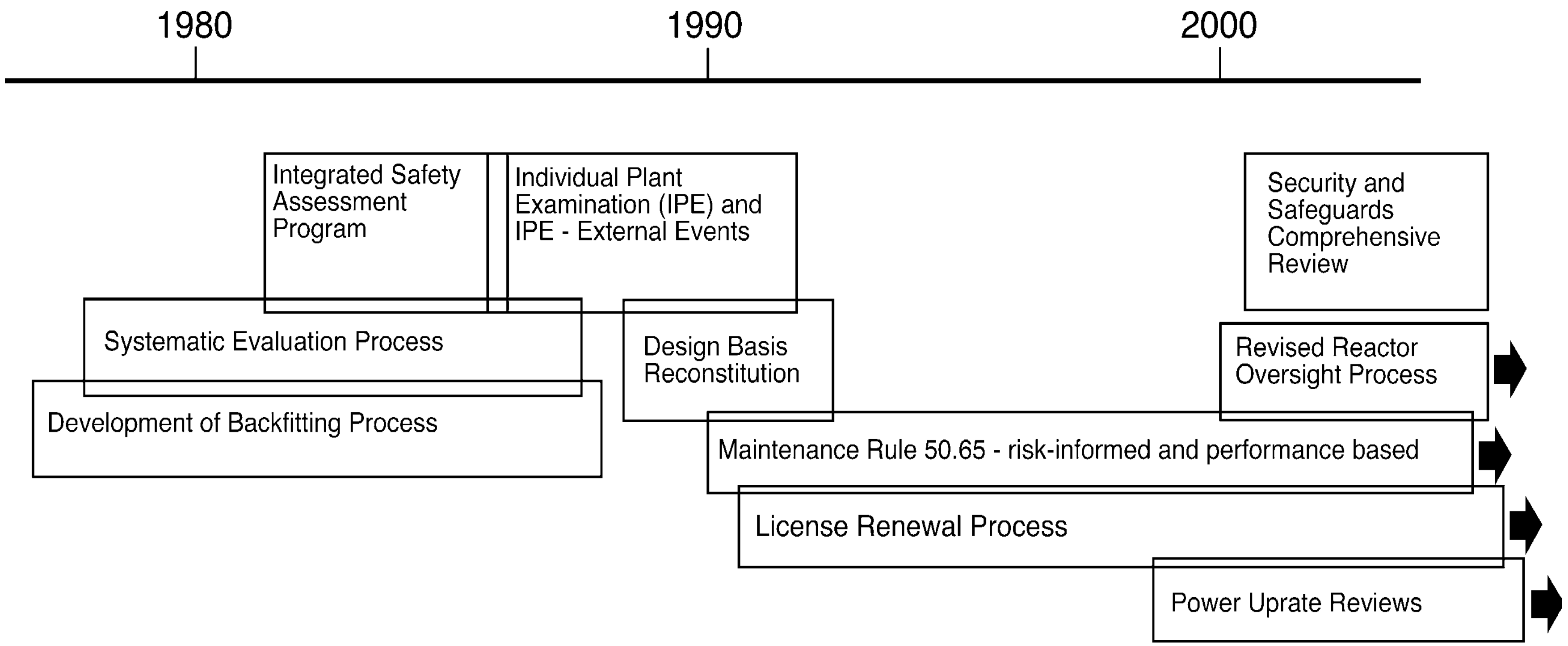
[i] IAEA Guidance: Safety Standards Series No. NS-G-2.10, Periodic Safety Review Of Nuclear Power Plants Safety Guide International Atomic Energy Agency Vienna, 2003. WENRA Guidance: Pilot Study on Harmonization of Reactor Safety in WENRA Countries, WENRA Working Group on Reactor Harmonization, March 2003.

Finally, U.S. licensees establish performance expectations above the thresholds required by NRC. These self-imposed expectations and initiatives — over and above the regulations — result from the licensee's self-described motivation to pursue excellence and by the recognition that, in the free-market competitive industry in the United States, safety and economics are directly linked.

Timeline of Significant U.S. Regulatory Broad Based Reviews



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. Requirements specifying verification are contained in the Technical Specifications (for surveillance) and national consensus codes (for testing and periodic inspections).

10 CFR 50.55a, “Codes and Standards,” defines the requirements for applying industry codes and standards to nuclear power reactors during design, construction, and operation. This regulation applies to both operating licenses and construction permits. 10 CFR 50.55a states, “Systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the ASME Boiler and Pressure Vessel Code specified in paragraphs (b) though (g) of this section.” The ASME Code has requirements for the construction and periodic inspection of boilers, pressure vessels, and nuclear components. These requirements apply to materials, design, fabrication, testing, inspection, and stamping. 10 CFR 50.55a also provides for alternatives to the ASME Code when authorized by NRC.

Through analysis, surveillance, testing, and inspection, NRC verifies 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 previously discussed in Article 6, NRC’s Reactor Oversight Process includes inspections to verify that licensees are fulfilling their obligations to conduct such surveillances and testing and take corrective action. NRC staff updates, revises, and improves existing regulatory programs in light of operating experience and significant new safety information. These activities are discussed in Article 19.

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 is reasonably achievable, and that no individual shall be exposed to radiation doses that exceed the prescribed national dose limits.

This section summarizes the authorities and principles of radiation protection, which include the regulatory framework, regulations, and radiation protection programs for controlling radiation exposure for occupational workers and members of the public. Article 17 addresses radiological assessments that apply to licensing and to facility changes.

The only significant change in this section is an updating of doses.

15.1 Authorities and Principles

Generally, U.S. radiation control measures are founded on radiological risk assessments by the United Nations Scientific Committee on the Effects of Atomic Radiation and the U.S. National Academy of Sciences Committee on the Biological Effects of Ionizing Radiation. These risk assessments are reflected in the risk management recommendations promulgated by the International Commission on Radiological Protection (ICRP) and the National Council on Radiation Protection and Measurements (NCRP). On the basis of these assessments and recommendations, the EPA develops “generally applicable radiation standards” for use by the other Federal agencies, including NRC. Considering these recommendations and standards, the responsible agencies, such as NRC, then establish regulations.

The principles upon which the U.S. radiation protection programs are based are generally consistent with the principles espoused by the ICRP. That is to say, (1) it is known that large doses of ionizing radiation can be deleterious to human health, and (2) it is considered prudent to assume that small doses may also be harmful, with the probability of a deleterious effect being proportional to the dose. The ICRP-recommended protection principles of “limitation,” “justification,” and “optimization” are acknowledged, but are proving difficult to carry out.

Of these, “limitation” is most practicable. Dose limits are established in the regulations, and these limits cannot be exceeded without violating the regulations. There is a lengthy history of the doses being kept within the limits for workers (NUREG-0713, Vol. 24, 2003) and members of the public living near nuclear power plants (NUREG/CR-2850, Vol. 14, 1996).

“Justification,” the recommendation that any activity involving radiation exposure should be shown to be beneficial before the activity is undertaken, has proved impracticable. The difficulty results from three basic facts. Specifically, (1) every human activity involves radiation exposure, (2) the outcome of a new activity can never be determined in advance, and (3) the U.S. Government (like other governments) lacks this degree of control over the activities of its citizens. Thus, the "justification" activities in the United States are generally limited to cost/benefit studies and analyses of the environmental impact of major actions, such as imposing a new regulation or building a new nuclear power plant.

Rather than “optimization,” the United States has used the expression “as low as is reasonably achievable” (ALARA), although the two concepts are consistent. As a guiding principle, ALARA (with varying terminology) dates back to 1939 (at least in the United States) and is defined in the regulations for occupational workers and members of the public.

For decades, the ALARA criterion for occupational radiation exposure has been addressed in 10 CFR Part 20, "Standards for Protection Against Radiation," but as an admonition rather than a requirement. In 1994, the regulation was changed to require that all licensees develop, document, and carry out an ALARA Program. Compliance with this requirement was to be judged on the basis of a licensee's capability to track and, if necessary, reduce exposures, and not on whether exposures and doses represented an absolute minimum or whether the licensee had used all possible methods to reduce exposures.

For control of radiation exposure to members of the public, NRC modified 10 CFR Part 50 by adding Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion As Low As Is Reasonably Achievable for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents." Issued in 1975, this appendix required that radioactive releases from nuclear power plants be kept ALARA. This requirement led to the establishment of numerical criteria [i.e., 0.00005 Sv (0.005 rem) in a year to the most highly exposed individual]. This NRC requirement was soon followed by similar EPA requirements for other facilities. It is not clear that these requirements satisfy the ICRP's intent, but they are sufficient to keep public doses well below the local variation in doses from natural sources.

Although U.S. regulations are generally consistent with ICRP recommendations, to date, certain constraints have limited the extent to which the U.S. regulations coincide with the ICRP recommendations. One important constraint has been the desire for regulatory stability. Revising the regulations to incorporate every new ICRP position would impose a serious burden on the licensees without a commensurate benefit. Furthermore, for nuclear power reactors, new requirements are constrained by the Backfit Rule (10 CFR 50.109), which essentially requires that any increase in regulatory requirements be justified by a commensurate improvement in safety. Consequently, U.S. regulations were founded on older (rather than the most recent) recommendations of the ICRP. Nevertheless, the Commission has directed NRC staff to work closely with the ICRP and other national and international organizations to assist in developing the 2005 ICRP recommendations. Subsequently, NRC may revise its regulations, in whole or in part, depending on the nature of these recommendations.

15.2 Regulatory Framework

Requirements for radiation protection were developed to implement laws passed by Congress. These laws are the Atomic Energy Act of 1954, the Energy Reorganization Act of 1974, and the Uranium Mill Tailings Radiation Control Act of 1978.

The direct controls over licensees are established through NRC regulations. Various documents provide additional guidance and clarification. Specifically, these documents include regulatory guides, topical staff and contractor reports (NUREG series), generic letters, technical specifications, and license conditions. These documents are supported by international standards, consensus national standards, and authoritative recommendations (such as those of the ICRP and NCRP). However, these supporting documents have no official status unless they are referenced in or adopted by a regulation or documents providing regulatory guidance, such as regulatory guides or standard review plans. Of particular importance are NUREG-0800, the "Standard Review Plan," which guides the staff in reviewing safety analysis reports, and Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports," which guides the applicant in writing safety analyses. Chapter 11, "Radioactive Waste Management," of the Standard Review Plan addresses the control of radioactive effluents. Chapter 12 addresses "Radiation Protection." Chapter 15, "Accident Analysis," details how to calculate offsite and control room operator doses for design-basis accidents. Paragraph (g) of 10 CFR 50.34, "Conformance with the Standard Review Plan," makes the evaluation of the facility against the Standard Review Plan a requirement.

As discussed under Article 6, the Reactor Oversight Process has cornerstones for radiation safety. The cornerstone, "Public Radiation Safety," focuses on the effectiveness of the plant's programs to meet applicable Federal limits involving the exposure, or potential exposure, of members of the public to radiation and ensure that the effluent releases from the plant are ALARA. The cornerstone, "Occupational Radiation Safety," focuses on the effectiveness of the plant's program(s) to maintain the worker dose within the regulatory limits and provide occupational exposures that are ALARA.

15.3 Regulations

The regulations that apply to radiation protection are 10 CFR Part 20, "Standards for Protection Against Radiation," and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

15.3.1 10 CFR Part 20, "Standards for Protection Against Radiation"

This part of NRC regulations establishes requirements for radiation protection for all NRC licensees. The requirements in 10 CFR Part 20 are supplemented by specific requirements for specific operations and specific kinds of licenses. In particular, these supplementary requirements include Part 30 for byproduct material licensees, Part 34 for radiographic operations, Part 35 for medical users of byproduct material, Part 39 for oil well logging, Part 40 for users of source materials, Part 50 for nuclear power plants, Part 70 for special nuclear material users, and Part 71 for the transport of radioactive materials.

The most recent major revision of 10 CFR Part 20, issued in 1991, adopted the recommendations, quantities, and models recommended in ICRP Publication 26, "Recommendations of the International Commission on Radiological Protection" and ICRP Publication 30, "Limits of Intakes of Radionuclides by Workers," as well as some recommendations from NCRP Report No. 91, "Recommendations on Limits for Exposure to Ionizing Radiation," issued in June 1987. 10 CFR Part 20 provides relatively comprehensive coverage of general requirements for radiation protection. It is divided into subparts, with each subpart addressing a specific area of radiation protection, such as occupational and public dose limits, positing, surveys, monitoring, waste disposal, reporting, etc.

The requirements in Part 20 are not entirely consistent, in detail, with international standards such as IAEA's Basic Safety Standards. The main areas of difference include: use of the effective dose equivalent in Part 20 versus the effective dose in the Basic Safety Standards; an annual occupational dose limit on the effective dose equivalent of 0.05 Sv in Part 20 versus 0.02 Sv in the Basic Safety Standards; and use of the ICRP-30 biokinetic models in Part 20 versus more recent models used in the Basic Safety 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-30, with prior NRC approval. In addition, many licensees and agencies have administrative dose limits that are similar to, or lower than, those in the Basic Safety Standards, and most other licensees operate at occupational doses far below those that are ALARA. The current Part 20 provides a level of radiation protection that in almost all situations is comparable to that provided by international standards.

15.3.2 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"

10 CFR Part 50 is the principal regulation that addresses the safety of nuclear power plants. However, only a small part directly addresses radiation protection. [The revised dose criteria for design-basis accidents are stated in 10 CFR 50.34(a)(1)(ii)(D) for future licensing actions after implementation of the revised rule in 1997. The dose criteria for most operating nuclear power reactors are stated in 10 CFR 100.11(a).] Even so, the parts of 10 CFR Part 50 that do affect radiation protection are significant. Of particular importance are paragraph (a) of 10 CFR 50.34, "Design Objectives for Equipment to Control Releases of Radioactive Material in Effluents — Nuclear Power Reactors," Appendix I to 10 CFR Part 50, and paragraph (g) of 10 CFR 50.34, "Conformance with the Standard Review Plan," which requires NRC review of the in-plant radiation protection program.

15.4 Radiation Protection Activities

Radiation protection activities apply to occupational workers and to members of the public.

15.4.1 Control of Radiation Exposure of Occupational Workers

In addition to focusing on personnel qualifications for licensing, NRC's oversight and regulation of the radiation protection programs ensure that the safety analysis report and radiation protection plan properly address each item in 10 CFR Part 20, as well as the "Instruction to Workers" provisions of 10 CFR Part 19 and the provisions of the relevant regulatory guides, such as Regulatory Guide 1.8, "Personnel Selection and Training," and Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposure As Low As Practicable (Nuclear Reactors)."

Once NRC issues a license, it maintains an active regulatory program, which includes routine monitoring of licensee and regional reports to alert NRC staff of potential problems in radiation safety. These problems range from major repairs of highly radioactive components inside the facility to contamination from small leaks of liquid and gaseous materials. The staff evaluates the reports and discusses them with regional NRC inspection staff. Significant health physics problems can trigger significant reactive regional inspections or a generic communication to the industry.

The program for occupational radiation control has succeeded in reducing doses. NRC staff has been collecting the annual occupational exposure data for light-water reactors since 1969. The doses are strongly influenced by the amount and kind of maintenance performed, so the data fluctuate from year to year. Still, clear trends are evident. Using the average collective dose per reactor as the reference statistic, one can conclude that the doses were almost randomly variable before the accident at TMI Unit 2. Thereafter, the doses increased as a result of the extensive modifications required of all nuclear power plants in response to new requirements. The average collective dose reached a peak of 7.91 person-Sv (791 person-rem) per reactor in 1980. Since then, doses have declined almost steadily to the current level of slightly above 1 person-Sv (100 person-rem) per reactor, where they have remained for the past 5 years (1998–2002, the last year for which the data have been compiled). The 2001 average collective dose value of 1.07 person-Sv (107 person-rem) per reactor was the lowest average collective dose recorded since data collection began in 1969. Although the average doses for both PWRs and BWRs have been steadily declining, over the past 5 years the average BWR dose has exceeded the average PWR dose by roughly 80 percent (due, in part, to the higher average dose rates and larger work force at BWRs). In 2002, the 73,242 workers at nuclear plants received 121 person-Sv (12,126 person-rem) for an average of 0.0017 Sv (0.17 rem) per worker. This represents an 82-percent drop in average worker dose from the 1973 value of 0.0095 person-Sv (0.95 person-rem) per worker.

15.4.2 Control of Radiation Exposure of Members of the Public

The regulations in 10 CFR 50.34a and Appendix I to 10 CFR Part 50 define the ALARA criteria. Specifically, Appendix I defines the following ALARA criteria:

- From liquid effluents, the annual dose commitments shall be no more than 0.00003 Sv (0.003 rem) to the whole body or 0.0001 Sv (0.01 rem) to any organ.

- From airborne effluents, the annual dose commitments shall be no more than0.01 cGy (0.01 rad) to air from gamma radiation or 0.02 cGy (0.02 rad) to air from beta radiation, and .00005 Sv (0.005 rem) to the whole body or .00015 Sv (0.015 rem) to any organ.

- In addition to meeting these criteria, the general guidance, on an interim basis until replaced by a different value, is to reduce the population dose as much as feasible at a cost not to exceed a figure of merit of $10 per person-Sv ($1000 per person-rem) avoided as part of a cost benefit analysis. (Current guidance suggests $2000 per person-rem.)

Appendix I also specifies effluent monitoring, environmental monitoring, investigations, land-use censuses, and reporting.

Appendix I was issued in 1975, but was not fully implemented for all operating reactors for several years. Data from programs that resulted from this appendix were documented for both releases (NUREG/CR-2907, Vol. 17, 1995) and dose (NUREG/CR-2850, Vol. 14, 1996) for 17 years (1975–1992). Over this period, the energy generation by U.S. nuclear power plants *increased* by a factor of 3.6, while the total annual dose from releases from nuclear power plants to the U.S. population *decreased* by a factor of 27.6 (NUREG/CR-2850, Vol. 14, 1996).

ARTICLE 16. EMERGENCY PREPAREDNESS

1. 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.

For any new nuclear installation, such plans shall be prepared and tested before [the installation] commences operation above a low power level agreed [to] by the regulatory body.

2. 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.

3. 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 appropriate steps for the preparation and testing of emergency plans for their territory that cover the activities to be carried out in the event of such an emergency.

This section discusses (1) emergency planning and emergency planning zones (2) offsite emergency planning and preparedness (3) emergency classification system and action levels (4) recommendations for protection in severe accidents (5) inspection practices and regulatory oversight (6) responding to an emergency and (7) international arrangements.

The major change to this section describes the fundamental changes have occurred in response to national emergencies as a result of the terrorist events of September 11, 2001.

16.1 Background

NRC's responsibilities for radiological emergency preparedness stem from NRC licensing functions under the Atomic Energy Act of 1954 and the Energy Reorganization Act of 1974. Both statutes specifically authorize the agency to promulgate regulations that it deems necessary to fulfill its responsibilities under the Acts. Following the accident at TMI Unit 2 in March 1979, the regulations were amended to require significant changes in emergency planning and preparedness for U.S. commercial nuclear power plants. NRC's emergency planning regulations are now an important part of the regulatory framework for protecting public health and safety, and have been adopted as an additional facet to NRC's defense-in-depth philosophy. 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 there is a spectrum of accidents that could exceed the design-basis accidents that nuclear plants are required to accommodate without significant public health and safety impacts. For design-basis accidents, the small releases that might occur would not likely require responses such as evacuating or sheltering the general public. These actions only become important in considering accidents that are much less probable than design-basis accidents. NUREG-0396, “Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light-Water Nuclear Power Plants,” December 1978, and NUREG-0654/FEMA-REP-1 (NUREG-0654), “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” Rev. 1, November 1980, describe the emergency planning basis.

16.2 Offsite Emergency Planning and Preparedness

The accident at TMI Unit 2 revealed that much better coordination and more comprehensive emergency plans and procedures were needed if NRC and the public were to have confidence in the readiness of onsite and offsite emergency response organizations to respond to a nuclear emergency. Participation by State and local governments in emergency planning for nuclear power plants in the United States was, and still remains, largely voluntary. Before the accident, there had been no clear obligation for the State and local governments to develop emergency plans for radiological accidents, and the Federal role was one of assistance and guidance. After the accident, NRC amended its emergency planning regulations to require, as a condition of licensing, that each applicant and licensee submit the radiological emergency response plans of State and local governments that are within the plume exposure zone, as well as the plans of State governments within the ingestion pathway zone [10 CFR 50.33(g) and 50.54(s)].

In December 1979, the President directed FEMA to take the lead in ensuring the development of acceptable State and local offsite emergency plans and activities for nuclear power facilities. FEMA’s role and responsibilities were subsequently codified in NRC and FEMA regulations and in a Memorandum of Understanding between the two agencies.

FEMA provides its findings regarding the acceptability of the offsite emergency plans to NRC, the agency that has the ultimate responsibility to determine the overall acceptability of radiological emergency plans and preparedness for a nuclear power reactor. NRC will not issue a license to operate a nuclear power reactor unless it finds that the state of onsite and offsite emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in a radiological emergency. NRC bases its finding on a review of the FEMA findings and determinations as to whether State and local emergency plans are adequate and capable of being carried out, and on its own assessment of whether the onsite emergency plans are adequate and capable of being implemented [10 CFR 50.47(a)].

The principal guidance for preparing and evaluating radiological emergency plans for licensee and State and local government emergency planners is NUREG-0654/FEMA-REP-1, Rev. 1, a joint NRC and FEMA document. NUREG-0654 gives evaluation criteria for meeting the emergency planning standards in NRC’s and FEMA’s regulations [10 CFR 50.47(b) and 44 CFR Part 350, respectively]. These criteria provide a basis for licensees and State and local governments to develop acceptable emergency plans.

NRC and FEMA coordinate their efforts in evaluating periodic emergency response exercises, which are required by 10 CFR Part 50, Appendix E. IV. F.2, to be conducted every two years at all operating nuclear power plant sites. These full-participation exercises are integrated efforts by the licensee and State and local radiological emergency response organizations that have a role under the plan. NRC evaluates the licensee’s performance, and FEMA evaluates the response by State and local agencies. In some cases, various Federal response agencies also participate in these exercises. Any weaknesses or deficiencies that are identified by NRC or

FEMA as a result of the exercise must be corrected through appropriate remedial actions. Besides the biennial exercise of the plume exposure pathway plans, States are required to participate in an ingestion pathway exercise every six years with a nuclear power plant located within the States. There is no requirement to involve members of the public in any of the emergency preparedness exercises.

16.3 Emergency Classification System and Emergency Action Levels

NRC regulations establish four classes of emergencies in order of increasing severity. Specifically, these are (1) Unusual Event, (2) Alert, (3) Site Area Emergency, and (4) General Emergency. The specific class of emergency is declared on the basis of plant conditions that trigger the emergency action levels. Typically, licensees have established specific procedures for carrying out emergency plans for each class of emergency. The event classification initiates all appropriate actions for that class, including notification of offsite authorities, activation of onsite and offsite emergency response organizations, and, where appropriate, protective action recommendations for the public. These same emergency classes are also found in the State and local plans that support each nuclear power plant.

NUREG-0654 gives examples of initiating conditions for each of the four emergency classes. These conditions form the basis for each licensee to establish specific indicators, called emergency action levels. These levels provide a clear basis for rapidly identifying a possible problem and notifying the onsite emergency response organization and the offsite authorities that an emergency exists. Under NRC regulations, the licensee and State and local governmental authorities must discuss and agree upon the levels, and NRC must approve them. In Regulatory Guide 1.101, “Emergency Planning and Preparedness for Nuclear Power Reactors” Rev. 4, July 2003, NRC endorsed the guidance in NUMARC/NESP-007, “ Methodology for Development of Emergency Action Levels,” Rev. 2, January 1992, and NEI 99-01, “Methodology for Development of Emergency Action Levels,” Rev. 4, January 2003, as acceptable alternatives to develop emergency action levels.

16.4 Recommendations for Protective Action in Severe Accidents

The technical basis and guidance for determining protective actions in the United States for severe (core damage) reactor accidents are given in NUREG-0654, “Criteria for Protective Action Recommendations for Severe Accidents,” Supplement 3, July 1996, and EPA 400-R-92-001, “Manual of Protective Action Guides and Protective Actions for Nuclear Incidents” May 1992. These documents reflect the conclusions that have been developed from severe accident studies, such as NUREG-1150, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants.”

Guidance for response procedures and training manuals for NRC staff is given in NUREG/BR-0150, “Response Technical Manual 96.” 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 TECDOC-955, “Generic Assessment Procedures for Determining Protective Actions During a Reactor Accident.”

NRC considers evacuation and sheltering to be the two primary protective actions and prefers prompt evacuation for the population near a plant in a severe reactor accident. However, NRC is currently evaluating this position, as under some circumstances, it may be bettter to shelter in place. In addition, a supplemental protective action for the general population involves using the thyroid-blocking agent potassium iodide. NRC amended its regulations for emergency planning in 2001. This amendment, “Consideration of Potassium Iodide in Emergency Plans,” requires that each State consider giving potassium iodide as a protective measure to the general public, as a supplement to evacuation and sheltering. NRC found that potassium iodide is a reasonable, prudent and inexpensive supplement to evacuation and sheltering for specific local conditions. In addition, NRC has funded an initial supply of potassium iodide for States that choose to give potassium iodide to the general public as part of their emergency plans. To date, 19 states have requested and received potassium iodide tablets. NRC recently updated its program to distribute 65 mg tablets in addition to 130 mg tablets. In January 2002, NRC, in cooperation with the cognizant agencies updated the Federal policy statement on potassium iodide prophylaxis to reflect the changes in NRC regulations. NRC also requested the National Academy of Sciences to develop recommendations for effective potassium iodide distribution. The final report of the National Academy of Sciences was published in December 2003.

16.5 Inspection Practices — Reactor Oversight Process for Emergency Preparedness

NRC’s Reactor Oversight Process, discussed in Article 6, addresses emergency preparedness. Specifically, the process allows the licensee latitude in managing emergency preparedness programs, including corrective actions, as long as the performance indicators and inspection findings are within an acceptable performance band. As explained in Article 6, NRC handles inspection findings through its Significance Determination Process.

Emergency preparedness is the final barrier between reactor operations and protection of public health and safety. As such, emergency preparedness is a major component of the Reactor Oversight Process, and is one of the seven recognized cornerstones of safety in the process. The objective established for this cornerstone is, “Ensure that the licensee is capable of implementing adequate measures to protect the public health and safety during a radiological emergency.” Oversight of this cornerstone is achieved through three performance indicators and a supporting risk-informed inspection program. The performance indicators are Drill and Exercise Performance, Emergency Response Organization Drill Participation, and Alert and Notification System Reliability. The performance indicator for Drill and Exercise Performance monitors timely and accurate licensee performance in drills, exercises, and actual events when presented with opportunities to classify emergencies, notify offsite authorities, and recommend protective actions. The indicator for Emergency Response Organization Drill Participation measures the percentage of key members of the licensee’s emergency response organization who have participated in proficiency enhancing drills, exercises, training opportunities, or an actual event over a certain time. The Alert and Notification System Reliability indicator monitors the reliability of the offsite alert and notification system, which is a critical link for alerting and notifying the public of the need to take protective actions.

Under the Reactor Oversight Process, this cornerstone includes the following inspectable areas:

- Problem Identification and Resolution: Inspectors evaluate the licensees’ programs for problem identification and resolution as they relate to emergency preparedness.
- Drill and Training Evolution Observation: Inspectors evaluate drills and simulator-based training evolutions in which shift operating crews participate.

- Biennial Exercise: Inspectors independently observe the licensee's performance in classifying, notifying, and developing recommendations for protective actions, and other activities during the exercise. The inspectors also ensure that the licensee's critique is consistent with their observations.

- Alert and Notification System: Inspectors verify the compliance of the testing program with program procedures.

- Emergency Action Level Revision Review: Inspectors review all of the licensee's changes to emergency action levels to determine if any of the changes have decreased the effectiveness of the emergency plan.

- Emergency Response Organization Augmentation: 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.

- Performance Indicator Verification: Inspectors verify that the data reported for the performance indicator values are valid.

- Emergency Plan Changes: Inspectors sample changes to the emergency plan to ensure that the effectiveness of the emergency plan has not decreased.

Although FEMA has no direct regulatory authority over State and local governments, and the evaluators of FEMA exercises are not considered inspectors, FEMA's exercise findings carry substantial weight in NRC's regulatory process. State governments and NRC are notified of significant deficiencies in offsite performance shortly after the exercise, and FEMA issues a formal exercise report about 90 days after the exercise. This report identifies FEMA's exercise findings. Because of the potential effect of deficiencies on offsite emergency preparedness, they are expected to be corrected within 120 days of the exercise. After the deficiencies are corrected, the FEMA findings are expected to be formally closed either before or during the next exercise. Failure of offsite organizations to correct deficiencies in a timely manner could lead to FEMA's withdrawal of its finding of "reasonable assurance."

16.6 Responding to an Emergency

Fundamental changes have occurred in response to national emergencies as a result of the terrorist events of September 11, 2001. This section explains the roles of the Federal government, licensees, State and local governments, and NRC. It also explains the security aspects supporting the response.

Federal Response

The Federal response structure has been revamped with the creation of the Department of Homeland Security and the implementation of Homeland Security Presidential Directive -5. This directive establishes the Secretary of Homeland Security as the principal Federal official for managing domestic incidents. Under the Homeland Security Act of 2002, the Secretary 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. The Secretary coordinates the resources of the Federal Government needed to or recover from terrorist attacks, major disasters, or other emergencies if and when any one of the following four conditions applies:

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

(2) The resources of State and local authorities are overwhelmed and Federal assistance has been requested by the appropriate State and local authorities.

(3) More than one Federal department or agency has become substantially involved in responding to the incident.

(4) The Secretary has been directed by the President to assume responsibility for managing the domestic incident.

The revised framework will implement a new all-hazards Federal Response Plan. During the transition from the current Federal incident management and emergency response plans to the full National Response Plan, the September 30, 2003, Initial National Response Plan will act as a bridging document. It implements, on an interim basis, the domestic incident management authorities, roles, and responsibilities of the Secretary of Homeland Security. It also provides interim guidance on Federal coordinating structures and processes for managing domestic incidents. It applies to managing domestic incidents in the context of terrorist attacks, major disasters, and other emergencies. Until the full National Response Plan becomes effective, current Federal incident management and emergency response plans remain in effect, except as specifically modified by the Initial National Response Plan.

The Federal response is primarily designed to support the efforts of the facility operator and offsite officials. For an emergency with potential radiological consequences, Federal response activities are conducted according to the “Federal Radiological Emergency Response Plan.” This document describes the role of the lead Federal agency (i.e., NRC in this type of emergency) and other support Federal agencies. The lead Federal agency will lead and coordinate all Federal on scene actions and assist State and local governments in determining measures to protect life, property, and the environment. It will ensure that the Department of Homeland Security/FEMA and other Federal agencies assist the State and local government agencies in implementing protective actions, if requested by the State and local government agencies. The lead Federal agency will oversee the onsite response, monitor and support owner or operator activities (when there is an owner or operator), provide technical support to the owner or operator, if requested, and serve as the principal Federal source of information about onsite conditions. The lead Federal agency will provide a hazard assessment of onsite conditions that might have significant offsite impact and ensure onsite measures are taken to mitigate offsite consequences.

Licensee, State, and Local Response

NRC recognizes the nuclear power plant operator (licensee) and the State or local government as the two primary decision makers in a radiological emergency at a licensed power reactor. The operator is primarily responsible for mitigating the consequences of an incident on site, and recommending timely and proper protective actions to State and local authorities. The State or local governments are ultimately responsible for implementing proper protective actions for public health and safety.

NRC's Response

In fulfilling its legislative mandate for protecting the public health and safety, NRC has developed a plan and procedures that detail NRC's response to incidents involving licensed material and activities (NUREG-0728, Rev. 3, "NRC Incident Response Plan"). In accordance with that plan, NRC will initially assess any reported event, and decide whether or how it will respond as an agency. NRC will generally dispatch a team to the site for all serious incidents to meet its statutory and regulatory obligations as the lead Federal agency including assisting the State in interpreting and analyzing technical information while keeping other responding Federal agencies updated on event conditions and coordinating any multi-agency Federal response.

Once NRC has decided to respond as an agency, the agency's Operations Center in the Washington, DC, area and the associated regional Incident Response Center are activated. The NRC headquarters Operations Center 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 (FEMA, DOE, EPA, etc.), and (5) respond to inquiries from the national media. The NRC headquarters Operations Center will be staffed with experts on the facility and its emergency plans, response to incidents, and liaison activities. Early in an incident, the Regional Administrator provides operational authority from the affected regional office, if necessary, from the regional incident response center because regional office personnel are usually most familiar with details of the affected facility. When a major NRC onsite presence is required, a team will be dispatched from the affected regional office. The NRC headquarters Operations Center will direct NRC response for about 4–8 hours until the lead is transferred to the NRC Site Team, if applicable.

As soon as the NRC Site Team arrives at the scene and declares that it is ready to assume the agency's leadership role, it is given certain authorities and responsibilities which may include the authority to direct the Agency's response. The NRC Site Team then sends representatives to response centers that are used by the facility and offsite officials to coordinate the response. The NRC Site Team has access to extensive radiological monitoring capabilities through DOE, including field teams and aerial monitoring. The Federal radiological monitoring efforts to support State and local officials are coordinated at the Federal Radiological Monitoring and Assessment Center established by DOE after coordination with the lead Federal agency and State and local responders. The NRC Site Team also sends representatives to the Joint Information Center established by the facility or local government, to interact with the media. NRC regularly participates in exercises of its response program to ensure readiness to respond, participating fully in about four nuclear power plant exercises and one fuel cycle facility exercise each year. Since September 2001, NRC has received a number of requests to participate in Federal interagency exercises intended to improve the Federal, State and local response to a wide spectrum of potential domestic incidents. NRC assisted in the planning and participated in the Top Officials 2 (TOPOFF2) exercise. NRC's participation in TOPOFF2 provided the agency valuable perspective in multi-event response. Improvements were also gained in interagency cooperation and a better understanding of response roles during emergency situations.

Security Aspects Supporting Response

Before September 11, 2001, the security measures in place at nuclear facilities provided reasonable assurance that the health and safety of the public would be protected in the event of an attack encompassed within the design basis threat of radiological theft and sabotage, which is described in 10 CFR 73.1. Since September 11, 2001, the defensive capability of the nuclear industry has been significantly enhanced as a result of the voluntary actions taken by licensees in response to the advisories issued by NRC, and as required by the Orders issued on February 25, 2002, and January 7, 2003, and followed by the three Orders issued April 29, 2003. The enhancements include security measures against threats from an insider, waterborne attack, vehicle bomb attack, and land-based assault. In addition, one of the Orders issued April 29, 2003, identified a revised design-basis threat against which licensees must be prepared to defend. NRC will consider additional measures in the future as necessary.

NRC receives a substantial and steady flow of information from the national intelligence community, law enforcement, and licensees. NRC continuously evaluates this information to assess threats to regulated facilities or activities. NRC works with a variety of other Federal agencies, in particular the Department of Homeland Security and the Homeland Security Council, to ensure that security around nuclear power plants is well coordinated and that responders are prepared if a significant event occurs. If an event were to occur, NRC would coordinate the resources of more than 18 Federal agencies as indicated in the previous section on NRC Response, to mitigate the radiological consequences of a serious accident or successful attack.

16.7 International Arrangements

NRC has agreements with Canada and Mexico and commitments to the IAEA.

NRC has signed agreements with Canada and Mexico under which it will promptly notify and exchange information in the event of an emergency that has the potential for trans-boundary effects. The agreement with Canada is the “Agreement Between the Government of the U.S. of America and the Government of Canada on Cooperation in Comprehensive Civil Emergency Planning and Management.” It is implemented by the procedure specified in “Administrative Arrangement Between the U.S. Nuclear Regulatory Commission and the Atomic Energy Control Board of Canada for Cooperation and the Exchange of Information in Nuclear Regulatory Matters.” (Both documents are dated June 21, 1989.)

The agreement with Mexico is the “Agreement for the Exchange of Information and Cooperation in Nuclear Safety Matters.” It is implemented by the “Implementing Procedure for the Exchange of Technical Information and Cooperation in Nuclear Safety Matters Between the Nuclear Regulatory Commission of the U.S. of America and the Commission National de Seguridad Nuclear y Salvaguardias of Mexico.” (Both documents are dated October 6, 1989.)

To meet the U.S. commitment under the IAEA “Convention on Early Notification of a Nuclear Accident,” NRC will promptly notify the IAEA if a serious accident occurs at a commercial nuclear power plant. Afterward, NRC will work with the U.S. Department of State to update the IAEA.

Since 2001, the United States has fully participated in the International Nuclear Event Scale (INES) by evaluating operating reactor events and reporting any events at Level 2 or higher to the IAEA. NRC has incorporated this scale into its Emergency Preparedness initial staff training module and has trained the appropriate staff in its use.

ARTICLE 17. SITING

Each Contracting Party shall take the appropriate steps to ensure that appropriate procedures are established and implemented for

(v) 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 NRC's responsibilities for siting: site safety, environmental protection, and emergency preparedness. First, this section discusses the regulations applying to site safety and their implementation. It emphasizes regulations applying to seismic, geological, and radiological assessments. Next, it explains environmental protection. Emergency preparedness is discussed in Article 16, "Emergency Preparedness." International arrangements, which would apply to Contracting Parties in obligation (iv), above, are also discussed in Article 16.

New information reported since the previous *U.S. National Report* includes early site permit review activities the issuance of Review Standard RS-002, "Processing Applications for Early Site Permits," and research into probabilistic seismic hazard analyses.

17.1 Background

NRC's siting responsibilities stem from the Atomic Energy Act of 1954, the Energy Reorganization Act of 1974 (as discussed earlier), and the National Environmental Policy Act of 1969. These statutes confer broad regulatory powers on the Commission, and specifically authorize NRC to promulgate regulations that it deems necessary to fulfill its responsibilities under the Acts.

NRC's siting regulations are integral to protecting public health and safety and the environment. Siting away from densely populated centers has been, and will continue to be, an essential component of NRC's defense-in-depth safety philosophy, which also includes multiple-barrier containment and redundant safety systems. NRC's defense-in-depth philosophy has been expanded to include safety, security, and emergency preparedness. The primary factors that determine public health and safety are the reactor design, and construction and operation of the facility. However, siting factors and criteria are important in ensuring that radiological doses from normal operation and postulated accidents will be acceptably low, natural phenomena and man-made hazards will be properly accounted for in the design of the plant, and the human environment will be protected during the construction and operation of the plant.

For the first time since the 1970s the nuclear power industry in the United States expressed an interest in seeking approval for sites that could host new nuclear power plants. To ensure that the Agency could effectively carry out its responsibilities associated with, among others, an early site permit application, NRC consolidated regulatory functions to (1) manage near term future licensing activities, (2) work with stakeholders regarding new reactor licensing activities, and (3) assess NRC's readiness to perform new reactor licensing reviews. In 2003, applicants submitted three early site permit applications to NRC for sites located in Virginia, Illinois and Mississippi. All the current sites under review have existing nuclear power plants, which afforded the applicants the opportunity to take advantage of existing physical and administrative infrastructures, of existing programs and siting information, and the opportunity to have a reduced impact to the environment when compared to undeveloped locations. In anticipation of these applications, and to ensure that future license applicants and the public understand NRC's review process of programs and siting information, NRC documented its review process and review criteria in Review Standard RS-002, "Processing Applications for Early Site Permits." RS-002 provides detailed direction for managing and conducting the review and expands upon existing regulatory guidance, for example, standard review plans, for the safety (including emergency planning discussed in Article 16) and environmental reviews.

17.2 Safety Elements of Siting

This section explains the safety elements of siting. After providing a short background, it explains seismic and geological assessments. It then discusses radiological assessments performed for initial licensing, as a result of facility changes, and according to regulatory developments that have occurred since the licensing of all U.S. operating plants.

17.2.1 Background

NRC's site safety regulations consider societal and demographic factors, man-made hazards (such as airports and dams), and physical characteristics of the site (such as seismic and meteorological factors) that could affect the design of the plant. The requirements are specified in 10 CFR Part 100, "Reactor Site Criteria," issued in 1962; 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," added in 1973; and Subpart B, including 10 CFR 100.23, "Geologic and Seismic Siting Criteria," which became effective in 1997. The requirements in 10 CFR 100.23 apply to applicants for an early site permit, a combined license, a construction permit, or an operating license on or after January 10, 1997. Regulatory Guide 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion," describes methods acceptable to NRC staff for implementing those requirements, and Standard Review Plan Section 2.5.2, "Vibratory Ground Motion," Rev. 3, guides the staff in its reviews.

The applicant's safety analysis report is required to describe characteristics in and around the site, and contain accident analyses that are relevant to evaluating the suitability of a site. A number of regulatory guides provide guidance regarding issues of site safety that applicants need to address. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," guides the staff in reviewing the site safety content of these reports. Parts of NUREG-0800 that apply to the review of early site permits are identified in Review Standard 002.

Once licensed to operate, the licensee is expected to monitor the environs around the nuclear power plant, and report changes in the environs in its safety analysis report that may affect the continued safe operation of the facility. Such changes may affect the regulatory status of the facility in different ways. For example, if population characteristics near the plant change during the plant's operating lifetime, the licensee must continue to assess the emergency planning criteria — rather than the siting criteria — to ensure the licensee's continued ability to carry out measures to protect the public. As another example, if the plant experiences an extreme natural event (such as an earthquake or flood), the licensee must demonstrate that no functional damage has occurred to those features that are deemed necessary for continued safe operation without undue risk to the public, and that the licensing basis is maintained.

17.2.2 Assessments of Seismic and Geological Aspects of Siting

Assessments applying to seismic and geological aspects of siting are detailed in the siting regulations stated above under 17.2.1. More recent developments in assessments include the use of probabilistic methods of analysis incorporating various models for earthquake size, ground motion, and other parameters. These methods have the advantage of not only incorporating various models (and data), but also weighting them on the basis of judgments as to their validity. Thus, these methods provide an explicit expression for the uncertainty in ground motion estimates and a means of assessing sensitivity to various input parameters.

Other recent developments include the following studies. NRC-sponsored research to develop further guidance on the spectrum of ground motion resulted in NUREG/CR-6728, "Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-Consistent Ground Motion Spectra Guidelines," issued in October 2001. In addition, EPRI updated parts of its probabilistic seismic hazard analysis for use in early site permit applications.

Additionally, the U.S. Geological Survey updated its national seismic hazard map in early 2003. The Survey's process for national seismic hazard assessment does not appear to differ much from the probabilistic seismic hazard analysis that has been followed in NRC-sponsored studies. NRC is sponsoring research to evaluate the similarity of the Survey's process and NRC studies. The feasibility of NRC's adapting it for use depends on the extent to which the Survey's results are suitable for low annual frequency events that generally apply to nuclear power plant facilities. Other government agencies are also involved in probabilistic seismic hazard analysis associated with very low annual frequency events. These studies and other efforts in the research communities are likely to affect the guidelines for applying the seismic and geological criteria of the siting rule.

17.2.3 Assessments of Radiological Consequences

The Reactor Site Criteria Rule, 10 CFR Part 100, is the regulation under which all U.S. operating plants were licensed. It contains provisions for assessing whether radiological doses from postulated accidents will be acceptably low. NRC has issued the following regulatory guidance for licensees to implement the requirements regarding the radiological criteria of 10 CFR Part 100:

- Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling-Water Reactors"

- Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized-Water Reactors"

- Regulatory Guide 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants”

Although applicants perform dose analyses primarily to support reactor siting, licensees are required to evaluate the potential increase in the consequences of accidents that might result from modifying facility systems, structures, and components. Commitments (including the radiological acceptance criteria) made by the applicant during siting and documented in its final safety analysis report, remain binding until modified. Consequently, a licensee must evaluate the potential consequences of design changes against these radiological criteria to demonstrate that the design changes result in a design that still conforms to the regulations and commitments. If the consequences increase more than minimally as outlined in 10 CFR 50.59, “Changes, Tests, and Experiments” (or require a change to the Technical Specifications), as discussed in Article 14, the licensee must obtain NRC approval before implementing the proposed modification.

There have been regulatory developments since the licensing of all U.S. plants now operating. In 1996, NRC revised 10 CFR Part 100, having considered the substantial additional body of information on fission product releases that had been developed, particularly as a result of severe-accident research after the accident at TMI in 1979. The 1996 revision also replaced the dose criteria of the 1962 version of 10 CFR Part 100 for new reactors with a single value of 0.25 Sv (25 rem) as the total effective dose equivalent.

Another regulatory development was the publication of NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants” in 1995. This document distilled the body of research, particularly research conducted after the accident at TMI Unit 2, into a practical guide to estimate more realistically the source terms being released into containment, including the timing, nuclide types, quantities, and chemical form, given a representative severe core melt accident. This NUREG provides the technical bases for revised source terms and forms the basis for developing regulatory guidance mainly for future reactor designs and sites. The revised source term is also referred to as an “alternative source term.” Finally, NRC issued 10 CFR 50.67, “Accident Source Term,” effective January 2000, which allows licensees to propose, by way of a license amendment, using alternative source terms.

Experience and Examples

NRC has applied the 1996 revision to 10 CFR Part 100, along with the revised source term, in its design certification review for a passive advanced light-water reactor, the AP600. More recently, the agency has applied the practice to the AP1000 with similar results and is expected to apply it for all contemplated light-water reactors. For other-than-light-water reactor designs, applicants will have to describe their rationale for an appropriate accident source term characterization that will be subject to NRC independent review.

NRC has also approved several applications of the alternative source term. Licensees of boiling-water reactors have used the alternative source term to justify relaxing certain operability requirements for the standby gas treatment system, supporting reactor power uprates, and to resolve concerns about control room habitability. Licensees of pressurized-water reactors have used the alternative source term to justify keeping the containment penetrations open during refueling, supporting power uprates, and to resolve concerns about control room habitability. The industry continues to explore the use of the alternative source term in implementing cost-beneficial licensing actions.

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 last operating plants in the United States have been licensed, issues have arisen that must be considered in siting reviews. This section explains the effect of these issues on siting reviews. A recent change, since the previous *U.S. National Report*, is the issuing of Review Standard RS-002, dealing with early site permits, to guide NRC's review.

17.3.1 Governing Documents and Process

The environmental protection elements of siting consist of the plant's demands on the environment (e.g., water use and effects of construction and operation). These elements are addressed in 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions." 10 CFR Part 51 implements the National Environmental Policy Act of 1969, consistent with NRC's statutory authority, and reflects the agency's policy to voluntarily apply the regulations of the President's Council on Environmental Quality, subject to certain conditions. NRC recognizes its continuing obligation to conduct its domestic licensing and related regulatory functions to be receptive to environmental concerns, and to serve responsibly as an independent regulatory agency to protect the radiological health and safety of the public. Integrating environmental reviews into its routine decisionmaking, NRC considers environmental protection issues and alternatives before taking any action that may significantly affect the human environment.

The site approval process leading to the construction or operation of a nuclear power plant requires NRC to prepare an environmental impact statement. The updated and revised environmental standard review plans (NUREG-1555) guide the staff's environmental reviews for a range of applications, including "green field" (i.e., undeveloped sites) reviews for construction permits and operating licenses in 10 CFR Part 50, for early site permits in 10 CFR Part 52, Subpart A, and for combined licenses in 10 CFR Part 52, Subpart C, when the application does not reference an early site permit. These governing documents and processes are discussed in detail in Article 19 and in Review Standard RS-002, dealing with early site permits. Environmental standard review plans are also appropriate for environmental reviews of applications for combined licenses in 10 CFR Part 52, Subpart C, when the applications reference an early site permit. Reviews of early site permit applications are limited in the sense that (1) the reviews focus on the environmental effects of reactor construction and operation that have characteristics that fall within the postulated site parameters, and (2) the reviews need not assess benefits (for example, the need for power). The environmental reviews of applications for combined licenses that reference an early site permit are limited to consideration of (1) information to demonstrate that the design of the facility falls within the parameters specified in the early site permit, and (2) any significant environmental issue that was not considered in any previous proceeding on the site or design.

The environmental standard review plans in Supplement 1 to NUREG-1555 guide the staff's environmental review for license renewal applications under 10 CFR Part 54 discussed in Article 14. Guidance in the environmental standard review plan (1) instructs NRC staff responsible for environmental reviews; (2) describes how NRC reaches judgments on environmental impacts caused by constructing and operating nuclear power plants, and by allowing for period of extended operation (license renewal) and refurbishment activities; and (3) specifies how to determine the significance of these impacts.

A number of other NRC actions on siting and site suitability require environmental reviews, including issuance of limited work authorizations [10 CFR 50.10(e)(1) to (e)(3), 10 CFR 52.25, and 10 CFR 52.91], early partial decisions (10 CFR 2.600, Subpart F), and pre-application early reviews of site suitability issues (10 CFR Part 52, Appendix Q).

17.3.2 Other Considerations for Siting Reviews

NRC granted the last site approval in 1978. Since then, and coincident with the publication of the initial environmental standard review plan, many changes to the regulatory environment have affected 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 NRC. Some of these changes and their effects on the environmental standard review plans are highlighted in the paragraphs below.

17.3.2.1 Early Site Permits, Standard Design Certifications, and Combined Licenses

In the late 1980s, NRC issued regulations that provided an alternative licensing framework to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," which provided for a construction permit followed by an operating license. The new framework provided in 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," introduced the concept of approving designs independent of sites, and approving sites independent of designs, and then efficiently linked the approvals to result in the approval to construct and operate the facility. As discussed earlier, NRC has received three early site permit applications under 10 CFR Part 52 and is actively conducting siting reviews. Toward that end, NRC issued a review standard (RS-002), which embodies the environmental guidance provided in NUREG-1555, the environmental standard review plan, and the outcome of interactions with stakeholders from 2000 to 2003. In addition, NRC is revising Part 52 to reflect experience gained in its use.

17.3.2.2 Environmental Justice

Presidential Executive Order 12898, issued in February 1994, instructed Federal agencies to make "environmental justice" part of each agency's mission by addressing disproportionately high and adverse human health or environmental effects of Federal programs, policies, and activities on minority and low-income populations. Although the order was not binding on independent agencies, NRC agreed to implement the executive order to the extent practicable. NRC has since gained experience in implementing the goals of the executive order during the conduct of its environmental reviews, for example, during the conduct of license renewal reviews under 10 CFR Part 54, discussed in Article 14. In November 2003, the Commission reiterated its views that environmental justice, as applied at NRC, means that the Agency will make an effort, during its environmental reviews conducted to comply with the National Environmental Policy Act, to become aware of the demographic and economic circumstances of local communities where nuclear facilities are to be sited, and take care to account for significant impacts attributable to the special character of the community.

17.3.2.3 Yellow Creek Decision

The authority of NRC is limited in matters that are expressly assigned to the EPA as shown by the decision on Yellow Creek, a proposed Tennessee Valley Authority facility, in 1978. Specifically, the decision (8 NRC 702) by the Atomic Safety and Licensing Board determined that NRC's authority is limited for those matters that are expressly assigned to the EPA by the Federal Water Pollution Control Act Amendments of 1972.

17.3.2.4 Open Access to Transmission Lines and Economic Deregulation

Recent changes in the economic regulation of utilities have expanded the options to be addressed in considering the need for power in environmental impact statements, as required by Section A(4) of Appendix A to 10 CFR Part 51. Regulatory agencies in some States have initiated economic deregulation, and the Federal Energy Regulatory Commission has adopted regulations to ensure that all power generators have open access to power transmission facilities. The effects of these changes on environmental review procedures are likely to be significant, especially in defining demand for power, service areas, and benefits. As outlined in 10 CFR Part 52, applicants for early site permits need not provide a benefits assessment (for example, need for power), but must address alternative sites. The selection of alternative sites had been influenced by the shape and size of the utilities' service areas; with deregulation, the traditional concept of service areas is being revisited. The early site permits currently under review were submitted by other-than-electric utility organizations who established a rationale for their consideration of alternative sites. If the benefits assessment is not resolved at the early site permit stage, it must be provided with the combined license application under Part 52.

17.3.2.5 Severe Accident Mitigation Alternatives

When NRC published the original environmental standard review plans, environmental impact statements did not consider design alternatives to mitigate the consequences of severe accidents. Current NRC policy requires consideration of such alternatives in the environmental impact statements that are prepared for an operating license or for license renewal. Severe accident mitigation design alternatives have been included in final environmental statements for the Limerick 1 and 2 (NUREG-0974) operating license reviews and in the Watts Bar supplemental final environmental impact statement (NUREG-0498). Severe accident mitigation alternatives have been included in license renewal supplements to the generic environmental impact statement for license renewal of nuclear power plants (NUREG-1437). As outlined in RS-002, to the degree that sufficient design information exists at the early site permit application stage, severe accident mitigation alternatives can be considered, otherwise consideration would be deferred to the combined license stage.

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 philosophy, and how 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. Article 14, under “Verification by Analysis, Surveillance, Testing and Inspection,” also addresses this obligation. Finally, this section discusses requirements regarding reliable, stable, and easily manageable operation, specifically considering human factors and the man-machine interface. This obligation is also addressed in Article 12, “Human Factors.”

The main changes reported since the previous *U.S. National Report* are an expanded explanation of defense-in-depth, activities concerning combined licenses and early site permits, and experience and examples.

18.1 Defense-in-Depth Philosophy

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

18.1.1 Governing Documents and Process

The defense-in-depth philosophy, as applied in regulatory practice, requires that nuclear plants contain a series of independent, redundant, and diverse safety systems. The physical barriers for defense-in-depth in a light-water reactor are the fuel matrix, the fuel rod cladding, the primary coolant pressure boundary, and the containment. The levels of protection in defense-in-depth are (1) a conservative design, quality assurance, and safety culture; (2) control of abnormal operation and detection of failures; (3) safety and protection systems; (4) accident management, including containment protection; (5) emergency preparedness; and (6) security.

The defense-in-depth philosophy is embodied in Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50. General design criteria cover protection by multiple fission product barriers, protection and reactivity control systems, fluid systems, containment design, and fuel and radioactivity control. NRC staff amplified its defense-in-depth philosophy in

Regulatory Guide 1.174, which provides guidance on using a PRA in risk-informed decisions on plant-specific changes. Regulatory Guide 1.174 states that defense-in-depth should always be considered when evaluating plant-specific changes and that it consists of a number of elements, as summarized below:

- A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.

- Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.

- System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (e.g., no risk outliers).

- Defenses against potential common cause failures are preserved, and the potential for the introduction of new common cause failure mechanisms is assessed.

- Independence of barriers is not degraded.

- Defenses against human errors are preserved.

- The intent of the General Design Criteria in Appendix A to Part 50 is maintained.

The general design criteria establish the minimum requirements for the principal design criteria, which in turn establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components that are important to safety.
To ensure that a plant is properly designed and built as designed, that proper materials are used in construction, that future design modifications are controlled, and that appropriate maintenance and operational practices are followed, a good quality assurance program is needed. To meet this need, General Design Criterion 1, “Quality Standards and Records,” of Appendix A to 10 CFR Part 50 and its implementing regulatory requirements specified in Appendix B to 10 CFR Part 50, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” establish quality assurance requirements for all activities affecting the safety-related functions of the structures, systems, and components.

Pursuant to the two-step licensing process set forth in 10 CFR Part 50, an applicant for a construction permit must present the principal design criteria for a proposed facility in its preliminary safety analysis report (see 10 CFR 50.34). As guidance in writing a safety analysis report, the applicant may use Regulatory Guide 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants.” The safety analysis report must also contain design information for the proposed reactor, and comprehensive data on the proposed site. The report must also discuss various hypothetical accident situations and the safety features to prevent accidents or, if they should occur, to mitigate their effects on both the public and the facility’s employees. After obtaining a construction permit under 10 CFR Part 50, the applicant must submit a final safety analysis report to support an application for an operating license, unless it submitted the report with the original application. This report gives the details of the final design of the facility, plans for operation, and procedures for coping with emergencies. The preliminary and final safety analysis reports are the principal documents that the applicant provides for the staff to determine whether the proposed plant can be built and operated without undue risk to the health and safety of the public. NRC expects that future applications to build nuclear power plants will use the combined license process under 10 CFR Part 52, which requires submittal of a final safety analysis report before authorization to construct. See Article 19 for a discussion of the combined license review process.

NRC staff reviews safety analysis reports according to NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” to ensure that the general design criteria and other applicable regulations are satisfied. The plan ensures the quality and uniformity of staff reviews of applications to construct or operate nuclear power plants and presents well-defined bases for evaluating proposed changes to the scope and requirements of reviews.

NRC staff reviews each application to determine whether the plant design meets the Commission’s regulations (10 CFR Parts 20, 50, 73, and 100). These reviews include, in part, the characteristics of the site, including the surrounding population, seismology, meteorology, geology and hydrology; the nuclear plant design; the anticipated response of the plant to postulated accidents; the plant operations, including the applicant’s technical qualifications to operate the plant; radiological effluents; and emergency planning. In addition, each application for a nuclear installation must have a comprehensive environmental report that provides a basis for evaluating the environmental impact of the proposed facility. Regulatory Guide 4.2, “Standard Format and Content for Environmental Reports,” guides applicants on writing environmental reports. NRC staff reviews the environmental reports according to NUREG-1555, “Standard Review Plan for Environmental Reviews of Nuclear Power Plants.” This plan guides the staff in developing its environmental impact statement.

In reviewing an applicant’s submittal, NRC staff, supported by outside experts, conducts its independent technical studies to review certain safety and environmental matters. The staff states its conclusions in an environmental impact statement and a safety evaluation report, which it may update prior to granting the license. Under the two-step licensing process in 10 CFR Part 50, NRC does not issue an operating license until construction is complete and the Commission makes the findings set forth in 10 CFR 50.57.

NRC maintains surveillance over nuclear power plant construction to ensure compliance with the agency’s regulations to protect public health and safety and the environment. NRC’s inspection program has been anticipating that future applicants for construction of a nuclear power plant will apply for a combined license under 10 CFR Part 52. A construction inspection program team and steering group of NRC managers have been formed and have been developing an inspection program for future nuclear plants licensed under 10 CFR Part 52.

The inspection program being developed revises the archived 10 CFR Part 50 construction inspection program. It incorporates inspections, tests, analyses, and acceptance criteria (ITAAC) from 10 CFR Part 52, as well as lessons learned from the inspection program used in the construction era (1970–1980), and takes into account modular construction at remote locations and state-of-the-art technology that might be used for future designs such as high-temperature gas-cooled reactors.

Before construction, the NRC inspection program focuses on the applicant establishing a quality assurance program to verify that applications submitted to NRC meet specified requirements in 10 CFR Part 52 and are of a quality suitable for docketing. Inspections for this phase are listed in Inspection Manual Chapter 2501, “Early Site Permit.”

Once NRC receives an application, the inspection program focuses on supporting the NRC staff's preparation for the mandatory Atomic Safety and Licensing Board hearing, and the final Commission decision on whether a combined license should be granted. Inspections for this phase will be listed in Inspection Manual Chapter 2502, "Pre-Combined License Phase."

During construction, inspectors sample the spectrum of the applicant's activities related to performance of the ITAAC in the design basis document to confirm that the applicant is adhering to quality and program requirements. NRC inspectors will verify successful ITAAC completion on a sampling bases and will review all ITAACs. NRC will publish notices in the *Federal Register* of those ITAACs that have been completed. Inspections for this phase are listed in Inspection Manual Chapter 2503, "ITAAC."

As the applicant completes construction, the inspection program focuses on verifying the adequacy of the licensee's pre-operational programs such as fire protection, security, training, radiation protection, start-up testing, and programs that transition the organization from construction to power operations. Inspections for this phase are listed in Inspection Manual Chapter 2504, "Non-ITAAC Inspections."

18.1.2 Experience and Examples

For more than 30 years, the Atomic Energy Commission and NRC have reviewed applications submitted under the two-step licensing process in 10 CFR Part 50 and documented their reviews in safety evaluation reports and their supplements for 110 nuclear installations. Subsequently, NRC certified three standard plant designs under the design certification process in 10 CFR Part 52. General Electric's Advanced Boiling-Water Reactor design (1997), and Westinghouse's (formerly Combustion Engineering) System 80+ and AP600 designs (1997 and 2000, respectively). Westinghouse's AP1000 design is the only design certification application that is currently under review.

18.2 Technologies Proven by Experience or Qualified by Testing or Analysis

The earlier discussion under this article (18.1.1) and Article 14, Section 14.2, addresses the qualification of currently used technologies. NRC ensures that new technologies are proven as required by paragraph (b) of 10 CFR 52.47, "Contents of Application." This rule requires demonstration of new technologies through analysis, appropriate test programs, experience, or a combination thereof. Most recently, Westinghouse used separate effects tests, integral systems tests, and analyses to demonstrate that its passive safety systems will perform as predicted in its safety analysis reports for the AP600 and AP1000 standard plant designs.

18.3 Design for Reliable, Stable, and Easily Manageable Operation

NRC specifically considers human factors and the human-system interface in the design of nuclear installations. For safety analysis reports, NRC reviews the human factors engineering design of the main control room and the control centers outside of the main control room. As guidance, the staff uses Revision 1 to Chapter 18 of NUREG-0800, and Revision 2 to NUREG-0700, "Human-System Interface Design Review Guideline." NRC also uses Revision 2 to NUREG-0711, "Human Factors Engineering Program Review Model," for design certification reviews that include evaluating the design process as part of the final design of next-generation main control rooms. Human factors are also discussed in Article 12.

ARTICLE 19: OPERATION

Each Contracting Party shall take appropriate steps to ensure that:

(i) **the initial authorization to operate a nuclear installation is based upon an appropriate safety analysis and a commissioning program demonstrating that the installation, as constructed, is consistent with design and safety requirements**

(ii) **operational limits and conditions derived from the safety analysis, test, and operational experience are defined and revised as necessary for identifying safe boundaries for operation**

(iii) **operation, maintenance, inspection, and testing of a nuclear installation are conducted in accordance with approved procedures**

(iv) **procedures are established for responding to anticipated operational occurrences and to accidents**

(v) **necessary engineering and technical support in all safety related fields is available throughout the lifetime of a nuclear installation**

(vi) **incidents significant to safety are reported in a timely manner by the holder of the relevant license to the regulatory body**

(vii) **programs to collect and analyze operating experience are established, the results obtained and the conclusions drawn are acted upon and that existing mechanisms are used to share, important experience with international bodies and with other operating organizations and regulatory bodies**

(viii) **the generation of radioactive waste resulting from the operation of a nuclear installation is kept to the minimum practicable for the process concerned, both in activity and in volume, and any necessary treatment and storage of spent fuel and waste directly related to the operation and on the same site as that of the nuclear installation take into consideration conditioning and disposal**

NRC relies on regulations in Title 10, “Energy,” of the *Code of Federal Regulations* (10 CFR) and internally developed associated programs in granting the initial authorization to operate a nuclear installation and in monitoring its safe operation throughout its life. The material that follows describes the more significant regulations and programs corresponding to each obligation of Article 19. No major changes are reported in this section.

19.1 Initial Authorization to Operate

19.1.1 Governing Documents and Process

In the past, NRC licensed nuclear power plants under the traditional (two-step) licensing process set forth in 10 CFR Part 50. This licensing process requires both a construction permit and an operating license. The additional licensing processes in 10 CFR Part 52, “Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants,” provide for site approvals and design approvals in advance of construction authorization. In addition, Part 52 includes a process that combines a construction permit and an operating license with conditions into one license (combined license). Both the two-step and the combined license processes require NRC approval to construct and operate a nuclear power plant.

Each application to construct or operate a nuclear power plant is reviewed by the Advisory Committee on Reactor Safeguards, an independent statutory committee established to advise NRC on reactor safety. The Committee begins its review early in the licensing process by selecting proper stages at which to hold a series of meetings with the applicant and NRC staff. Upon completing its review, the Committee reports to the Commission.

The public also has an opportunity to have its concerns addressed. The Atomic Energy Act requires that a public hearing be held before a construction permit, early site permit, or a combined license may be issued for a nuclear power plant. The public hearing is conducted by a three-member Atomic Safety and Licensing Board, which consists of one lawyer who acts as chairperson, and two technically qualified persons. Members of the public may submit statements to the licensing board, or they may petition for leave to intervene as full parties in the hearing.

To obtain NRC approval to construct or operate a nuclear power plant, an applicant must submit safety analysis reports. Article 18 describes the final safety analysis report and NRC's review of the application for an operating license. A public hearing is neither mandatory nor automatic for an application for an operating license under 10 CFR Part 50. However, soon after NRC accepts the application for review, it publishes a notice that it is considering issuing the license. This notice states that any person whose interest might be affected by the proceeding may petition NRC for a hearing. If a public hearing is held, the same process described for the hearing for the construction permit applies.

A combined license, issued under Subpart C of 10 CFR Part 52, authorizes construction of a facility in a manner similar to a construction permit under 10 CFR Part 50. Just as for a construction permit, a hearing must be held before the decision on issuance of a combined license. However, the combined license will specify the inspections, tests, and analyses that the licensee must perform and the acceptance criteria that, if met, are necessary and sufficient to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the license and the applicable regulations. After issuing a combined license, NRC staff will verify that the required inspections, tests, and analyses have been performed and, before operation of the facility, the Commission must find that the acceptance criteria have been met. At periodic intervals during construction, NRC staff will publish notices of the successful completion of inspections, tests, and analyses in the *Federal Register*. Then, not less than 180 days before the date scheduled for initial loading of fuel, NRC will publish a notice of intended operation of the facility in the *Federal Register*. An opportunity for a second hearing exists, but petitions for this hearing will only be considered if the petitioner demonstrates that one or more of the acceptance criteria have not been (or will not be) met, and the specific operational consequences of nonconformance that would be contrary to providing reasonable assurance of adequate protection of the public health and safety.

An early site permit, issued under Subpart A of 10 CFR Part 52, provides for resolution of site safety, environmental protection, and emergency preparedness issues, independent of a specific nuclear plant design review. The application for an early site permit must address the safety and environmental characteristics of the site, and evaluate potential physical impediments to the development of an acceptable emergency plan or security plan. Additional detail may be submitted 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 findings on environmental protection issues in an environmental impact statement. The early site permit may also allow non-safety site preparation activities, subject to redress, before the issuance of a combined license. NRC will issue a *Federal Register* notice for a mandatory public hearing and the Advisory Committee on Reactor Safeguards will perform an independent safety review. The early site permit may be referenced in a construction permit or combined license application.

Under Subpart B of 10 CFR Part 52, NRC may certify and approve a standard plant design through a rulemaking, independent of a specific site. The issues resolved in a design certification have a more restrictive backfit requirement than issues resolved under other licenses. That is, a certified design cannot be modified by NRC unless the modification is necessary to meet the applicable regulations in effect during design certification, or to ensure adequate protection of public health and safety. An application for a combined license under 10 CFR Part 52 can incorporate by reference a design certification, an early site permit, or both. The advantage of this approach is that the issues resolved by rulemaking for design certification and those resolved during the early site permit hearing process are precluded from reconsideration at the combined license stage.

19.1.2 Experience and Examples

All currently operating reactors were licensed under the two-step process in 10 CFR Part 50. In each case, NRC scheduled its review of an operating licensing application for three years, but the actual timing varied, depending on such factors as the completeness of necessary information by the license applicant, resolution of safety issues, duration and acceptability of plant construction, and the duration of the public hearing. For the Millstone Unit 3 operating license, no public hearing was requested, and the operating license review was completed in three years. For the Comanche Peak operating license, resolution of substantial construction quality issues extended the time required to complete the operating license review.

19.2 Operational Limits and Conditions Are Defined and Revised

The license for each nuclear facility must contain technical specifications that set operational limits and conditions derived from the safety analyses, tests, and operational experience. 10 CFR 50.36, “Technical Specifications,” states 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 is required to identify items that directly apply to maintaining the integrity of the physical barriers that are designed to contain radioactive material. Specifically, 10 CFR 50.36 requires that the technical specifications must be derived from the analyses and evaluation in the safety analysis report. Licensees cannot change the technical specifications without prior NRC approval.

In 1992, NRC issued improved vendor-specific (e.g., Babcock and Wilcox, Westinghouse) standard technical specifications and periodically revises them on the basis of experience. NRC issued Revision 3 in June 2004.

NRC encourages licensees to use the improved standard technical specifications as the basis for plant-specific technical specifications. NRC also considers requests to adopt parts of the improved standard technical specifications, even if the licensee does not adopt all of the improvements. These parts will include all related requirements, and will normally be developed as line-item improvements. To date, 69 operating commercial nuclear plants have converted their technical specifications to the improved standard technical specifications.

A licensee may propose relocating the limiting conditions for operation that do not meet any of the criteria in 10 CFR 50.36 and their associated actions and surveillance requirements from technical specifications to licensee-controlled documents, such as the final safety analysis report. In such cases, the change is processed as a typical license amendment request.

Consistent with the Commission's policy statements on technical specifications and the use of PRA, NRC and the nuclear industry are developing risk-informed improvements to technical specifications. These improvements, or initiatives, are intended to maintain or improve safety while reducing unnecessary burden, and to bring technical specifications into congruence with the Agency's other risk-informed regulatory requirements [in particular, the risk management requirements of the Maintenance Rule in 10 CFR 50.65(a)(4)].

19.3 Approved Procedures

In the United States, operations, maintenance, inspection, and testing of a nuclear installation are conducted in accordance with approved procedures. Each nuclear facility is required to follow the quality assurance requirements in Appendix B to 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." The Quality Assurance Program is described in Article 13. Criterion V, "Instructions, Procedures, and Drawings," to Appendix B to 10 CFR Part 50, requires that licensees must establish measures to ensure that activities that affect quality will be prescribed by appropriate documented instructions, procedures, or drawings. Activities that affect quality include operation, maintenance, inspection, and testing of the facility.

Revision 3 to NRC Regulatory Guide 1.33, "Quality Assurance Program Requirements Operations," provides supplemental guidance. An appendix to this regulatory guide lists specific activities that should be covered by written procedures, such as administrative procedures; general plant procedures; operating procedures; and procedures for startup, operation, and shutdown of safety-related systems.

The rule that addresses the need to perform maintenance according to approved procedures is 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." 10 CFR 50.65(a)(4) requires licensees to assess and manage the increase in risk that may result from proposed maintenance activities.

19.4 Procedures for Responding to Anticipated Operational Occurrences and Accidents

The documents providing recommendations and guidance on procedures for responding to anticipated operational occurrences and accidents are NUREG-0737, "Clarification of TMI Action Plan Requirements"; NUREG-0737, Supplement 1, "Requirements for Emergency Response Capability"; and NUREG-0899, "Guidelines for the Preparation of Emergency Operating Procedures."

After the 1979 accident at TMI Unit 2, 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 in 1983. These two documents recommend that licensees develop procedures to cope with accidents and transients that are caused by initiating events analyzed in the final safety analysis report with multiple failures of equipment. If such failures were unmitigated, conditions of inadequate core cooling would exist. (An example of multiple failure events would be failure of main and auxiliary feedwater systems.)

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, NRC reviewed each plant using the guidance in NUREG-0800, Section 13.5.2, “Operating and Maintenance Procedures,” which focused on ensuring that the licensee’s process to develop the procedures was sound and well documented.

19.5 Availability of Engineering and Technical Support

NRC’s Reactor Oversight Process, discussed in Article 6, includes techniques to ensure that adequate engineering and technical support is available throughout the lifetime of a nuclear installation. Several of the inspection procedures focus on ensuring that adequate support programs are maintained. Licensees also report performance indicators. Depending on findings, NRC conducts additional inspections to focus upon the causes of the performance problems.

19.6 Incident Reporting

Requirements for incident reporting are specified in 10 CFR 50.72, “Immediate Notification Requirements for Operating Nuclear Power Reactors,” and in 10 CFR 50.73, “Licensee Event Report System.” NRC modified these rules in 1992 and 2000 to delete reporting requirements for some events that were determined to be of little or no safety significance. The modified rules continue to provide the Commission with reporting of significant events for which NRC may need to act to maintain or improve reactor safety or to respond to heightened public concern. The modified rules also better align requirements on event reporting with the type of information that NRC needs to carry out its safety mission. NRC issued Revision 2 to NUREG-1022, “Event Reporting Guidelines, 10 CFR 50.72 and 50.73,” concurrently with the rule changes.

Event reporting under these rules since 1984 has contributed significantly to focusing the attention of NRC and the nuclear industry on the lessons learned from operating experience to improve reactor safety. Over the years, decreasing trends in the number of reactor transients and significant events and improvements in reactor safety system performance have been evident.

19.7 Programs to Collect and Analyze Operating Experience

Operational safety data are reported to or identified by NRC in event notifications, licensee event reports, inspection reports, component failure reports, industry reports, reports on operational, safeguards, and security events, reports submitted under 10 CFR Part 21, “Reporting of Defects and Noncompliances,” and reports of operational experience at foreign facilities. NRC staff screens operational safety data for safety significance and generic implications. The staff responsible for generic communications systematically assesses and screens all nuclear power reactor-related events, reports, and data to determine their significance and need for further action. The staff also has the following five responsibilities:

(1) Develop, coordinate, and issue generic communications (such as regulatory issue summaries, generic letters, bulletins, and information notices) to alert the industry to safety concerns that are identified as a result of power reactor events and conditions.

(2) Identify the need for an augmented inspection team or incident investigation team response, and coordinate NRC's participation in establishing the teams.

(3) Coordinate briefings of operating events, and serve as a focal point for interfacing with NRC's regional offices and the industry for incoming reports.

(4) Maintain and administer an "on-call emergency officer" roster, and staff the daytime emergency officer functions.

(5) Develop and conduct programs for major team inspections at licensee facilities that require a number of engineering and operational specialties.

A group of NRC experts in event evaluation, risk assessment, and human factors reviews issues that have potentially generic implications. On the basis of these reviews, NRC responds by further analyzing the issue, preparing a generic communication, or simply closing out the issue. Typically, this group analyzes about 1,000 events per year, and follows up on 175 of those events. NRC rates these events in accordance with INES.

To alert the rest of the staff of significant issues, the staff rapidly disseminates operating experience through the use of a Web page, improved search capabilities in NRC's Agencywide Documents Access and Management System (ADAMS), and email distribution to various groups. NRC also actively participates in the IAEA's Incident Reporting System, and publishes events on its internal Web site to disseminate this international operating experience to the staff. NRC also issues about 40 generic communications each year to alert the industry to safety concerns. All of NRC's event-related reports can be found on the agency's public Web site at http://www.nrc.gov/reading-rm/doc-collections/event-status/, and generic communications can be found at http://www.nrc.gov/what-we-do/regulatory/gencomms.html.

Feedback of operational experience consists of carrying out the actions identified by analysis to maintain or improve licensees' safety and safeguards activities and NRC regulatory programs. Followup measures may include collecting additional relevant information, and recommending immediate or long-term changes. Specific followup action may involve changing facility operations or procedures, modifying facility components, systems, or structures, improving operator or staff training, changing regulations, regulatory guides, licensing review procedures or criteria, the inspection program, and research or risk assessment activities, or issuing a generic communication.

Recent activities include chartering an operating experience task force to evaluate the Agency's reactor operating experience program and to recommend improvements that address the recommendations of the Davis-Besse Lessons Learned Task Force. Some of the recommendations are to establish a central clearinghouse for operating experience, improve mechanisms for technical staff to identify potential safety issues and provide feedback to inspectors, performing additional evaluations to identify trends, recurring events, or safety issues and improving the follow-up to verify adequate resolution of the issues of concern.

The staff uses the Accident Sequence Precursor Program, described in Article 6, to analyze events using probabilistic risk assessment techniques to determine conditional core damage probabilities. This program quantitatively evaluates operational experience, and serves as one of several tools to ensure that important operating lessons are not overlooked. In addition, the staff uses the Reactor Oversight Process to analyze risk-significant events or conditions to ensure that plants are operated within prescribed safety limits.

19.8 Radioactive Waste

NRC has regulations and guidance for nuclear power reactor licensees to help ensure that low-level radioactive waste is safely managed and disposed. Onsite low-level waste must be managed in accordance with the provisions in NRC regulations codified in 10 CFR Parts 20 and 50. For example, Subpart K of 10 CFR Part 20, paragraph 20.2004 deals with treatment or disposal by incineration. In addition, Generic Letter 81-38, "Storage of Low-Level Radioactive Wastes at Power Reactor Sites," provides guidance on measures for ensuring the safe storage of low-level waste.

Notwithstanding the preceding regulations and guidance, the economics of waste disposal in the U.S. have encouraged practices to minimize radioactive waste. In the past 10 years, disposal costs have risen by a factor of about 10, and volumes of waste produced have decreased by roughly the same amount. Nuclear power reactors now generate only small amounts (1,000 – 2,000 cubic feet) of operational waste each year.

For storage, waste is put into a form that is stable and safe, and minimizes the likelihood that it will migrate (i.e., wastes in liquid form may be solidified). Waste that is put into storage, is in a form that is suitable for disposal, or at least a form that can be made suitable for disposal. For designing and operating low-level waste disposal facilities, NRC has detailed regulations in 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste."

The U.S. Government addresses the spent fuel and radioactive waste programs, including high-level waste, in detail in its National Report to comply with reporting requirements of the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. This report is available on the U.S. Department of Energy's Environmental Management Web site at http://web.em.doe.gov/integrat/National_Report_05-02-03_1.pdf.

APPENDICES

APPENDIX A: NRC MAJOR MANAGEMENT CHALLENGES FOR THE FUTURE

By law, the Inspector General of each Federal agency (described in Article 8) is to describe what he or she considers to be the most serious management and performance challenges facing the agency and assess the agency's progress in addressing those challenges. Accordingly, NRC's Inspector General prepared his annual assessment of the major management challenges confronting the agency. The full report can be found on NRC's public Web site at http://www.nrc.gov/reading-rm/doc-collections/insp-gen/2004/04-01.pdf. In his assessment, the Inspector General defined serious management challenges as "mission-critical areas or programs that have the potential for a perennial weakness or vulnerability that, without substantial management attention, would seriously impact agency operations or strategic goals." The most serious management challenges facing NRC may be, but are not necessarily, areas that are problematic for the agency. The challenges, as identified, represent critical areas or difficult tasks that warrant high-level management attention. In its 2003 report, the Inspector General considered the following nine management challenges to be the most serious. They are not ranked in order of importance.

Challenge 1: Protection of nuclear material used for commercial purposes. Over the past fiscal year, NRC has been overseeing the implementation of enhanced security measures and determining the appropriate level of security needed to protect commercial nuclear power facilities. NRC continues to work with the U.S. Department of Homeland Security (created as a result of the terrorist attacks on September 11, 2001), as well as other Federal agencies and State and local law enforcement and emergency planning officials, to ensure an integrated approach to protecting these critical facilities.

NRC's requirements for nuclear plant security are based on the need to protect the public from exposure to radioactive release caused by acts of sabotage and the need to protect against theft of special nuclear material. Therefore, NRC has taken a number of steps to strengthen security at NRC-licensed facilities. These steps include issuing requirements to enhance facility security, training of licensee security officers and addressing security personnel fatigue at nuclear reactor sites to increase capability of licensees to detect and respond to threats. NRC also required plants to enhance access controls to prevent entry of unauthorized persons to nuclear facilities. The staff continually assesses the design-basis threat for nuclear power plants to ensure that it adequately addresses credible threats.

NRC has also made organizational changes to further strengthen its security response capabilities. In April 2002, the Commission established the Office of Nuclear Security and Incident Response to consolidate NRC's security, safeguards, and incident response functions. This office restarted security exercises at operating nuclear power reactor facilities, which had been discontinued after September 11, 2001. In June 2003 established the senior management position of Deputy Executive Director for Homeland Protection and Preparedness. This new Deputy is responsible for working across agency lines of authority to resolve homeland protection and preparedness issues. In January 2004, the Emergency Perparedness Program Office was established to increase attention on NRC activities that affect emergency preparedness.

Challenge 2: Protection of information. NRC employees often generate and work on information that is sensitive and needs to be protected. Such information can be sensitive unclassified information or classified national security information that is contained in written documents and electronic databases. Recent audits found that improvements were needed in the areas of administrative procedures, information technology controls, and physical security controls.

In response to those findings, NRC has taken various steps to protect sensitive information from inappropriate disclosure, such as adding warning messages to NRC's electronic recordkeeping system, known as the Agencywide Documents Access and Management System (ADAMS), which houses vast quantities of agency information, to highlight to the staff that the information is sensitive.

Challenge 3: Development and implementation of a risk-informed and performance-based regulatory oversight approach. NRC faces numerous challenges in making its regulatory framework more risk-informed. In April 2000, NRC implemented the Reactor Oversight Process to move toward a more risk-informed regulatory philosophy. NRC is currently working to further its risk-informed approach by revising its nuclear plant regulations. NRC has progressed over the past 10 years in implementing risk-informed regulation; however, the agency still has a long way to go to fully implement the process. NRC expects that risk-informed regulation will continue to be a major area of focus.

Currently, NRC is studying potential new performance indicators to see if it can establish a stronger connection to risk in the Reactor Oversight Process. NRC is also seeking performance indicators that will help to identify emergent problems so as to avoid them, rather than applying performance indicators that merely confirm existing problems.

Challenge 4: Ability to modify regulatory processes to meet changing external demands. NRC faces numerous challenges related to the changing regulatory and business environment. The increased demand for electric power in the United States has created challenges pertaining to such areas as reactor license renewals, license amendment requests to increase reactor power output, and new plant designs.

Reactor License Renewal The improved performance of nuclear power plants over the past decade has caused many of NRC's licensees to consider renewing their licenses rather than decommissioning their plants when the licenses expire. Approximately half of the operating nuclear reactor units in the U.S. are currently involved in some stage of the license renewal process. To regulate this activity, NRC established a license renewal inspection program to verify information submitted in the renewal applications. This program also includes an agency safety evaluation and environmental impact analysis.

Applications To Increase Power Output NRC has completed many requests to increase power output. Such requests involve complex, technical issues and NRC's review of the application to ensure safe operation. NRC considers such requests as high priority that requires input from many technical areas of the agency. It has completed a number of these reviews more quickly than the agency's estimate of 18 months needed to accomplish the task.

New Plant Designs New proposals for nuclear power plant design are emerging with the maturation of the nuclear power industry. Numerous reactor designs have been submitted for NRC review; three are being actively pursued at this time. The staff is improving the infrastructure to ensure that tools, information, and regulatory processes are in place for the efficient, effective, and realistic review of these applications and to ensure that an appropriate level of safety is maintained.

Challenge 5: Acquisition and implementation of information resources. Federal agencies' acquisition and implementation of information resources are crucial to support mission-critical operations and provide more effective and cost-efficient Government services to the public.

NRC relies on a wide variety of information systems to help fulfill its responsibilities and support its business flow. Like other Federal agencies, NRC continues to struggle in its effort to obtain a good return on these investments. In recent years, NRC has created large databases of publicly available information, including ADAMS and the agency's public Web site. In addition, NRC recently issued a final rule that allows licensees and others to electronically submit almost all correspondence by CD-ROM, email, or fax.

Challenge 6: Administration of all aspects of financial management. NRC must be a prudent steward of its fiscal resources through sound financial management. Sound financial management includes the production of timely, useful, and reliable financial information to support agency management; an effective cost accounting system; well-developed strategic planning; and an integrated method for planning, budgeting, and assessing performance to better enable NRC to align programs with outcomes. Sound financial management also includes how the agency procures goods and services.

NRC has been tightening controls over financial management processes. For example, NRC hired a consulting firm to assist the agency to more fully address the challenges associated with managerial cost accounting.

Challenge 7: Communication with external stakeholders throughout NRC's regulatory activities. To maintain public trust and confidence, NRC must be viewed as an independent, open, efficient, clear, and reliable regulator. Toward this end, NRC needs to provide its diverse group of external stakeholders with clear, accurate, and timely information about the agency's regulatory process. This is a challenging task because of the highly technical nature of NRC's operations, the sensitivity of its information, and the balance the agency must maintain to remain independent. The challenge is to afford all stakeholders, including the public, with appropriate and meaningful access to the regulatory process while protecting sensitive security and safeguards information from unauthorized access.

In June 2003, NRC created a task force to develop strategies for comprehensive and effective communication with external stakeholders. The task force determined that NRC's effectiveness in communicating with its stakeholders varies and that while, in many cases, the agency is communicating reasonably well with its stakeholders, there is room for significant improvement. The task force made 10 recommendations to improve NRC's external communications.

To provide integrated leadership and direction for external communications, the Chairman established the position of Director of Communications to report directly to his office. This Director is responsible for enhancing the effectiveness of NRC's communications with the public, the media, and the Congress.

NRC has implemented activities to improve the public meeting process such as creating a page on its public Web site that provides information on public meetings, using public meeting feedback forms, and creating a public meeting checklist to provide guidance to the staff on planning and conducting successful public meetings.

Challenge 8: Intra-agency communication (up, down, and across organizational lines). Internal communication is a fundamental and necessary aspect of conducting agency business.

Information is the key resource that links NRC managers with staff, the organization, and other internal stakeholders, enabling them to do their work cooperatively and efficiently in a coordinated manner. Results from the Inspector General's Safety Culture and Climate Survey, December 2002, showed that most NRC employees believe the agency has not established a climate where traditional ways of doing business can be challenged, or innovative ideas can fail, without penalty.

NRC has taken steps to improve its internal communications over the past year. Responding to the Safety Culture and Climate Survey findings, NRC established a safety culture and climate task force to work on improving internal communications. This task force issued a report with recommendations to the agency on how to address communications issues. The agency communicates internally by using electronic updates from upper management to the entire staff. NRC also recently redesigned its internal Web site to facilitate information access and service delivery. In addition, the Commission continues to use the annual All Employees Meeting as an important and effective tool for direct two-way communication between the Commission and agency employees. For more details, see Article 10.

Challenge 9: Managing human capital. NRC must have a dynamic, diverse workforce with the appropriate knowledge, skills, and abilities to achieve its mission. Consequently, the agency has identified human capital management as a major challenge and a potential high-risk area. The demands include declining workforce numbers, institutional knowledge, critical skills, and new workforce demographic trends (e.g., aging workforce), as well as increasing market competition for a shrinking labor pool. Thirty percent of the Federal workforce will be eligible to retire in five years, and an additional 20 percent could seek early retirement. This does not mean that 50 percent of Government employees will retire in the short-term, but rather that Federal agencies must start planning for the workforce of the future.

NRC developed a set of strategic human capital management initiatives to mitigate the expected loss of personnel with the technical competencies necessary for the future. Some of these initiatives include early replacement hiring, recruitment bonuses, and undergraduate fellowship programs. NRC also strives to ensure that 20 percent of new employees are hired into entry-level positions. In addition, the agency is identifying gaps and developing strategies, such as increased recruiting efforts and training budgets, that will maintain the essential core scientific and technical capacities. With these efforts, NRC believes that it will successfully identify its critical skill needs and hire, develop, motivate, and retain the employees who possess the skills needed to support the agency's strategic goals and outcomes in the future.

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APPENDIX C: ABBREVIATIONS

ADAMS	Agencywide Documents Access and Management System (NRC)
ALARA	as low as is reasonably achievable
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ATHEANA	A Technique for Human Event Analysis
BPV	Boiler and Pressure Vessel
BWR	boiling-water reactor
CFR	*Code of Federal Regulations*
CNRA	Committee on Nuclear Regulatory Activities
CRGR	Committee To Review Generic Requirements (NRC)
CSNI	Committee on the Safety of Nuclear Installations
DOE	Department of Energy (U.S.)
EDO	Executive Director for Operations
EPA	Environmental Protection Agency (U.S.)
EPRI	Electric Power Research Institute
ERDA	Energy Research and Development Administration (U.S.)
FEMA	Federal Emergency Management Agency (U.S.)
FTE	full-time equivalent
FY	Fiscal Year
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
INES	International Nuclear Event Scale
IPE	individual plant examination
IRRT	International Regulatory Review Team
ISAP	Integrated Safety Assessment Program
ISO	International Organization for Standardization
ITAAC	inspections, tests, analyses, and acceptance criteria
NCRP	National Council on Radiation Protection and Measurements
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission (U.S.)
NRR	Office of Nuclear Reactor Regulation (NRC)
OM	Operation and Maintenance
OSART	Operational Safety Assessment Review Team
PRA	probabilistic risk assessment
PSR	periodic safety review
PWR	pressurized-water reactor
QA	quality assurance
RCC-M	French Nuclear Construction Code
SAPHIRE	Systems Analysis Programs for Hands-On Integrated Reliability Evaluation
SEP	systematic evaluation program
TMI	Three Mile Island
TVA	Tennessee Valley Authority
WENRA	Western European Nuclear Regulators' Association

APPENDIX D: ACKNOWLEDGMENTS

Contributors to this report included the NRC technical and regulatory experts listed below. The project manager was Thomas Hiltz; the preparer of the report was Merrilee Banic.

MaryAnn Ashley
Goutam Bagchi
Steven Baggett
Merrilee Banic
Daniel Barss
Tom Blount
James Bongarra
Tom Boyce
June Cai
Gene Carpenter
Douglas Coe
Steve Crockett
Robert Denning
David Desaulniers
Ira Dinitz
Joseph Donoghue
Laura Dudes
Michael Dusaniwskyj
Robert Elliot
Jack Foster
Rani Franovich
Paul Gill
Giorgio Gnugnoli
Clare Goodman
Karen Henderson
Donald Harrison
Thomas Hiltz
Charles Hinson
Steve Hoffman
Kahtan Jabbour
David Jaffe
Robert Kahler
Rebecca Karas
Betsy Keeling
Steve Kementowicz
Mark Kowal
Steve LaVie
Paul Lain
Jay Lee
James Lyons
Stewart MaGruder
Roy Mathew
Tim McGinty
Richard McIntyre
Eileen McKenna
Tony McMurtray
Mary Muessle
Renee Pedersen
Roger Pedersen
Richard Pelton
Terry Reis
Brett Rini
Trip Rothchild
Sami Sherbini
David Skeen
Shawn Smith
Andrew Szukiewicz
John Stang
Dale Thatcher
Robert Tjader
David Trimble
Rosetta Virgilio
Mike Webb
Sunil Weerakoddy
Eric Weiss
Jim Wigginton
Jerry Wilson
Barry Zalcman

ANNEX 1: U.S. COMMERCIAL NUCLEAR POWER REACTORS

SOURCE: NRC and licensee data as compiled by NRC

RELEVANT ARTICLE: Introduction and Article 6

U.S. Commercial Nuclear Power Reactors

Unit Operating Utility	Type NSSS AE Constructor	Licensed MWt	Net MDC	CP Issued OL Issued Comm. Op Exp. Date	1995-2000* Average Capacity Factors (Percent)
Arkansas Nuclear 1 Entergy Nuclear South	PWR-DRYAMB B&W LLP BECH BECH	2568	0883	12/06/1968 05/21/1974 12/19/1974 05/20/2034	99.0 82.6 91.7 87.3 93.9 89.7
Arkansas Nuclear 2 Entergy Nuclear South	PWR-DRYAMB COMB CE BECH BECH	2815	0912	12/06/1972 09/01/1978 03/26/1980 07/17/2018	92.6 86.9 82.8 69.9 105.3 106.5
Beaver Valley 1 First Energy Nuclear Operating Company	PWR-DRYSUB WEST 3LP S&W S&W	2689	0821	06/26/1970 07/02/1976 10/01/1976 01/29/2016	56.3 33.2 86.1 82.7 83.3 97.2
Beaver Valley 2 First Energy Nuclear Operating Company	PWR-DRYSUB WEST 3LP S&W S&W	2689	0831	05/03/1974 08/14/1987 11/17/1987 05/27/2027	85.7 16.9 80.1 86.5 98.8 90.7
Braidwood 1 Exelon	PWR-DRYAMB WEST 4LP S&L CWE	3586	1161	12/31/1975 07/02/1987 07/29/1988 10/17/2026	83.9 78.6 101.0 96.4 93.4 104.3

U.S. Commercial Nuclear Power Reactors (Cont.)

Unit Operating Utility	Type NSSS AE Constructor	Licensed MWt	Net MDC	CP Issued OL Issued Comm. Op Exp. Date	1997-2002* Average Capacity Factors (Percent)
Braidwood 2 Exelon	PWR-DRYAMB WEST 4LP S&L CWE	3586	1154	12/31/1975 05/20/1988 10/17/1988 12/18/2027	85.5 97.4 92.0 98.4 98.2 93.5
Browns Ferry 1 Tennessee Valley Authority	BWR-MARK1 GE 4 TVA TVA	3293	1118	05/10/1967 12/20/1973 08/01/1974 12/20/2013	0.0 0.0 0.0 0.0 0.0 0.0
Browns Ferry 2 Tennessee Valley Authority	BWR-MARK1 GE 4 TVA TVA	3458	1155	05/10/1967 08/02/1974 03/01/1975 06/28/2014	89.7 98.9 89.1 99.1 85.9 91.0
Browns Ferry 3 Tennessee Valley Authority	BWR-MARK 1 GE TVA TVA	3458	1155	07/31/1968 08/18/1976 03/01/1977 07/02/2016	91.4 80.8 99.4 92.6 100.1 94.6
Brunswick 1 Progress Energy	BWR-MARK 1 GE 4 UE&C BRRT	2558	0872	02/07/1970 11/12/1976 03/18/1977 09/08/2016	102.1 83.6 97.4 93.7 101.7 93.2
Brunswick 2 Progress Energy	BWR-MARK 1 GE 4 UE&C BRRT	2558	0811	02/07/1970 12/27/1974 11/03/1975 12/27/2014	91.7 95.4 85.8 99.0 92.1 99.6

U.S. Commercial Nuclear Power Reactors (Cont.)

Unit Operating Utility	Type NSSS AE Constructor	Licensed MWt	Net MDC	CP Issued OL Issued Comm. Op Exp. Date	1997-2002* Average Capacity Factors (Percent)
Byron 1 Exelon	PWR-DRYAMB WEST 4LP S&L CWE	3411	1105	12/31/1975 02/14/1985 09/16/1985 10/31/2024	74.0 77.6 92.0 95.7 102.0 96.5
Byron 2 Exelon	PWR-DRYAMB WEST 4LP S&L CWE	3586	1131	12/31/1975 01/30/1987 08/21/1987 11/06/2026	94.0 85.7 94.8 103.1 99.2 96.3
Callaway Ameren	PWR-DRYAMB WEST 4LP BECH DANI	3565	1125	04/16/1976 10/18/1984 12/19/1984 10/18/2024	90.9 84.8 87.2 101.1 85.1 85.1
Calvert Cliffs 1 Constellation Energy Group	PWR-DRYAMB COMB CE BECH BECH	2700	0825	07/07/1969 07/31/1974 05/08/1975 07/31/2034	97.9 81.9 96.8 89.0 103.2 64.3
Calvert Cliffs 2 Constellation Energy Group	PWR-DRYAMB COMB CE BECH BECH	2700	0835	07/07/1969 11/30/1976 04/01/1977 08/13/2036	81.2 97.7 86.6 100.8 84.8 102.3
Catawba 1 Duke Power Co.	PWR-ICECND WEST 4LP DUKE DUKE	3411	1129	08/07/1975 01/17/1985 06/29/1985 12/06/2044	92.8 88.2 91.7 90.0 100.9 95.9

U.S. Commercial Nuclear Power Reactors (Cont.)

Unit Operating Utility	Type NSSS AE Constructor	Licensed MWt	Net MDC	CP Issued OL Issued Comm. Op Exp. Date	1997-2002* Average Capacity Factors (Percent)
Catawba 2 Duke Power Co.	PWR-ICECND WEST 4LP DUKE DUKE	3411	1129	08/07/1975 05/15/1986 08/19/1986 02/24/2046	86.8 85.2 89.5 90.6 86.7 102.9
Clinton Exelon	BWR-MARK 3 GE 6 S&L BALD	2894	1022	02/24/1976 04/17/1987 11/24/1987 09/29/2026	0.0 0.0 57.7 84.3 96.7 85.5
Columbia Generating Station Energy Northwest	BWR-MARK 2 GE 5 B&R BECH	3486	1107	03/19/1973 04/13/1984 12/13/1984 12/20/2023	63.0 68.1 62.8 88.5 85.1 92.6
Comanche Peak 1 Texas Utilities Electric Co.	PWR-DRYAMB WEST 4LP G&H BRRT	3458	1150	12/19/1974 04/17/1990 08/13/1990 02/08/2030	94.1 86.2 85.4 95.2 83.8 87.3
Comanche Peak 2 Texas Utilities Electric Co.	PWR-DRYAMB WEST 4LP BECH BRRT	3458	1150	12/19/1974 04/06/1993 08/03/1993 02/02/2033	80.0 95.3 86.9 87.8 98.1 87.3
Cooper Nebraska Public Power District	BWR-MARK 1 GE 4 B&R B&R	2381	0764	06/04/1968 01/18/1974 07/01/1974 01/18/2014	81.5 75.2 97.3 70.6 77.8 94.4

U.S. Commercial Nuclear Power Reactors (Cont.)

Unit Operating Utility	Type NSSS AE Constructor	Licensed MWt	Net MDC	CP Issued OL Issued Comm. Op Exp. Date	1997-2002* Average Capacity Factors (Percent)
Crystal River 3 Florida Power Corp.	PWR-DRYAMB B&W LLP GIL JONES	2544	0834	09/25/1968 01/28/1977 03/13/1977 12/03/2016	0.0 88.2 88.9 97.2 89.2 99.9
Davis-Besse First Energy Nuclear Operating Company	PWR-DRYAMB B&W LLP BECH BECH	2772	0882	03/24/1971 04/22/1977 07/31/1978 04/22/2017	93.9 78.1 96.4 87.4 99.5 12.0
D.C. Cook 1 Indiana/Michigan Power Co.	PWR-ICECND WEST 4LP AEP AEP	3250	1000	03/25/1969 10/25/1974 08/28/1975 10/25/2014	61.6 95.3 51.9 0.0 0.0 1.5
D.C. Cook 2 Indiana/Michigan Power Co.	PWR-ICECND WEST 4LP AEP AEP	3411	1060	03/25/1969 12/23/1977 07/01/1978 12/23/2037	63.3 0.0 0.0 51.4 85.8 82.8
Diablo Canyon 1 Pacific Gas & Electric Co.	PWR-DRYAMB WEST 4LP PG&E PG&E	3338	1100	04/23/1968 11/02/1984 05/07/1985 09/22/2021	87.1 98.0 87.5 83.3 99.8 74.0
Diablo Canyon 2 Pacific Gas & Electric Co.	PWR-DRYAMB WEST 4LP PG&E PG&E	3411	1100	12/09/1970 08/26/1985 03/13/1986 04/26/2025	93.3 84.5 88.7 96.2 90.9 97.5

U.S. Commercial Nuclear Power Reactors (Cont.)

Unit Operating Utility	Type NSSS AE Constructor	Licensed MWt	Net MDC	CP Issued OL Issued Comm. Op Exp. Date	1997-2002* Average Capacity Factors (Percent)
Dresden 2 Exelon	BWR-MARK 1 GE 3 S&L UE&C	2957	0850	01/10/1966 02/20/1991 06/09/1970 12/22/2029	82.5 79.1 92.1 101.3 89.8 101.1
Dresden 3 Exelon	BWR-MARK 1 GE 3 S&L UE&C	2527	0850	10/14/1966 03/02/1971 11/16/1971 01/12/2031	59.5 88.2 90.6 93.7 95.5 81.4
Duane Arnold Nuclear Management Co.	BWR-MARK 1 GE 4 BECH BECH	1658	0565	06/22/1970 02/22/1974 02/01/1975 02/21/2014	91.2 82.3 80.1 97.5 77.9 92.5
Edwin I. Hatch 1 Southern Nuclear Operating Co.	BWR-MARK 1 GE 4 BECH GPC	2763	0924	09/30/1969 10/13/1974 12/31/1975 08/06/2034	85.7 96.5 81.1 84.5 99.2 88.4
Edwin I. Hatch 2 Southern Nuclear Operating Co.	BWR-MARK 1 GE 4 BECH GPC	2763	0924	12/27/1972 06/13/1978 09/05/1979 06/13/2038	84.2 80.6 94.4 89.5 85.6 97.4
Fermi 2 Detroit Edison Co.	BWR-MARK 1 GE 4 S&L DANI	3430	1089	09/26/1972 07/15/1985 01/23/1988 03/20/2025	63.6 67.8 100.3 86.2 89.8 97.5

U.S. Commercial Nuclear Power Reactors (Cont.)

Unit Operating Utility	Type NSSS AE Constructor	Licensed MWt	Net MDC	CP Issued OL Issued Comm. Op Exp. Date	1997-2002* Average Capacity Factors (Percent)
Fort Calhoun Omaha Public Power District	PWR-DRYAMB COMB CE GHDR GHDR	1524	0508	06/07/1968 08/09/1973 09/26/1973 08/09/2033	91.2 77.8 85.6 92.8 84.2 91.0
Ginna Constellation Energy Group	PWR-DRYAMB WEST 2LP GIL BECH	1520	0480	04/25/1966 12/10/1984 07/01/1970 09/18/2029	92.6 104.1 84.0 90.5 101.9 91.4
Grand Gulf 1 Entergy Nuclear South	BWR-MARK 3 GE 6 BECH BECH	3833	1250	09/04/1974 11/01/1984 07/01/1985 11/01/2024	102.9 82.0 79.9 100.6 93.6 95.1
H.B. Robinson 2 Progress Energy	PWR-DRYAMB WEST 3LP EBSO EBSO	2300	0710	04/13/1967 09/23/1970 03/07/1971 07/31/2030	103.6 87.9 95.0 104.0 92.2 93.7
Hope Creek 1 Public Service Electric & Gas Co.	BWR-MARK1 GE 4 BECH BECH	3339	1049	11/04/1974 07/25/1986 12/20/1986 04/11/2026	70.9 92.3 85.3 80.3 87.8 96.2

U.S. Commercial Nuclear Power Reactors (Cont.)

Unit Operating Utility	Type NSSS AE Constructor	Licensed MWt	Net MDC	CP Issued OL Issued Comm. Op Exp. Date	1997-2002* Average Capacity Factors (Percent)
Indian Point 2 Entergy Nuclear Northeast	PWR-DRYAMB WEST 4LP UE&C WDCO	3071	0951	10/14/1966 09/28/1973 08/01/1974 09/28/2013	38.4 23.0 88.5 12.1 93.5 90.7
Indian Point 3 Entergy Nuclear Northeast	PWR-DRYAMB WEST 4LP UE&C WDCO	3025	0979	08/13/1969 04/05/1976 08/30/1976 12/15/2015	51.3 89.8 86.0 99.5 93.9 98.3
James A. FitzPatrick Entergy Nuclear Northeast	BWR-MARK 1 GE 4 S&W S&W	2536	0813	05/20/1970 10/17/1974 07/28/1975 10/17/2014	94.7 73.2 93.5 84.4 99.6 92.6
Joseph M. Farley 1 Southern Nuclear Operating Co.	PWR-DRYAMB WEST 3LP SSI DANI	2775	0888	08/16/1972 06/25/1977 12/01/1977 06/25/2017	75.2 78.9 97.4 71.5 87.6 99.0
Joseph M. Farley 2 Southern Nuclear Operating Co.	PWR-DRYAMB WEST 3LP SSI BECH	2775	0842	08/16/1972 03/31/1981 07/30/1981 03/31/2041	101.1 84.7 71.7 100.0 78.2 87.6
Kewaunee Nuclear Management Co.	PWR-DRYAMB WEST 2LP PSE PSE	1673	0526	08/06/1968 12/21/1973 06/16/1974 12/21/2013	52.8 78.4 98.8 82.7 77.3 99.8

U.S. Commercial Nuclear Power Reactors (Cont.)

Unit Operating Utility	Type NSSS AE Constructor	Licensed MWt	Net MDC	CP Issued OL Issued Comm. Op Exp. Date	1997-2002* Average Capacity Factors (Percent)
La Salle 1 Exelon	BWR-MARK 2 GE 5 S&L CWE	3489	1111	09/10/1973 08/13/1982 01/01/1984 05/17/2022	0.0 30.8 88.3 99.6 101.2 91.7
La Salle 2 Exelon	BWR-MARK 2 GE 5 S&L CWE	3489	1111	09/10/1973 03/23/1984 10/19/1984 12/16/2023	0.0 0.0 73.1 92.4 99.5 92.4
Limerick 1 Exelon	BWR-MARK 2 GE 4 BECH BECH	3458	1134	06/19/1974 08/08/1985 02/01/1986 10/26/2024	95.3 77.6 98.1 89.5 101.2 93.5
Limerick 2 Exelon	BWR-MARK 2 GE 4 BECH BECH	3458	1134	06/19/1974 08/25/1989 01/08/1990 06/22/2029	85.0 93.5 85.0 99.0 92.3 100.8
McGuire 1 Duke Power Co.	PWR-ICECND WEST 4LP DUKE DUKE	3411	1100	02/23/1973 07/08/1981 12/01/1981 06/12/2041	70.8 80.9 89.1 103.4 90.1 94.4
McGuire 2 Duke Power Co.	PWR-ICECND WEST 4LP DUKE DUKE	3411	1100	02/23/1973 05/27/1983 03/01/1984 03/03/2043	67.2 92.1 89.2 87.5 102.5 92.5

U.S. Commercial Nuclear Power Reactors (Cont.)

Unit Operating Utility	Type NSSS AE Constructor	Licensed MWt	Net MDC	CP Issued OL Issued Comm. Op Exp. Date	1997-2002* Average Capacity Factors (Percent)
Millstone 2 Dominion	PWR-DRYAMB COMB CE BECH BECH	2700	0871	12/11/1970 09/26/1975 12/26/1975 07/31/2035	0.0 0.0 57.9 81.7 95.6 81.3
Millstone 3 Dominion	PWR-DRYSUB WEST 4LP S&W S&W	3411	1130	08/09/1974 01/31/1986 04/23/1986 11/25/2045	0.0 34.0 82.7 99.9 82.1 88.3
Monticello Nuclear Management Co.	BWR-MARK 1 GE 3 BECH BECH	1775	0578	06/19/1967 01/09/1981 06/30/1971 09/08/2010	76.8 82.4 91.8 83.6 76.5 99.0
Nine Mile Point 1 Constellation Energy Group	BWR-MARK 1 GE 2 NIAG S&W	1850	0565	04/12/1965 12/26/1974 12/01/1969 08/22/2009	54.5 87.9 72.0 94.3 88.5 99.1
Nine Mile Point 2 Constellation Energy Group	BWR-MARK 2 GE5 S&W S&W	3467	1120	06/24/1974 07/02/1987 03/11/1988 10/31/2026	91.7 71.4 89.3 81.1 90.3 85.8
North Anna 1 Dominion	PWR-DRYSUB WEST 3LP S&W S&W	2893	0925	02/19/1971 04/01/1978 06/06/1978 04/01/2018	91.5 90.5 103.8 92.0 87.9 100.8

U.S. Commercial Nuclear Power Reactors (Cont.)

Unit Operating Utility	Type NSSS AE Constructor	Licensed MWt	Net MDC	CP Issued OL Issued Comm. Op Exp. Date	1997-2002 Average Capacity Factors (Percent)
North Anna 2 Dominion	PWR-DRYSUB WEST 3LP S&W S&W	2893	0917	02/19/1971 08/21/1980 12/14/1980 08/21/2040	99.7 89.0 91.4 101.8 74.4 68.6
Oconee 1 Duke Power Co.	PWR-DRYAMB B&W LLP DBDB DUKE	2568	0846	11/06/1967 02/06/1973 07/15/1973 02/06/2033	43.0 77.1 83.8 84.9 94.0 89.2
Oconee 2 Duke Power Co.	PWR-DRYAMB B&W LLP DBDB DUKE	2568	0846	11/06/1967 10/06/1973 09/09/1974 10/06/2033	79.2 72.1 84.4 100.9 90.2 89.2
Oconee 3 Duke Power Co.	PWR-DRYAMB B&W LLP DBDB DUKE	2568	0846	11/06/1967 07/19/1974 12/16/1974 12/16/2034	62.7 79.8 99.4 88.5 72.8 100.7
Oyster Creek Exelon	BWR-MARK 1 GE 2 B&R B&R	1930	0619	12/15/1964 07/02/1991 12/01/1969 12/15/2009	93.6 74.3 99.4 71.9 96.4 92.8
Palisades Nuclear Management Co.	PWR-DRYAMB COMB CE BECH BECH	2565	0730	03/14/1967 02/21/1991 12/31/1971 03/24/2011	90.8 80.0 80.2 89.6 36.8 99.6

U.S. Commercial Nuclear Power Reactors (Cont.)

Unit Operating Utility	Type NSSS AE Constructor	Licensed MWt	Net MDC	CP Issued OL Issued Comm. Op Exp. Date	1997-2002* Average Capacity Factors (Percent)
Palo Verde 1 Arizona Public Service Co.	PWR-DRYAMB COMB CE80 BECH BECH	3876	1227	05/25/1976 06/01/1985 01/28/1986 12/31/2024	98.6 87.4 88.7 100.4 87.8 89.1
Palo Verde 2 Arizona Public Service Co.	PWR-DRYAMB COMB CE80 BECH BECH	3990	1227	05/25/1976 04/24/1986 09/19/1986 12/09/2025	85.6 101.8 90.0 87.2 92.6 92.0
Palo Verde 3 Arizona Public Service Co.	PWR-DRYAMB COMB CE80 BECH BECH	3876	1230	05/25/1976 11/25/1987 01/08/1988 03/25/2027	86.5 87.6 100.3 90.3 83.9 102.0
Peach Bottom 2 Exelon	BWR-MARK 1 GE 4 BECH BECH	3458	1116	01/31/1968 12/14/1973 07/05/1974 08/08/2033	100.0 75.9 98.8 88.8 97.9 92.3
Peach Bottom 3 Exelon	BWR-MARK 1 GE 4 BECH BECH	3458	1093	01/31/1968 07/02/1974 12/23/1974 07/02/2034	79.0 90.1 89.4 99.5 89.0 100.8
Perry 1 First Energy Nuclear Operating Company	BWR-MARK 3 GE 6 GIL KAIS	3558	1235	05/03/1977 11/13/1986 11/18/1987 03/18/2026	80.2 96.7 89.8 93.9 71.6 92.2

U.S. Commercial Nuclear Power Reactors (Cont.)

Unit Operating Utility	Type NSSS AE Constructor	Licensed MWt	Net MDC	CP Issued OL Issued Comm. Op Exp. Date	1997-2002* Average Capacity Factors (Percent)
Pilgrim 1 Entergy Nuclear Northeast	BWR-MARK 1 GE 3 BECH BECH	1998	0653	08/26/1968 09/15/1972 12/01/1972 06/08/2012	73.4 96.9 76.2 93.7 89.9 100.9
Point Beach 1 Nuclear Management Co.	PWR-DRYAMB WEST 2LP BECH BECH	1519	0485	07/19/1967 10/05/1970 12/21/1970 10/05/2030	19.4 54.9 78.4 92.3 82.9 89.0
Point Beach 2 Nuclear Management Co.	PWR-DRYAMB WEST 2LP BECH BECH	1519	0485	07/25/1968 03/08/1973 10/01/1972 03/08/2033	19.0 77.5 80.0 78.4 96.8 89.3
Prairie Island 1 Nuclear Management Co.	PWR-DRYAMB WEST 2LP FLUR NSP	1650	0513	06/25/1968 04/05/1974 12/16/1973 08/09/2013	78.4 89.7 89.0 98.9 79.6 95.6
Prairie Island 2 Nuclear Management Co.	PWR-DRYAMB WEST 2LP FLUR NSP	1650	0512	06/25/1968 10/29/1974 12/21/1974 10/29/2014	81.2 78.6 100.5 91.1 93.4 93.9
Quad Cities 1 Exelon	BWR-MARK 1 GE 3 S&L UE&C	2957	0855	02/15/1967 12/14/1972 02/18/1973 12/14/2032	82.6 42.1 94.1 91.3 99.6 76.2

U.S. Commercial Nuclear Power Reactors (Cont.)

Unit Operating Utility	Type NSSS AE Constructor	Licensed MWt	Net MDC	CP Issued OL Issued Comm. Op Exp. Date	1997-2002* Average Capacity Factors (Percent)
Quad Cities 2 Exelon	BWR-MARK 1 GE 3 S&L UE&C	2957	0855	02/15/1967 12/14/1972 03/10/1973 12/14/2032	39.0 50.6 97.9 92.1 93.1 87.5
River Bend 1 Entergy Nuclear South	BWR-MARK 3 GE 6 S&W S&W	3039	0966	03/25/1977 11/20/1985 06/16/1986 08/29/2025	83.2 95.1 69.6 89.4 95.3 100.1
Salem 1 Public Service Electric & Gas Co.	PWR-DRYAMB WEST 4LP PUBS UE&C	3459	1096	09/25/1968 12/01/1976 06/30/1977 08/13/2016	0.0 63.1 82.7 92.2 80.3 89.8
Salem 2 Public Service Electric & Gas Co.	PWR-DRYAMB WEST 4LP PUBS UE&C	3459	1092	09/25/1968 05/20/1981 10/13/1981 04/18/2020	25.5 80.9 82.0 86.3 99.5 87.5
San Onofre 2 Southern California Edison Co.	PWR-DRYAMB COMB CE BECH BECH	3438	1070	10/18/1973 09/07/1982 08/08/1983 10/18/2013	70.5 89.1 87.9 90.7 101.3 90.8
San Onofre 3 Southern California Edison Co.	PWR-DRYAMB COMB CE BECH BECH	3438	1080	10/18/1973 09/16/1983 04/01/1984 10/18/2013	72.1 95.8 88.9 101.6 60.0 100.9

U.S. Commercial Nuclear Power Reactors (Cont.)

Unit Operating Utility	Type NSSS AE Constructor	Licensed MWt	Net MDC	CP Issued OL Issued Comm. Op Exp. Date	1997-2002* Average Capacity Factors (Percent)
Seabrook 1 Florida Power and Light Company	PWR-DRYAMB WEST 4LP UE&C UE&C	3411	1155	07/07/1976 03/15/1990 08/19/1990 10/17/2026	78.3 81.1 85.8 78.1 85.9 91.8
Sequoyah 1 Tennessee Valley Authority	PWR-ICECND WEST 4LP TVA TVA	3411	1125	05/27/1970 09/17/1980 07/01/1981 09/17/2020	85.1 87.8 101.6 78.8 91.8 100.9
Sequoyah 2 Tennessee Valley Authority	PWR-ICECND WEST 4LP TVA TVA	3411	1126	05/27/1970 09/15/1981 06/01/1982 09/15/2021	89.2 97.3 91.8 92.3 101.6 86.6
Shearon Harris 1 Progress Energy	PWR-DRYAMB WEST 3LP EBSO DANI	2775	0900	01/27/1978 01/12/1987 05/02/1987 10/24/2026	78.3 93.4 96.2 91.0 71.3 99.4
South Texas Project 1 STP Nuclear Operating Co.	PWR-DRYAMB WEST 4LP BECH EBSO	3800	1251	12/22/1975 03/22/1988 08/25/1988 08/20/2027	90.1 98.4 88.0 78.2 94.4 99.2
South Texas Project 2 STP Nuclear Operating Co.	PWR-DRYAMB WEST 4LP BECH EBSO	3800	1251	12/22/1975 03/28/1989 06/19/1989 12/15/2028	91.0 90.1 89.4 96.1 87.1 75.0

U.S. Commercial Nuclear Power Reactors (Cont.)

Unit Operating Utility	Type NSSS AE Constructor	Licensed MWt	Net MDC	CP Issued OL Issued Comm. Op Exp. Date	1997-2002* Average Capacity Factors (Percent)
St. Lucie 1 Florida Power & Light Co.	PWR-DRYAMB COMB CE EBSO EBSO	2700	0839	07/01/1970 03/01/1976 12/21/1976 03/01/2036	77.8 94.9 88.9 102.0 91.3 94.1
St. Lucie 2 Florida Power & Light Co.	PWR-DRYAMB COMB CE EBSO EBSO	2700	0839	05/02/1977 06/10/1983 08/08/1983 04/06/2043	88.4 90.8 98.1 92.3 91.3 101.0
Summer South Carolina Electric & Gas Co.	PWR-DRYAMB WEST 3LP GIL DANI	2900	0966	03/21/1973 11/12/1982 01/01/1984 08/06/2042	87.5 101.8 88.2 74.9 79.9 87.2
Surry 1 Dominion	PWR-DRYSUB WEST 3LP S&W S&W	2546	0810	06/25/1968 05/25/1972 12/22/1972 05/25/2032	80.4 78.4 104.4 93.1 83.7 100.8
Surry 2 Dominion	PWR-DRYSUB WEST 3LP S&W S&W	2546	0815	06/25/1968 01/29/1973 05/01/1973 01/29/2033	91.9 100.1 83.7 92.9 94.1 91.4
Susquehanna 1 Pennsylvania Power & Light Co.	BWR-MARK 2 GE 4 BECH BECH	3489	1105	11/02/1973 11/12/1982 06/08/1983 07/17/2022	95.2 68.9 92.3 85.4 98.6 82.9

U.S. Commercial Nuclear Power Reactors (Cont.)

Unit Operating Utility	Type NSSS AE Constructor	Licensed MWt	Net MDC	CP Issued OL Issued Comm. Op Exp. Date	1997-2002* Average Capacity Factors (Percent)
Susquehanna 2 Pennsylvania Power & Light Co.	BWR-MARK 2 GE 4 BECH BECH	3489	1094	11/02/1973 06/27/1984 02/12/1985 03/23/2024	80.6 94.7 81.3 97.3 86.3 95.6
Three Mile Island 1 Exelon	PWR-DRYAMB B&W LLP GIL UE&C	2568	0802	05/18/1968 04/19/1974 09/02/1974 04/19/2014	86.0 97.7 77.4 103.5 78.7 104.1
Turkey Point 3 Florida Power & Light Co.	PWR-DRYAMB WEST 3LP BECH BECH	2300	0693	04/27/1967 07/19/1972 12/14/1972 07/19/2032	86.5 87.7 100.7 93.4 91.0 102.4
Turkey Point 4 Florida Power & Light Co.	PWR-DRYAMB WEST 3LP BECH BECH	2300	0693	04/27/1967 04/10/1973 09/07/1973 04/10/2033	89.7 101.7 94.5 91.9 100.6 96.4
Vermont Yankee Entergy Nuclear Northeast	BWR-MARK 1 GE 4 EBSO EBSO	1593	0510	12/11/1967 02/28/1973 11/30/1972 03/21/2012	95.5 71.9 90.9 101.5 93.4 88.7
Vogtle 1 Southern Nuclear Operating Co.	PWR-DRYAMB WEST 4LP SBEC GPC	3565	1215	06/28/1974 03/16/1987 06/01/1987 01/16/2027	81.2 99.6 93.5 91.2 100.9 85.9

U.S. Commercial Nuclear Power Reactors (Cont.)

Unit Operating Utility	Type NSSS AE Constructor	Licensed MWt	Net MDC	CP Issued OL Issued Comm. Op Exp. Date	1997-2002* Average Capacity Factors (Percent)
Vogtle 2 Southern Nuclear Operating Co.	PWR-DRYAMB WEST 4LP SBEC GPC	3565	1215	06/28/1974 03/31/1989 05/20/1989 02/09/2029	101.3 80.2 87.0 102.4 94.0 83.6
Waterford 3 Entergy Nuclear South	PWR-DRYAMB COMB CE EBSO EBSO	3390	1091	11/14/1974 03/16/1985 09/24/1985 12/18/2024	71.4 89.3 79.0 89.8 101.3 94.0
Watts Bar 1 Tennessee Valley Authority	PWR-ICECND WEST 4LP TVA TVA	3411	1125	01/23/1973 02/07/1996 05/27/1996 11/09/2035	77.7 94.7 84.4 92.4 97.7 92.1
Wolf Creek 1 Wolf Creek Nuclear Operating Corp.	PWR-DRYAMB WEST 4LP BECH DANI	3565	1165	05/31/1977 06/04/1985 09/03/1985 03/11/2025	82.7 101.5 89.3 88.3 101.0 88.6

**Note:* Average capacity factors are listed in year order starting with 1997.
Source: NRC and licensee data as compiled by NRC.

Abbreviations Used In Annex 1

ABB-CE	Asea Brown Boveri-Combustion Engineering	
ACE	ACEOWEN, Ateliers de Constructions Electriques de Charleroi S.A. (ACEC) and Cocerill Ougree-Providence (COP); with Westinghouse (Belgium)	
ACLF	ACECO/Creusot-loire/Framatome/Westinghouse-Europe	
AE	Architect-Engineer	
AEC	Atomic Energy Commission	
AECL	Atomic Energy of Canada, Ltd.	
AEE	Atomenergoexport	
AEP	American Electric Power	
AGN	Aerojet-General Nucleonics	
ASEA	Asea Brown Boveri-Asea Atom	
B&R	Burns & Roe	
B&W	Babcock & Wilcox	
BALD	Baldwin Associates	
BECH	Bechtel	
BRRT	Brown & Root	
BWR	Boiling-Water Reactor	
COMB	Combustion Engineering	
COMM. OP.	Date of Commercial Operation	
CON TYPE	Containment Type	
	DRYAMB	Dry, Ambient Pressure
	DRYSUB	Dry, Subatmospheric
	HTG	High-Temperature Gas-Cooled
	ICECND	Wet, Ice Condenser
	LMFB	Liquid Metal Fast Breeder
	MARK 1	Wet, Mark I
	MARK 2	Wet, Mark II
	MARK 3	Wet, Mark III
	OCM	Organic Cooled & Moderated
	PTHW	Pressure Tube, Heavy Water
	SCF	Sodium Cooled, Fast
	SCGM	Sodium Cooled, Graphite Moderated
CP	Construction Permit	

Abbreviations Used In Annex 1 (Cont.)

CP ISSUED	Date of Construction Permit Issuance	
CPPR	Construction Permit Power Reactor	
CWE	Commonwealth Edison Company	
CX	Critical Assembly	
DANI	Daniel International	
DBDB	Duke & Bechtel	
DER	Design Electric Rating	
DOE	Department of Energy	
DPR	Demonstration Power Reactor	
DUKE	Duke Power Company	
EBSO	Ebasco	
EXP. DATE	Expiration Date of Operating License	
FENOC	FirstEnergy Nuclear Operating Co.	
FRAM	Framatome	
FLUR	Fluor Pioneer	
G&H	Gibbs & Hill	
GCR	Gas-Cooled Reactor	
GE	General Electric	
GHDR	Gibbs & Hill & Durham & Richardson	
GIL	Gilbert Associates	
GPC	Georgia Power Company	
HIT	Hitachi	
HTG	High-Temperature Gas-Cooled	
HWR	Pressurized Heavy-Water Reactor	
IES	Iowa Electric	
JONES	J. A. Jones	
KAIS	Kaiser Engineers	
KWU	Kraftwerk Union, Siemens AG	
LIC. TYPE:	License Type	
	CP	Construction Permit
	OL-FP	Operating License-Full Power
	OL-LP	Operating License-Low Power
MAE	Ministry of Atomic Energy, Russian Federation	

Abbreviations Used In Annex 1 (Cont.)

MDC	Maximum Dependable Capacity - Net	
MHI	Mitsubishi Heavy Industries, Ltd.	
MWe	Megawatts Electrical	
MWt	Megawatts Thermal	
NIAG	Niagara Mohawk Power Corporation	
NPF	Nuclear Power Facility	
NSP	Northern States Power Company	
NSSS	Nuclear Steam System Supplier & Design Type	
	1	GE Type 1
	2	GE Type 2
	3	GE Type 3
	4	GE Type 4
	5	GE Type 5
	6	GE Type 6
	2LP	Westinghouse Two-Loop
	3LP	Westinghouse Three-Loop
	4LP	Westinghouse Four-Loop
	CE	Combustion Engineering
	CE80	CE Standard Design
	LLP	B&W Lowered Loop
	RLP	B&W Raised Loop
OL	Operating License	
OL ISSUED	Date of Latest Full Power Operating License	
PECO	Philadelphia Energy Company	
PG&E	Pacific Gas & Electric Company	
PHWR	Pressurized Heavy-Water-Moderated Reactor	
PSE	Pioneer Services & Engineering	
PUBS	Public Service Electric & Gas Company	
PWR	Pressurized-Water Reactor	
R	Research	
S&L	Sargent & Lundy	

Abbreviations Used In Annex 1 (Cont.)

S&W	Stone & Webster
SBEC	Southern Services & Bechtel
SSI	Southern Services Incorporated
STP	South Texas Project
TXU	Texas Utilities
TNPG	The Nuclear Power Group
TOSH	Toshiba
TR	Test Reactor
TVA	Tennessee Valley Authority
UE&C	United Engineers & Constructors
UTR	Universal Training Reactor
VT	Vermont
WDCO	Westinghouse Development Corporation
WEST	Westinghouse Electric