Regulatory Analysis for the Direct Final Rule: Appendix H to 10 CFR Part 50—Reactor Vessel Material Surveillance Program Requirements

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U.S. Nuclear Regulatory Commission
Office of Nuclear Material Safety and Safeguards
Division of Rulemaking

Abstract

The purpose of the direct final rule is to amend the requirements of Appendix H, "Reactor Vessel Material Surveillance Program Requirements" (Appendix H), to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR), "Domestic Licensing of Production and Utilization Facilities," and thus reduce the regulatory burden on reactor licensees and the U.S. Nuclear Regulatory Commission (NRC) for issues that are not significant to safety. Because the NRC considers this action to be noncontroversial, the agency is using the "direct final rule process" for this rulemaking. This document provides the regulatory analysis for the direct final rule revising the regulations in Appendix H to 10 CFR Part 50.

The current reactor vessel material surveillance regulations in Appendix H to 10 CFR Part 50 require licensees of commercial light-water nuclear power reactors with a peak neutron fluence at the end of the design life of the reactor vessel exceeding $1x10^{17}$ neutrons per centimeter-squared (with energy greater than 1 million electron volts) to maintain a reactor vessel material surveillance program. This program monitors the changes in mechanical properties of the reactor vessel materials. The material surveillance programs include a number of capsules that contain test specimens (e.g., Charpy and tensile) and monitoring materials (temperature and dosimetry) that are located inside the reactor vessel and are placed closer to the core than to the vessel's inside wall. Based on the location of these capsules, the amount of neutron fluence they receive typically exceeds that received by the reactor vessel wall itself. Therefore, the specimens within the surveillance capsule experience operating conditions identical to the vessel wall but at higher levels of neutron irradiation. This practice allows for the collection of bounding test data on the change in material properties of the reactor vessel following irradiation, which informs the NRC's regulatory decisions and operational assessments of the reactor vessel material at operating plants.

This analysis describes the regulatory framework for reactor vessel material surveillance programs, summarizes the background of these surveillance programs, outlines the regulatory topics that have motivated this direct final rule effort, and presents alternatives to address these topics.

The analysis shows that this direct final rule can be implemented with no decrease in public health and safety and recommends using the direct final rule process to minimize the use of agency resources, which would allow the benefits from the revised requirements to become effective earlier.

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Abbreviations and Acronyms

AEC	Atomic Energy Commission
ADAMS	Agencywide Documents Access and Management System
ASME	American Society of Mechanical Engineers
ASME Code	ASME Boiler and Pressure Vessel Code
ASTM	ASTM International (formerly American Society for Testing and Materials)
BWRVIP	Boiling Water Reactor Vessel and Internals Project
C	Celsius
CFR	*Code of Federal Regulations*
cm	centimeter(s)
CMM	correlation monitor material
CPI-U	consumer price index for all urban consumers
E	energy
F	Fahrenheit
FR	*Federal Register*
ft-lb	foot-pound
GALL	generic aging lessons learned
GDC	general design criterion
HAZ	heat-affected zone
IAEA	International Atomic Energy Agency
MeV	million electron-volts
n	neutron
NPV	net present value
NRC	U.S. Nuclear Regulatory Commission
OMB	U.S. Office of Management and Budget
PERT	program evaluation and review technique
PTS	pressurized thermal shock
RIN	regulation identifier number
RPV	reactor pressure vessel
RT_{NDT}	reference temperature for nil-ductility transition
RT_{PTS}	reference temperature for pressurized thermal shock
SECY	Office of the Secretary
SRM	staff requirements memorandum
SLR	subsequent license renewal
T_0	reference temperature

U.S.	United States of America
USE	upper-shelf energy
ΔYS	change in yield strength

Executive Summary

The U.S. Nuclear Regulatory Commission (NRC), partly based the requirements in Appendix H, "Reactor Vessel Material Surveillance Program Requirements" (Appendix H), to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR), "Domestic Licensing of Production and Utilization Facilities," on the information contained in the American Society for Testing and Materials International (ASTM) E 185-73, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels"; ASTM E 185-79, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels"; and ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," all of which Appendix H to 10 CFR Part 50 incorporates by reference.

Appendix H to 10 CFR Part 50 requires light-water nuclear power reactor licensees to have a reactor vessel material surveillance program to monitor changes in the fracture toughness properties of the reactor vessel materials adjacent to the reactor core. Unless it can be shown that the end of design life neutron fluence is below certain criteria, the NRC requires licensees to implement a materials surveillance program that tests irradiated material specimens that are located in test capsules in the reactor vessels. The program evaluates changes in material fracture toughness and thereby assesses the integrity of the reactor vessel. For each capsule withdrawal, the test procedures and reporting requirements must meet the requirements of ASTM E 185-82 to the extent practicable for the configuration of the specimens in the capsule. The design of the material surveillance program and the withdrawal schedule must meet the requirements of the edition of ASTM E 185 that is current on the issue date of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) to which the licensee purchased the reactor vessel. Licensees may use later editions of ASTM E 185, up to and including those editions through 1982. In sum, the reactor vessel material surveillance program must comply with ASTM E 185, as modified by Appendix H to 10 CFR Part 50. The licensee establishes the number, design, and location of these surveillance capsules within the reactor vessel during the design of the program, before initial plant operation.

Appendix H to 10 CFR Part 50 also specifies that the licensee must include each capsule withdrawal and the test results in a summary technical report it submits to the NRC within one year of the date of capsule withdrawal, unless the Director of the NRC Office of Nuclear Reactor Regulation grants an extension. The NRC uses the results from the reactor vessel material surveillance program to assess licensee submittals related to pressure-temperature limits in accordance with Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50 and to assess the compliance of pressurized-water reactor licensees with § 50.61, "Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events," or § 50.61a, "Alternate Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events."

This regulatory analysis discusses two alternatives. Alternative 1, the no-action alternative, would maintain the current requirements in Appendix H to 10 CFR Part 50 (i.e., status quo) and, thus, the specimens and testing required by ASTM E 185-73, E 185-79, and E 185-82, as applicable. Alternative 1 avoids the costs that the rule would impose; however, the NRC will continue to require licensees to do the following:

- Test Charpy impact specimens for the weld heat-affected zone (HAZ).
- Test tension specimens for the weld metal and base metal at various temperatures.
- Test correlation monitor materials (CMMs), if they were included, and examine thermal monitors in each surveillance capsule in accordance with ASTM E 185-82, to the extent practicable.

Furthermore, licensees that need additional time to submit their surveillance capsule reports would continue to submit extension requests for NRC review and approval.

As a result, Alternative 1 (status quo) is not deemed viable and was not considered further in this regulatory analysis.

Under Alternative 2, the NRC would prepare a direct final rule to revise the regulations to alleviate the burden to existing licensees and to future applicants with no adverse impact on public health and safety. The rulemaking alternative achieves the objective of burden reduction for the reactor vessel material surveillance program while maintaining a comparable level of safety. This alternative also has the advantage of being relatively simple to implement.

The NRC considered another alternative during the regulatory basis phase and eliminated it at that time. That rulemaking alternative considered incorporating by reference the 2016 editions of ASTM E 185 and ASTM E 2215, "Standard Practice for Evaluation of Surveillance Capsules from Light-Water Moderated Nuclear Power Reactor Vessels," into Appendix H to 10 CFR Part 50. However, the NRC's assessment determined that the burden associated with implementing these ASTM standards would be significant without a corresponding benefit to public health and safety. As a result, the NRC concluded that this alternative was not viable and did not consider it further.

The NRC makes the key findings described below.

Rulemaking Analysis. The direct final rule is projected to result in a cost-justified change based on net averted costs to the industry and the NRC. Table ES-1 shows these net benefits over a 21-year period, which result in net benefits to the industry and the NRC ranging from $701,000 using a 7-percent discount rate to $940,000 using a 3-percent discount rate.

Table ES-1 Direct Final Rule Total Costs and Benefits

Description	Alternative—Direct Final Rule[b]		
	Undiscounted	7% NPV[a]	3% NPV
Industry Operation	$1,051,000	$609,000	$816,000
NRC Operation	$159,000	$92,000	$1,243,000
Total	$1,210,000	$701,000	$940,000
Average Annual Cost	$57,600		
Annualized Cost with 7% Discounting		$64,700	
Annualized Cost with 3% Discounting			$61,000

[a] NPV = net present value
[b] Totals may not match due to rounding

According to Executive Order, “Regulatory Planning and Overview” (58 FR 51735; October 4, 1993), an economically significant regulatory action is one that would have an annual effect on the economy of $100 million or more. This final rule does not reach this threshold because the annualized cost of the direct final rule would be $58,000 ($61,000 using a 3-percent discount rate or $65,000 using a 7-percent discount rate).

Nonquantified costs and benefits. The rule’s nonquantified benefits include those related to the principles of good regulation: independence, openness, efficiency, clarity, and reliability. Each nonquantified benefit is described below.

Independence. Final decisions would be based on objective, unbiased assessments of all information and would be documented with reasons explicitly stated.

Openness. The NRC has engaged the regulated community, the public, and other interested stakeholders through public meetings during the early development of the regulatory basis and rulemaking to ensure that the NRC considered diverse views in the regulatory decision making process.

Efficiency. The direct final rule process is the most effective and efficient approach to conducting this rulemaking effort because (1) it would minimize the use of agency resources and (2) the revised requirements would become effective earlier, giving licensees the greatest benefit. The direct final rule would continue to ensure the protection of public health and safety.

Clarity. The rulemaking effort to revise Appendix H to 10 CFR Part 50 would result in coherent, logical, and practical regulations. The revised requirements would be readily understood and easily applied.

Reliability. The rulemaking effort would result in regulations that are based on the best available knowledge from research and operational experience. The NRC would use this information to revise the requirements in Appendix H to 10 CFR Part 50 to lend stability to the design and implementation of a reactor vessel material surveillance program.

Sensitivity Analysis. The NRC performed a sensitivity analysis on the effect of delaying the capsule withdrawals and testing. The analysis evaluates the impact if the removal and testing of the remaining capsules were delayed seven years to model the possible impact of second license renewals. The NRC determined that the effect of this delay reduces the potential averted costs of this rule by ($265,000).

Uncertainty Analysis. This regulatory analysis uses estimates of values that are sensitive to plant-specific cost drivers and plant dissimilarities. To address these uncertainties in the analysis, the NRC used a Monte Carlo simulation to quantify this uncertainty and to determine those variables having the greatest effect on the value of the output variable. Figure ES-1 displays the probability distribution function and the descriptive statistics of the incremental benefits and costs of the direct final rule alternative compared to the no-action alternative.

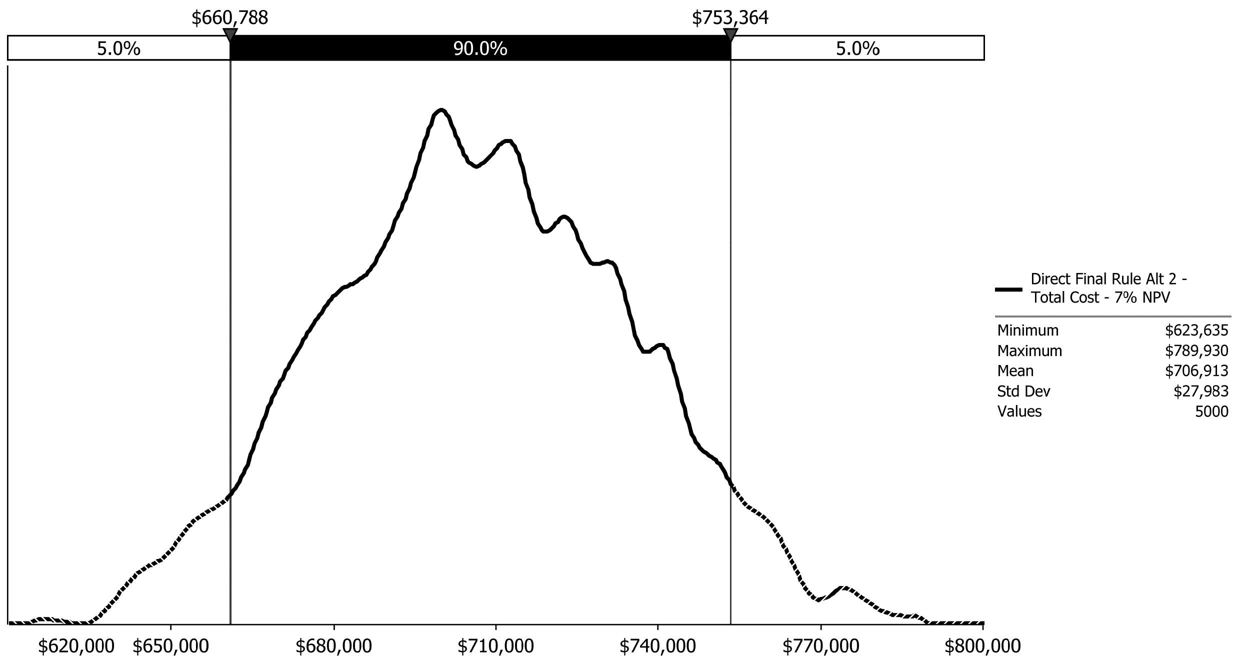

Figure ES-1 Net Cost and Benefits of the Final Rule at 7-Percent Discounting (2020$)

Decision Rationale. Relative to the no-action baseline, the NRC concludes that the direct final rule is justified from a quantitative standpoint because its provisions would result in net averted costs (i.e., net benefits) to the industry and is effectively cost-neutral to the NRC. In addition, the NRC concludes that the direct final rule is also justified when considering nonquantified costs and benefits because the significance of the nonquantified benefits outweighs that of the nonquantified costs.

1.0 Introduction

1.1 Scope of Document

This document presents the regulatory analysis for the direct final rule, which encompasses commercial light-water nuclear power reactors required to have a reactor vessel material surveillance program under Appendix H, "Reactor Vessel Material Surveillance Program Requirements" (Appendix H), to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR), "Domestic Licensing of Production and Utilization Facilities."

1.2 Background

In 2001, the U.S. Nuclear Regulatory Commission (NRC) began a rulemaking to revise Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50 (RIN 3150-AG98; NRC Docket ID: NRC-2008-0582) to eliminate the pressure-temperature limits related to the metal temperature of the reactor vessel closure head flange and vessel flange areas. The NRC expanded the rulemaking scope in 2008 to include revisions to Appendix H to 10 CFR Part 50, because the fracture toughness analysis required by Appendix G to 10 CFR Part 50 relies on data obtained from the reactor vessel material surveillance program established under Appendix H to 10 CFR Part 50.

In COMSECY-14-0027, "Rulemaking to Revise Title 10, *Code of Federal Regulations*, Part 50, Appendix H, 'Reactor Vessel Material Surveillance Program Requirements,'" dated June 25, 2014 (NRC, 2014b, not publicly available), the NRC staff requested Commission approval to separate the rulemaking activities to revise Appendices G and H to 10 CFR Part 50 and to proceed separately with rulemaking for Appendix H to 10 CFR Part 50.

In its staff requirements memorandum (SRM) to COMSECY-14-0027, dated August 8, 2014 (NRC, 20142, not publicly available), the Commission approved the staff's recommendation to proceed with a separate rulemaking for Appendix H to 10 CFR Part 50. The SRM to COMSECY-14-0027 directed staff to begin the rulemaking for Appendix H to 10 CFR Part 50 independent of the completion date or conclusions of the technical-basis development activities for Appendix G to 10 CFR Part 50. Subsequently, the Commission directed the staff in SRM-SECY-16-0009, "Recommendations Resulting from the Integrated Prioritization and Re-Baselining of Agency Activities," dated April 13, 2016 (NRC, 2016), to stop all work on the development of the technical basis for a potential change to Appendix G to 10 CFR Part 50.

1.3 Problem Statement

Since the issuance of Appendix H to 10 CFR Part 50 in 1973, reactor vessel material surveillance programs have produced substantial material data analyses, knowledge, and experience. Thus, the NRC is undertaking this direct final rule to lessen the regulatory burden on both reactor licensees and the NRC by reducing testing and reporting requirements without affecting public health and safety. Section 3.0 discusses the regulatory topics that this direct final rule considers.

This regulatory action uses the direct final rule process to revise the testing and reporting requirements in Appendix H to 10 CFR Part 50. The NRC has determined that this is the appropriate way to develop a rule that reduces regulatory burden and is cost-beneficial for issues that are not significant to safety. A direct final rule minimizes the use of agency resources and permits the revised requirements to become effective earlier, giving licensees the greatest benefit. This action involves no technical, policy, or legal issues, and the NRC is not revising any existing regulatory guidance or developing new guidance as part of this activity. For these reasons, NRC views this action to be noncontroversial and does not expect to receive significant adverse public comment that would result in withdrawal of the direct final rule.

2.0 Existing Regulatory Framework

2.1 Appendix H to 10 CFR Part 50

2.1.1 Current Requirements under Appendix H to 10 CFR Part 50

Light-water reactor vessels are fabricated from low-alloy steel, which can become less ductile and thereby more susceptible to unstable fracture because of the cumulative effects of neutron irradiation. Appendix H to 10 CFR Part 50 requires a material surveillance program for reactor vessels for which the peak neutron fluence at the end of the design life of the vessel will exceed $1x10^{17}$ neutrons per centimeter-squared (n/cm^2) (with energy greater than 1 million electron volts (E > 1.0 MeV)). The purpose of the material surveillance program required by Appendix H to 10 CFR Part 50 is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light-water nuclear power reactors that result from exposure of these materials to neutron irradiation and the thermal environment. This material surveillance program generates fracture toughness test data from irradiated material specimens exposed in surveillance capsules, which the licensee periodically withdraws from the reactor vessel.

The activities addressed as part of designing a reactor vessel material surveillance program include selecting materials to be monitored by the surveillance program, choosing appropriate test specimen types and numbers of specimens, establishing the number of capsules and their placement in the reactor vessel, and developing the surveillance capsule withdrawal schedule. The activities included in a reactor vessel material surveillance program include maintaining a surveillance capsule withdrawal schedule, periodically withdrawing capsules, performing tests on the specimens contained in the capsules, and reporting the test results.

The design of this material surveillance program and the withdrawal schedule must meet the requirements of the edition of the American Society for Testing and Materials International (ASTM) E 185 that is current on the issue date of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) to which the licensee purchased the reactor vessel. Licensees may use later editions of ASTM E 185, up to and including those editions through 1982 (ASTM E 185-82, “Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels”). Appendix H to 10 CFR Part 50 specifically incorporates by reference ASTM E 185-73, “Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels”; ASTM E 185-79, “Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels”; and ASTM E 185-82. In sum, the material surveillance program must comply with ASTM E 185, as modified by Appendix H to 10 CFR Part 50. The licensee must submit the withdrawal schedule, including any subsequent changes, to the NRC for approval prior to its implementation.

Appendix H to 10 CFR Part 50 requires that surveillance specimen capsules be located near the inside reactor vessel wall in the beltline region so that the specimen irradiation history duplicates, to the extent practicable, the neutron spectrum, temperature history, and maximum neutron fluence experienced by the reactor vessel inner surface. Furthermore, the design and location of the surveillance capsule holders must permit the insertion of replacement capsules.

For each capsule withdrawal, the test procedures and reporting requirements must meet the requirements of ASTM E 185-82, to the extent practicable for the configuration of the specimens in the capsule. This is to ensure that the changes in mechanical properties of the ferritic reactor vessel materials can be evaluated and to provide experimental data to benchmark against dosimetry calculations.

As an alternative to a plant-specific material surveillance program, Appendix H to 10 CFR Part 50 permits the development of an integrated surveillance program, which requires NRC approval, on a case-by-case basis. An integrated surveillance program involves representative materials chosen for surveillance for a certain reactor being irradiated in one or more other reactors that have similar design and operating features. Appendix H to 10 CFR Part 50 requires that an integrated surveillance program incorporate the following criteria:

- The reactor in which the materials will be irradiated and the reactor for which the materials are being irradiated must have sufficiently similar design and operating features to permit accurate comparisons of the predicted amount of radiation damage.

- Each reactor must have an adequate dosimetry program.

- There must be adequate arrangements for data sharing among plants.

- There must be a contingency plan to ensure that the material surveillance program for each reactor will not be jeopardized by operating at a reduced power level or by an extended outage of another reactor from which data are expected.

- There must be substantial advantages to be gained, such as fewer power outages or reduced personnel exposure to radiation, as a direct result of not requiring surveillance capsules in all reactors in the set.

For an integrated surveillance program, Appendix H to 10 CFR Part 50 does not permit a reduction in the requirements for the number of materials to be irradiated, the specimen types, or the number of specimens per reactor, nor is a reduction in the amount of testing permitted unless previously authorized by the Director of the NRC Office of Nuclear Reactor Regulation.

Following each withdrawal and testing of a surveillance capsule, the test results must be the subject of a technical report, which must include the data required by ASTM E 185 and the results of all fracture toughness tests conducted on the beltline materials in the irradiated and unirradiated conditions. The licensee must submit the report to the NRC within one year of the date of capsule withdrawal, unless the Director of the NRC Office of Nuclear Reactor Regulation grants an extension.

2.1.2 Current Regulatory Guidance for Appendix H to 10 CFR Part 50

Initial Period of Operation

Appendix H to 10 CFR Part 50 requires that reactor vessels have their beltline materials monitored by a material surveillance program complying with ASTM E 185. Specifically, the design of the material surveillance program and the withdrawal schedule must meet the requirements of the edition of the ASTM E 185 that is current on the issue date of the ASME Code to which the licensee purchased the reactor vessel. Licensees may use later editions of ASTM E 185, up to and including those editions through 1982. In sum, the material surveillance program must comply with ASTM E 185, as modified by Appendix H to 10 CFR Part 50. Furthermore, the test procedures and reporting requirements must meet the requirements of ASTM E 185-82 to the extent practicable for the configuration of the specimens in the capsule for each capsule withdrawal.

The ASTM E 185 contains the necessary procedures and guidelines for the design of a material surveillance program. Specifically, this includes the selection of reactor vessel materials to be monitored and the contents within the surveillance capsule, the means to encapsulate these contents, and the location of the surveillance capsules within the reactor vessel. The ASTM E 185 also contains the necessary procedures and guidelines for measuring and testing the contents of the surveillance capsule and for reporting the results to the NRC; specifically, measuring the mechanical properties and radiation exposure conditions and determining the irradiation effects. Under Appendix H to 10 CFR Part 50 and ASTM E 185, the material surveillance program and the withdrawal schedule are originally established and designed for the initial 40-year operating license of a nuclear power plant (see Section 7.6.2 of ASTM E 18579 and ASTM E 18582).

Renewal of Operating License—License Renewal and Subsequent License Renewal

To renew its operating license or combined license for plant operation beyond 40 years, a licensee must comply with the regulations in 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," and demonstrate that the licensee will adequately manage the effects of aging to maintain the intended function of systems, structures and components within the scope of 10 CFR Part 54 consistent with the current licensing basis. Therefore, licensees have continued to use their material surveillance program under Appendix H to 10 CFR Part 50, as supplemented by additional guidance, to demonstrate that they will adequately manage embrittlement on the reactor vessel during extended operation.

The reactor vessel material surveillance programs are ongoing programs that extend beyond the original license of a nuclear power plant (i.e., during license renewal to operate for 60 years and potentially during subsequent license renewal to operate for 80 years). The objective of the material surveillance program during extended plant operations remains the same as it was during the initial 40-year operating license, to continue monitoring changes in fracture toughness of the reactor vessel materials to ensure the integrity of the reactor vessel. As such, there are no aspects of the material surveillance program that are uniquely affected by license renewal and subsequent license renewal.

Because the withdrawal schedule of surveillance capsules was initially based on plant operation during the original 40-year license term, it may be necessary for the reactor vessel material surveillance program to incorporate standby capsules or capsules containing reconstituted specimens (i.e., specimens from previously tested capsules) to provide monitoring during plant operation beyond the original 40-year license term. As an additional alternative, applicants may join an integrated surveillance program. The NUREG-1801, "Generic Aging Lessons Learned (GALL) Report—Final Report," Revision 2, issued December 2010 (NRC, 2010), contains guidance for licensees seeking plant operation for 60 years, and the NRC provides guidance for licensees seeking plant operation for 80 years in NUREG-2191, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report," issued July 2017 (NRC, 2017).

2.1.3 History of Appendix H to 10 CFR Part 50

As published in the *Federal Register* on July 3, 1971 (36 FR 12697), "Fracture Toughness Requirements for Nuclear Power Reactors," the Atomic Energy Commission (AEC) issued for public comment a proposed rulemaking to add to 10 CFR Part 50 a new Appendix G, "Fracture Toughness Requirements," and new Appendix H, "Reactor Vessel Material Surveillance Program Requirements." The AEC stated that the purpose of the proposed amendments was to specify minimum fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary for boiling- and pressurized-water power reactors and to require surveillance of the fracture toughness specimens of the reactor vessel material by periodic tests.

The AEC indicated that the proposed amendments to add Appendices G and H to 10 CFR Part 50 would specify minimum fracture toughness requirements needed to ensure that a plant meets General Design Criterion (GDC) 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 and describe methods to determine the fracture toughness of reactor coolant pressure boundary materials. Because of the special importance to safety of the reactor vessel and because the fracture toughness properties of the reactor vessel beltline region may change because of neutron irradiation, the proposed amendments would specify special requirements for periodic testing of irradiated specimens of reactor vessel beltline materials.

GDC 31 states the following:

> The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady-state and transient stresses, and (4) size of flaws.

On July 17, 1973 (38 FR 19012), the AEC issued the final rule, "Fracture Toughness and Surveillance Program Requirements," amending 10 CFR Part 50, to include Appendices G and H. The AEC explained that Appendix H to 10 CFR Part 50 differs from the amendments published for public comment in the following ways:

(1) Terminology was changed to be consistent with that of Appendix G to 10 CFR Part 50 and the ASME Code. In particular, the adjustment for irradiation effects is described in these amendments as an adjustment of the reference temperature (T_0) for nil ductility transition, RT_{NDT}, and the

amount of temperature shift is determined by a slightly different treatment of the Charpy data than that given in the proposed amendment.

(2) Provision was made for accelerated irradiation capsules and for modification of capsule withdrawal schedules based on the results of tests of specimens that received the accelerated irradiation.

(3) A general provision for an integrated surveillance program was substituted for the specific requirements given in the proposed rule. It appeared from comments that it would be impractical to meet the requirements of the proposed rule for a commonality of multiple reactors.

The AEC reiterated that Appendices G and H to 10 CFR Part 50 are intended to implement GDC 31. The AEC further explained that the margin of safety against brittle fracture would be controlled more quantitatively by these amendments than by the proposed rule, particularly with regard to specific guidelines for the treatment of heat-up and cooldown conditions. Appendices G and H to 10 CFR Part 50 use language consistent with the ASME Code and have adopted certain of its requirements but also include several key supplemental requirements. For the vessel beltline, in-service requirements were based on the reference temperature (T_0), as adjusted, to account for irradiation damage, and there was an additional fracture toughness requirement in the form of upper-shelf energy (USE) values from the Charpy curve for the material in its unirradiated condition.

Appendix H to 10 CFR Part 50 has undergone several revisions following the issuance of the 1973 final rule. The significance of these amendments has varied from strictly administrative changes to the revision of material surveillance program requirements. The sections below include further details about the substantive changes.

On September 26, 1979 (44 FR 55328), the NRC amended Appendix H to 10 CFR Part 50 to permit greater flexibility in meeting the material surveillance program requirements and to simplify requirements by substituting references to national standards that the NRC's regulations had already incorporated by reference. The NRC revised Appendix H, paragraph II.C.2, to no longer prohibit attachment of surveillance capsules to the reactor vessel wall, because, for some vessel designs, the advantages of attachment to the wall (fewer problems in achieving the desired lead factor and the structural integrity of the capsule holder) outweighed the disadvantage of concern for reactor vessel integrity. Furthermore, the NRC added requirements to state that, if capsule holders are attached to the vessel wall, the attachments must meet ASME Code requirements for construction and inspection of permanent structural attachments to reactor vessels. Additionally, the NRC revised Appendix H to 10 CFR Part 50 to remove the fixed limits on lead factor (i.e., the ratio of neutron flux at the capsule to the maximum flux at the reactor vessel inner wall) of greater than one but less than three. The NRC explained that enforcement of the then-present requirement would require modification of certain designs that had satisfactorily met all surveillance and structural requirements in service. Furthermore, retention of the general requirement on the lead factor satisfied safety concerns.

On May 27, 1983 (48 FR 24008), the NRC amended 10 CFR Part 50 to clarify the applicability of the requirements to all plants, modify certain requirements, and shorten and simplify these regulations by more extensively incorporating by reference appropriate national standards. Specifically, the NRC revised Appendix H to 10 CFR Part 50 to incorporate ASTM E 185-73, E 185-79, and E 185-82 by reference. The NRC also revised the proposed requirement that licensees submit surveillance reports within 90 days after completion of testing to require submittal of these reports within 1 year of capsule withdrawal, unless the Director of the NRC Office of Nuclear Reactor Regulation grants an extension. This revision still accomplished the primary purposes of this requirement for timely reporting of test results and notification of any problems.

On December 19, 1995 (60 FR 65456), the NRC amended Appendix H to 10 CFR Part 50 to remove the provision for integrated surveillance programs that permitted the reduction in the amount of testing if the initial results agreed with the predictions. The NRC described the other principal change as a clarification of the editions of ASTM E 185 that apply to the various portions of the reactor vessel material surveillance programs. The NRC explained that a material surveillance program consists of two essential parts: (1) the design of the program and (2) the subsequent testing and reporting of results from the surveillance capsules. Once the NRC approves the design of a material surveillance program, it cannot be changed without prior approval. However, the testing and reporting requirements are updated, along with technical improvements made to ASTM E 185. The NRC revised Appendix H to 10 CFR Part 50 so that, for each capsule withdrawal, the test procedures and reporting requirements must meet the requirements of ASTM E 185-82 to the extent practicable for the configuration of the specimens in the capsule.

2.2 ASTM Standards for Reactor Vessel Material Surveillance Programs

Appendix H to 10 CFR Part 50 incorporates by reference ASTM E 185-73, ASTM E 185-79, and ASTM E 185-82. These standards provide procedures for monitoring the radiation-induced changes in the mechanical properties of ferritic materials in the beltline of light-water cooled nuclear power reactor vessels and include guidelines for designing a minimum material surveillance program, selecting materials, and evaluating test results. The purpose of this material surveillance program is to monitor changes in the properties of actual vessel materials caused by long-term exposure to the neutron radiation and temperature environment of the given reactor vessel.

The aspects of ASTM E 185 on designing a reactor vessel material surveillance program fall into the following four categories:

(1) test material
(2) test specimens
(3) irradiation conditions
(4) capsules and withdrawal schedule

Since its incorporation into Appendix H to 10 CFR Part 50, ASTM E 185 was revised in 2002 to divide the contents of the standard so that ASTM E 185 included the details on reactor vessel material surveillance program design, while ASTM E 2215, “Standard Practice for Evaluation of Surveillance Capsules from Light-Water Moderated Nuclear Power Reactor Vessels,” contained details on surveillance capsule testing and evaluation. The aspects for surveillance capsule testing and evaluation specified in ASTM E 185 and, ultimately, in ASTM E 2215, fall into the following five categories:

(1) characterization of the reactor environment
(2) materials to test and specimen testing
(3) test data evaluation
(4) adjustment of the capsule withdrawal schedule

(5) retention of tested specimens

2.2.1 Changes to the ASTM Standards

The operation of commercial light-water nuclear power plants since the 1970s provided empirical evidence of the effects of irradiation embrittlement on reactor vessel steels. This, combined with a better scientific understanding of irradiation embrittlement, prompted revisions and updates to the ASTM requirements for surveillance monitoring programs. When the NRC prepared the regulatory basis, the 2016 edition of ASTM E 185 and ASTM E 2215 were the most up-to-date versions of these standards.

During the development of this direct final rule, the NRC assessed the option of incorporating by reference the 2016 editions of ASTM E 185 and ASTM E 2215 into Appendix H to 10 CFR Part 50 but concluded that the burden associated with implementing these ASTM standards would be significant without a corresponding benefit to public health and safety. As a result, the NRC concluded that this alternative was not viable and did not consider it further.

2.2.2 Differences in ASTM Standards Related to Aspects Required by Appendix H to 10 CFR Part 50

The NRC reviewed the 1973, 1979, and 1982 editions of ASTM E 185 to determine whether there were any differences in the standards that would affect the regulatory topics addressed during the development of this direct final rule. These aspects are related to the inclusion and testing of heat-affected zone (HAZ) specimens, tension specimens, correlation monitor material (CMM), and thermal monitors in surveillance capsules.

<u>Test Materials and Test Specimens</u>

The 1973, 1979, and 1982 editions of ASTM E 185 consistently specify that the surveillance test materials be prepared from samples taken from the actual materials used in fabricating the beltline of the reactor vessel and that these surveillance test materials include the base metal, butt weld, and weld HAZ. Furthermore, these three editions of ASTM E 185 consistently require 12 Charpy impact specimens for base metal, weld metal, and weld HAZ, per capsule, in the irradiated condition; and 15 Charpy impact specimens for base metal, weld metal, and weld HAZ in the unirradiated condition.

The 1973 edition of ASTM E 185 only required tension specimens if the predicted increase in transition temperature of the reactor vessel steel is greater than 37.8 degrees Celsius (C) (100 degrees Fahrenheit (F)) or where the calculated peak neutron fluence (E > 1 MeV) of the reactor vessel is greater than $5x10^{18}$ n/cm^2. Specifically, ASTM E 185-73 required two tension specimens for base metal and weld metal, per capsule, in the irradiated condition and three tension specimens for base metal and weld metal in the unirradiated condition. On the other hand, ASTM E 185-79 and ASTM E 185-82 required three tension specimens for base metal and weld metal, per capsule, in the irradiated condition and three tension specimens for base metal and weld metal in the unirradiated condition, regardless of the predicted increase in transition temperature of the reactor vessel steel.

Because Appendix H to 10 CFR Part 50 incorporates by reference the 1973, 1979, and 1982 editions of ASTM E 185, it is likely that there is a variation between the contents of surveillance capsules (i.e., presence of tension specimens and number of tension specimens) in the current operating fleet. This is because the test material requirements in the current operating fleet were established during the design of the plant's material surveillance programs, which may have occurred before the issuance of the 1973 final rule that incorporated the 1973 edition of ASTM E 185 and its subsequent amendment in 1995 that incorporated the 1979 and 1982 versions of ASTM E 185.

On December 19, 1995 (60 FR 65456), the NRC revised Appendix H to 10 CFR Part 50 to specify that, for each capsule withdrawal, the test procedures and reporting requirements must meet the requirements of ASTM E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Thus, any variations in requirements and recommendations for testing specimens in the 1973, 1979, and 1982 editions of ASTM E 185 are not significant.

Correlation and Thermal Monitors

The 1973 edition of ASTM E 185 specified that the testing of specimens should be modified as outlined in ASTM E 184-79, "Standard Practice for Effects of High-Energy Neutron Radiation on the Mechanical Properties of Metallic Materials," which recommends that a metal specimen from a standard reference material be used to correlate one irradiation experiment with another. This is done so that the mechanical property changes of the reference material may serve as a relative standard for estimating exposure. The 1979 and 1982 editions of ASTM E 185 explicitly categorize correlation monitors as optional for inclusion in surveillance capsules and discuss them in the ASTM standard instead of in a secondary reference. Consistently, these three editions of ASTM E 185 only recommend the inclusion of CMMs in surveillance capsules.

The 1973, 1979, and 1982 editions of ASTM E 185 consistently specify the insertion of thermal monitors within surveillance capsules. These three editions of ASTM E 185 proposed the use of low melting point elements or eutectic alloys, instead of instrument monitors, to detect significant variations in exposure temperature to provide evidence of the maximum exposure temperature of the specimens. These monitor materials should be selected to indicate unforeseen capsule temperatures.

2.3 Material Surveillance Data Required by Appendix H to 10 CFR Part 50

The NRC uses the material surveillance data required under Appendix H to 10 CFR Part 50 for the purposes listed below.

2.3.1 Section 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation"

In § 50.60, the NRC requires licensees of light-water nuclear power reactors to meet the fracture toughness requirements of Appendix G to 10 CFR Part 50 and the material surveillance program requirements in Appendix H to 10 CFR Part 50. The regulations permit these licensees to use alternatives to the requirements as described in Appendices G and H 10 CFR Part 50, when the NRC grants an exemption under § 50.12, or § 52.7, both titled "Specific Exemptions."

2.3.2 Section 50.61, "Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events," and Section 50.61a, "Alternate Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events"

The operational characteristics of pressurized-water reactors make them susceptible to a severe transient identified as pressurized thermal shock. The initiating pressurized thermal shock event is a small-break loss-of-coolant accident, followed by rapid cooling (i.e., thermal shock) of the internal vessel surface from safety injection, which is then coupled with repressurization of the reactor coolant system. With a sufficiently embrittled reactor vessel, the combination of cold vessel surface, high thermal stresses, and high pressure can cause the brittle propagation of small cracks in the reactor vessel, potentially resulting in propagation of a through-wall crack and possible failure of the vessel. As a condition of their license, pressurized-water reactors must demonstrate compliance with § 50.61 or § 50.61a to ensure that they do not approach the levels of embrittlement that make them susceptible to failure as a result of pressurized thermal shock.

In § 50.61, the NRC requires the estimation of the reference temperature (T_0) for pressurized thermal shock (i.e., RT_{PTS}) of the steels in the reactor vessel beltline using the end of license neutron fluence levels and a demonstration that the reactor vessel RT_{PTS} values are below the screening criteria specified in the rule. This estimation uses material surveillance program results[1] in conjunction with formulae and tables in § 50.61.

The screening criteria in § 50.61 restrict the maximum values of RT_{PTS} permitted during the plant's operational life to 132 degrees C (270 degrees F) for axial welds, plates, and forgings, and 149 degrees C (300 degrees F) for circumferential welds. Should RT_{PTS} exceed these screening criteria, § 50.61 requires the licensee to either take actions to keep RT_{PTS} below the screening criteria or perform plant-specific analyses to demonstrate operating the plant beyond the § 50.61 screening limits.

In § 50.61a, the NRC provides an alternate approach to demonstrating adequate toughness, including less restrictive screening criteria than those included in § 50.61. The approach in § 50.61a includes (1) an alternate embrittlement trend correlation for use in predicting irradiation-induced shifts in the RT_{NDT}, (2) new requirements for evaluating plant and heat-specific surveillance data to ensure the applicability of the alternate embrittlement trend correlation, and (3) new requirements for evaluating reactor vessel inservice inspection data. In § 50.61a, the NRC also defines generic procedures and criteria to ensure compliance with the revised pressurized thermal shock evaluation requirements. If licensees cannot meet these generic criteria, this alternate pressurized thermal shock rule allows them to perform additional plant-specific evaluations to demonstrate that the reactor vessel has adequate resistance to fracture during pressurized thermal shock events and to submit them to the NRC for approval.

2.3.3 Section 50.66, "Requirements for Thermal Annealing of the Reactor Pressure Vessel"

This regulation is intended for use by those light-water nuclear power reactors where neutron radiation has reduced the fracture toughness of the reactor vessel materials. The licensee may apply a thermal annealing treatment to the reactor vessel to recover the fracture toughness of the material, as subject to the requirements in § 50.66. Licensees must submit a report describing the plan for conducting the thermal annealing at least three years before the date at which the reactor vessel would exceed the limiting fracture toughness criteria in § 50.61 or Appendix G to 10 CFR Part 50.

In § 50.66(b)(3)(ii)(B), the NRC states that the licensee must estimate the post-anneal reembrittlement trend of both the RT_{NDT} and Charpy USE and must monitor them using a

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1 Material surveillance program results are any data that demonstrate the embrittlement trends for the limiting beltline material, including but not limited to data from test reactors or from material surveillance programs at other plants, with or without a surveillance program integrated under Appendix H to 10 CFR Part 50.

material surveillance program defined in the thermal annealing report, which conforms to the intent of Appendix H to 10 CFR Part 50.

2.3.4 Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements"

Appendix G to 10 CFR Part 50 specifies requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light-water nuclear power reactors. These requirements are necessary so that there are adequate margins of safety during any condition of normal operation to which the pressure boundary may be subjected over its operating life.

Specifically, reactor vessel materials must meet the fracture toughness requirements of the ASME Code. In addition, the reactor vessel beltline materials must have an unirradiated Charpy USE of no less than 102 joules (75 foot-pounds (ft-lb)) and maintain USE throughout the life of the reactor vessel of no less than 68 joules (50 ft-lb). The NRC must approve lower values of USE, and the licensee must demonstrate that such low USE values will provide margins of safety against fracture equivalent to those required by Appendix G to Section XI of the ASME Code. Appendix K to Section XI of the ASME Code provides one approach to demonstrate equivalent margins for USE. Furthermore, ASTM 185-79 and E 185-82 define the methodology to determine the Charpy USE.

Appendix G to 10 CFR Part 50 also includes pressure-temperature limits and minimum temperature requirements for the reactor vessel that licensees must follow to ensure that the reactor vessel maintains the fracture toughness requirements for the reactor coolant pressure boundary.

2.4 Capsule Withdrawal Schedule

Appendix H to 10 CFR Part 50 requires light-water nuclear power reactor licensees to have a reactor vessel material surveillance program to monitor changes in the fracture toughness properties of the reactor vessel materials adjacent to the reactor core. The NRC requires licensees to periodically test irradiated material specimens from test capsules in their reactor vessels to evaluate changes in material fracture toughness to assess the integrity of the reactor vessel. The program must meet the design, test procedures, and reporting requirements of ASTM E 185-82, or earlier editions. The licensee establishes the number, design, and location of these surveillance capsules within the reactor vessel during the design of the program before initial plant operation. A majority of reactor licensees have already completed the withdrawal and testing of their capsules for plant operation through 40 years, while some reactor licensees have also completed the withdrawal and testing of their capsules for plant operation through 60 years.

The NRC has determined that the remaining 40 capsule withdrawals would occur between the years 2020 and 2041. The NRC assumes that the industry would withdraw the remaining capsules on the schedule defined in Table 1.

Table 1 Capsule Withdrawal Schedule

Year	No. of Capsules Scheduled for Withdrawal
2020	2
2021	2
2022	2
2023	1
2024	1
2025	5
2026	1
2027	2
2028	5
2029	4
2030	4
2031	2
2032	2
2033	2
2034	2
2035	0
2036	1
2037	0
2038	0
2039	0
2040	1
2041	1
Total	**40**

The capsule withdrawal schedule is based on information contained in a 2011 report for pressurized-water reactors, a 2012 proprietary report for boiling-water reactors, and information from Watts Bar Nuclear Plant, Unit 2, supplemented by NRC staff engineering judgment. The schedule includes withdrawals for all operating units except for those that have announced early cessation of operation, identified in Section 5.1 of this document.

3.0 Regulatory Topics

This section describes the regulatory topics used to determine whether it is necessary to amend the requirements of Appendix H to 10 CFR Part 50. The primary purpose of this direct final rule is to reduce the regulatory burden on reactor licensees and the NRC that is associated with test specimens contained within surveillance capsules and the reporting of surveillance test results, with no effect on public health and safety.

The NRC investigated the following regulatory topics:

- HAZ specimens
  - Eliminate the requirement for inclusion of weld HAZ specimens.

- Eliminate the requirement for testing weld HAZ specimens.

- Tension specimens

  - Reduce the number of tension specimens included in surveillance capsules (new or reconstituted).

  - Reduce the requirement for testing tension specimens.

  - Specify the required test temperatures for irradiated materials (i.e., at room temperature and service temperature).

- CMM

  - Specify that CMM testing is not required.

- Thermal monitors

  - Eliminate the requirement for inclusion of thermal monitors.
  - Eliminate the requirement for examining thermal monitors.

- Surveillance test results reporting

  - Extend licensee's submittal of surveillance capsule reports from 1 year to 18 months after the withdrawal of the capsule.

3.1 Heat-Affected Zone Specimens

The first regulatory topic investigated during the development of this rulemaking eliminates the requirements for (1) including HAZ specimens in new and reconstituted surveillance capsules and (2) testing HAZ specimens in existing surveillance capsules.

The editions of ASTM E 185 incorporated by reference in Appendix H to 10 CFR Part 50 specify that the surveillance test specimens shall include base metal, weld metal, and HAZ materials. Heat-affected zone specimens were first required in reactor vessel material surveillance programs in 1966 (ASTM E 185-66, "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors"). Cracks in HAZ materials had been observed to cause the failure of components in nonnuclear applications, and from early research, these failures were in HAZ materials with high hardness measurements, which is associated with low fracture toughness.

The heat-affected zone has been shown to exhibit superior fracture toughness compared to the base metal. In addition, test results from surveillance specimens have shown significant scatter of the heat-affected zone Charpy test data because of the inhomogeneous nature of the heat-affected zone material. This was the basis for eliminating the requirement for HAZ specimens after the 1994 edition of ASTM E 185, as discussed in "Irradiation Embrittlement of Reactor Pressure Vessels (RPVs) in Nuclear Power Plants" (Soneda, 2015): "Since the weld HAZ has been shown to exhibit superior fracture toughness compared to the plate or forging and does not provide relevant embrittlement data with respect to the non-HAZ weld metal, it is prudent to no longer require the inclusion or testing of HAZ specimens."

More recently, Masaki et al. (2013) investigated the continued need to include HAZ material in reactor vessel material surveillance programs. This paper investigated the features of HAZ inhomogeneity in reactor vessel steels to determine the need for surveillance test specimens of HAZ materials in Japan. The authors performed a structural integrity assessment of the inhomogeneous distribution of fracture toughness for HAZ materials using a probabilistic fracture mechanics analysis code and determined the following:

- The HAZ region close to the weld metal has coarse grain HAZ that has high toughness, causing arrest of postulated cracks.

This outcome is expected metallurgically, because the HAZ is a tempered version of the plate or forging and, as such, it should exhibit superior fracture toughness compared to the plate or forging. T.U. Marston and W. Server, in "Assessment of Weld Heat-Affected Zones in a Reactor Vessel Material" (Marston, 1978), also demonstrated this by determining that, for the conditions evaluated in the paper, the HAZs of the nuclear quality welds have higher fracture toughness than those of the parent base material.

For these reasons, the NRC is issuing a direct final rule that results in (1) no requirement for current reactor vessel material surveillance programs to test and report results for HAZ specimens or to include HAZ specimens in reconstituted or new surveillance capsules, and (2) no requirement for new reactor vessel material surveillance programs to include HAZ specimens during the design of the program.

3.2 Tension Specimens

The second regulatory topic investigated during the development of this rulemaking reduces the number of tensions specimens required (1) in new and reconstituted surveillance capsules and (2) for testing in existing surveillance capsules.

The editions of ASTM E 185 currently incorporated by reference in Appendix H to 10 CFR Part 50 specify the following with respect to tensile testing:

- For unirradiated material, tension specimens shall be tested for both the base and weld material at specified temperatures.

- For irradiated material, tension specimens shall be included for both the base and weld material and tested at specified temperatures.

- Tensile testing shall be conducted in accordance with ASTM Method E8, "Standard Test Methods for Tension Testing of Metallic Materials," and recommended practice ASTM E21, "Standard Test Methods for Elevated Temperature Tension Tests of Metallic Materials."

Testing tension specimens over a range of temperatures establishes the variation of tensile properties (e.g., yield strength, tensile strength, and elongation) with test temperatures. Performing tensile tests both before and after irradiation permits quantification of the hardening effect of irradiation using the increase in yield strength, or ΔYS. The NRC's regulations have no requirements related to strength properties. Furthermore, the NRC's regulations do not specify an approach to directly assess reactor vessel integrity from strength properties. Tensile data provide an indication of the radiation-induced strength property changes in the reactor vessel material and serve as a consistency check relative to Charpy data, in particular for cases for which the Charpy data show unexpected or inconsistent trends with prior data.

For example, McElroy and Lowe (1996) identified general correlations between shifts in fracture transition temperature and ΔYS. If the data from the Charpy tests are inconsistent, the trends described in the cited paper make it possible to predict the shift in transition temperature from the change in yield strength caused by embrittlement. In this case, a comparison of the change in yield strength with the Charpy data could provide additional information to gain an understanding of the causes for inconsistent results.

Furthermore, for optional fracture toughness testing, the calculation of relevant fracture parameters (e.g., J-integral) requires tensile data. However, the inclusion of fracture toughness specimens in surveillance capsules is optional under ASTM E 185-82. For example, ASME Code, Section XI, Code Case N-629, provides an alternative to the methods in Appendix G to 10 CFR Part 50 to allow the use of fracture toughness data in developing a master curve reference temperature (T_0) for ferritic materials in place of RT_{NDT}. Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," incorporates this code case by reference into 10 CFR 50.55a, "Codes and Standards." To use this alternative requires the yield strength of the material, which is determined from tests of tensile specimens, at the proper embrittlement level.

Experience (Westinghouse, 2015) has demonstrated that the differences in the test temperatures specified in ASTM E 185 can be small, which could yield small differences in tensile properties (e.g., the irradiated midrange transition temperature and the upper-end Charpy transition temperature can be close in value). Therefore, the requirement to test three specimens for each material at the specified temperatures could produce redundant tensile information. However, eliminating one test temperature and testing at room temperature and service temperature at all irradiation levels allows for the comparison of the change in strength properties from both irradiation and temperature.

Based on its evaluation, the NRC is issuing a direct final rule that would reduce the number of required tensile tests and tension specimens in surveillance capsules. Specifically, current reactor vessel material surveillance programs would only be required to test one tension specimen at room temperature and one tension specimen at service temperature for all materials and irradiation levels. The disposition of the remaining tension specimens in existing surveillance capsules, if any, would be at the discretion of the licensee. Furthermore, the number of tension specimens required for reconstituted and new surveillance capsules would align with the two test temperatures described above for current and new reactor vessel material surveillance programs.

3.3 Correlation Monitor Material

The third regulatory topic investigated during the development of this rulemaking is to specify that testing of CMM is optional if this material is included in existing, new, and reconstituted surveillance capsules.

A CMM is a prototypical reactor vessel material that has been fabricated to maximize homogeneous behavior, has been used in many surveillance capsules, and has an established trend from extensive testing (ASTM, DS54; IAEA, 2001; Stallman, 1987). The purpose of a CMM in a surveillance capsule is to provide reference data for comparison to the established trends for the CMM. The intent of the CMM reference data is to demonstrate that the irradiation conditions of the surveillance capsule have provided embrittlement in the CMM comparable to the established trend for the CMM. Thus, this provides additional information to understand the results from the reactor vessel materials in the surveillance capsule. The CMM is selected to have a composition and processing history comparable to the reactor vessel material. The editions of ASTM E 185 currently incorporated by reference in Appendix H to 10 CFR Part 50 specify that it is optional to include CMM in surveillance capsules. These editions of ASTM E 185 do not explicitly indicate whether licensees should test CMMs if they optionally included them in a surveillance capsule. However, ASTM E 185 contains reporting requirements for supplemental or additional specimens, which include the CMM specimens, if licensees do test them. Therefore, it is ambiguous whether correlation monitor material testing is required even though it is optional to include this material in surveillance capsules.

In practice, CMM testing has demonstrated variability in the measured material properties, which has limited the practical use of the data. Several references (Stallman, 1987; Wang, 1996; Wallin, 1999) have shown that the fitted CMM data are in general agreement with the predictions of NRC Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials" (NRC, 1988a); however, the raw CMM data exhibit significant scatter.

Based on its evaluation, the NRC is issuing a direct final rule that would not affect the design of reactor vessel material surveillance programs nor the optional inclusion of CMMs in surveillance capsules. Furthermore, the rule would specify that testing of CMMs is optional if they are included in surveillance capsules.

3.4 Thermal Monitors

The fourth regulatory topic investigated during the development of this rulemaking eliminates the requirements for (1) including temperature monitors in new and reconstituted surveillance capsules and (2) examining temperature monitors in existing surveillance capsules.

The ASTM E 185 specifies that surveillance capsules should include one set of temperature monitors within the capsule where the specimen temperature is predicted to be the maximum, but the licensee may place additional sets of temperature monitors at other locations to characterize the temperature profile. The ASTM E 185 further specifies that the licensee should determine the maximum exposure temperature of the surveillance capsule materials, and, if a discrepancy greater than 14 degrees C (25 degrees F) occurs between the observed and the expected capsule exposure temperatures, the licensee should analyze the operating conditions to determine the magnitude and duration of these differences. The standard specifies reporting of the temperature monitor results and an estimate of the maximum capsule exposure temperature.

Irradiation temperature is one of the parameters that is closely correlated with the effects of neutron embrittlement of reactor vessel steels, with lower embrittlement measured at higher irradiation temperatures within a range close to the standard operating temperature of 288 degrees C (550 degrees F). Therefore, knowledge of the irradiation temperature history of surveillance capsules is important to ensure that the surveillance data are properly interpreted and do not portray a nonconservative estimate of the reactor vessel neutron embrittlement. Typically, the temperature monitors used in surveillance capsules are high-purity, low-melting-point elements, or eutectic alloys. They are targeted to melt at specific temperatures, normally somewhat in excess of the planned operating temperature, to identify the highest temperature seen by the surveillance capsule. Some of these temperature monitors are housed in glass tubes (Westinghouse, 2011); others are in tubular aluminum alloy crucibles, which are stacked in a stainless-steel holder tube and inserted into machined locations within the aluminum spacer blocks inside the capsule (Lowe, 1999). The latter are evaluated using radiography (Lowe, 1999). These temperature monitors indicate whether the melt temperature was observed but do not provide a time-based exposure history of the monitor; thus, they are a "'go/no-go" indication of the maximum surveillance capsule temperature.

Use of temperature melt wire monitors to identify the peak capsule temperature does not provide information on the actual time-based temperature exposure conditions of the surveillance capsule, which is important to properly interpret the surveillance data. This merely indicates the highest temperature experienced by the surveillance capsule, not the duration of the exposure at that temperature. As described in Lowe (1999), several things can complicate the interpretation of the information from temperature melt wire monitors. The first complication results when the surveillance capsule experiences a short-duration thermal transient that increases the coolant inlet temperature. This could result in a positive indication from the temperature melt wire monitors, which is insignificant to the overall exposure conditions of the surveillance capsule. A second complication is caused by possible interpretation issues, where apparent "melting" of the temperature melt wire monitors is caused by long-term exposure of the monitor to temperatures near, but below, its melting point, and a resulting creep mechanism, which causes slumping of the monitor in its crucible (Lowe, 1999).

As an alternative to temperature melt wire monitors, an estimate of the average capsule temperature during full-power operation for each reactor fuel cycle would provide the irradiation temperature history of the surveillance capsule. In a typical pressurized-water reactor and boiling-water reactor, the coolant inlet temperature and the recirculation temperature, respectively, provide a reasonable estimate of the capsule irradiation temperature history. To date, licensees have been able to determine the irradiation temperature history of surveillance capsules to properly interpret the data based on the plant parameters that are already being monitored.

Based on its evaluation, the NRC is issuing a direct final rule that would result in (1) no requirement for current reactor vessel material surveillance programs to test and report results for thermal monitors or to include thermal monitors in reconstituted or new surveillance capsules, and (2) no requirement for new reactor vessel material surveillance programs to include thermal monitors during the design of the program.

3.5 Surveillance Test Results Reporting

The fifth regulatory topic investigated during the development of this rulemaking extends the time period given to a licensee following each capsule withdrawal to submit the technical report containing the test results required by ASTM E 185 and Appendix H to 10 CFR Part 50.

Appendix H to 10 CFR Part 50 currently requires that within one year of the date of the surveillance capsule withdrawal, licensees submit a summary technical report to the NRC that contains the data required by ASTM E 185, and the results of all fracture toughness tests conducted on the beltline materials in the irradiated and unirradiated conditions, unless the Director of the NRC Office of Nuclear Reactor Regulation grants an extension. The NRC first included this one-year limit in Appendix H to 10 CFR Part 50 on May 27, 1983 (48 FR 24008). The primary purpose of this requirement was the timely reporting of test results and notification of any problems determined from surveillance tests. At that time, timely reporting of surveillance data was crucial, because there was a limited amount of available data from irradiated materials from which to estimate embrittlement trends. Since the NRC first adopted this requirement, the number of commercial light-water reactors operating in the United States and the associated number of years of operation have increased significantly. This has led to an extensive amount of embrittlement data being collected and analyzed, the results of which support the reduced need for prompt reporting of the test results.

The one-year requirement to submit a report following each capsule withdrawal is a challenge for some licensees, particularly those participating in the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program. Implementation of this integrated surveillance program requires significant coordination among the multiple licensees participating in the program. In general, these licensees continue to request a 6-month extension to the 1-year reporting requirement, and, to date, the Director of the NRC Office of Nuclear Reactor Regulation has approved these requests. In addition, as surveillance capsules remain in the reactor vessel to achieve higher neutron fluence levels to support plant operation through 60 years and 80 years, longer periods of radioactive decay may be necessary before the capsule can be shipped to hot-cell laboratories for testing.

The purpose of issuing a direct final rule change to the reporting requirement is to reduce the regulatory burden for licensees to submit and for the NRC to review these extension requests, while still ensuring adequate protection of public health and safety. Furthermore, increasing the time given to licensees to submit a summary report following each capsule withdrawal from 1 year to 18 months is appropriate, because (1) a significant number of test specimens have been analyzed since 1983, the results of which support this change, and (2) this is a reasonable accommodation of the extension period requested previously by licensees.

Based on its evaluation, the NRC is issuing a direct final rule that would afford reactor licensees 18 months following the withdrawal of a surveillance capsule to submit the capsule report to the NRC. Therefore, the need for reactor licensees participating in the BWRVIP Integrated Surveillance Program to submit extension requests to the reporting requirements would be substantially reduced because of administrative challenges, and the NRC would review fewer requests.

4.0 Description of Alternatives

This section considers two alternatives for amending the requirements of Appendix H to 10 CFR Part 50 that the NRC identified in the regulatory basis associated with test specimens contained within surveillance capsules and the reporting of surveillance test results:

(1) no action (status quo) [not Selected]
(2) direct final rule to revise Appendix H to 10 CFR Part 50 [selected]

4.1 Alternative 1—No Action (Status Quo)

The no action (status quo) alternative is a non-rulemaking alternative. This alternative would retain the current requirements in Appendix H to 10 CFR Part 50 and the specimens and testing required by ASTM E 185-73, E 185-79, and E 185-82, as applicable. Licensees would continue to (1) test Charpy impact specimens for the weld HAZ, (2) test tension specimens for the weld metal and base metal at various temperatures, (3) test correlation monitors, if they were included, and (4) examine thermal monitors in each surveillance capsule in accordance with ASTM E 185-82, to the extent practicable. Licensees needing additional time to submit their surveillance capsule reports would continue to submit extension requests for NRC review and approval.

4.2 Alternative 2: Direct Final Rule to Revise Appendix H to 10 CFR Part 50

Under this alternative, the NRC would prepare a direct final rule to revise the underlying regulations to alleviate the regulatory burden on existing licensees and future applicants. These revisions that follow would not impose any additional requirements for the current fleet of operating reactors:

- HAZ specimens
  - Eliminate the requirement for inclusion of weld HAZ specimens.
  - Eliminate the requirement for testing weld HAZ specimens.
- Tension specimens
  - Reduce the number of tension specimens included in surveillance capsules (new or reconstituted).
  - Reduce the requirement for testing tension specimens.
  - Specify the required test temperatures for irradiated materials (i.e., at room temperature and service temperature).
- CMM
  - Specify that CMM testing is not required.
- Thermal monitors
  - Eliminate the requirement for inclusion of thermal monitors.
  - Eliminate the requirement for examining thermal monitors.
- Surveillance test results reporting
  - Extend submittal of surveillance capsule reports to 18 months after the withdrawal of the capsule.

Table 2 summarizes the applicability of these changes to nuclear power reactors applicants and licensees.

Table 2 Applicability of Final Changes to Applicants and Licensees

Description of Change	Current and Future Power Reactor Applicants	Power Reactor Licensees
HAZ specimens	Eliminate HAZ specimen inclusion in capsules.	Eliminate HAZ inclusion in new and reconstituted capsules. Eliminate testing HAZ specimens for existing capsules.
Tension specimens	Reduce tensile specimens in capsules. Specify the required test temperatures for irradiated materials.	Reduce tensile specimens in new and reconstituted capsules. Reduce tensile testing. Specify the required test temperatures for irradiated materials.
Correlation monitor materials	Eliminate correlation monitor testing.	Eliminate correlation monitor testing for existing capsules.
Thermal monitors	Eliminate thermal monitor inclusion in capsules.	Eliminate thermal monitor inclusion in new and reconstituted capsules. Eliminate examination of thermal monitors in existing capsules.
Surveillance test results reporting	Extend reporting requirements from 1 year to 18 months following each capsule withdrawal.	Extend reporting requirements from 1 year to 18 months following each capsule withdrawal.

In conducting the cost-benefit analysis, the NRC evaluated the direct final rule process against the status quo alternative.

Under the direct final rule alternative, the NRC would use the direct final rule process[2] to revise the underlying regulations to alleviate the regulatory burden on existing licensees and future applicants as described above. The NRC staff assumes that the effective date for the direct final rule process is 2020, and the industry would incur benefits beginning in 2020.

4.3 Other Alternatives Considered

The NRC considered another alternative during the regulatory basis phase that it eliminated at that time. This alternative considered incorporating by reference the 2016 editions of ASTM E 185 and ASTM E 2215 into Appendix H of 10 CFR Part 50 but concluded that the burden associated with implementing these ASTM standards would be significant without a corresponding benefit to public health and safety. As a result, the NRC concluded that this alternative was not viable and so did not considered it further. The regulatory basis for this rule (NRC, 2018c) contains more information.

2 The public usually has 30 days to comment after publication of the direct final rule.

5.0 Estimation and Evaluation of Benefits and Costs

5.1 Methodology and Assumptions

The potential costs and benefits of the alternatives must be considered for light-water power reactor licensees and the NRC. The analyses in this section are based on the NRC's assessment and input, as well as on input from external stakeholders.

5.1.1 Analysis Baseline

The analyses in this section present the incremental costs and benefits that the licensees and the NRC would realize from the rulemaking action. Incremental costs and benefits are calculated values that are above the status quo condition (Alternative 1). The status quo condition for this rulemaking action includes the benefits and costs to comply with the current requirements in Appendix H to 10 CFR Part 50.

The NRC examined the direct final rule process. Table 3 shows the direct final rule activities.

Table 3 Alternative 2 Rulemaking Activities

Rulemaking Phase	Direct Final Rule Activities
Rulemaking	Develop the direct final rule and companion final rule.
	Publish the direct final rule with the companion rule for public comment.
	Verify there are no significant adverse comments,[a] resolve public comments, withdraw final rule.
	Prepare confirmation.

[a] If significant adverse comments are received, the NRC would withdraw the direct final rule and could proceed either using the standard notice and comment rule process or recommend to the Commission that rulemaking activities should cease because the activity is not cost justified.

5.1.2 Affected Facilities

The NRC estimates that the direct final rule will cover all U.S. commercial light-water reactor operating units and units under construction.[3] However, as of August 2019, the following plants have shut down or have announced plans to permanently shut down before their license expiration:

- Oyster Creek Nuclear Generating Station was shut down in September 2018.
- Pilgrim Nuclear Power Station was shut down in May 2019.
- Three Mile Island Nuclear Station, Unit 1, plans to shut down by September 30, 2019.
- Duane Arnold Energy Center plans to shut down in late 2020.
- Indian Point Nuclear Generating, Units 2 and 3, plan to shut down by April 30, 2021.
- Beaver Valley Power Station, Units 1 and 2, plan to shut down by October 31, 2021.
- Palisades Nuclear Plant plans to shut down by spring of 2022.

[3] This analysis does not include reactor units that have received a construction permit or a combined license that are not currently under construction.

- Diablo Canyon Power Plant, Units 1 and 2, plan to shut down in 2025.

The analysis evaluates the incremental costs and benefits on a per-unit basis for all operating units except for those facilities that have announced early cessation of operations. Additionally, some units have completed their capsule withdrawals under their reactor vessel material surveillance program and would not experience any burden reduction.

5.1.3 Base Year

All monetized costs are expressed in 2020 dollars. The analysis assumes that ongoing costs of operation related to the alternative being analyzed will begin no earlier than 30 days after publication of the direct final rule in the NRC's regulations unless otherwise stated. The analysis assumes that the publication of the direct final rule will occur in 2020. The timeframe for the base case analysis runs from 2020 through 2041. The sensitivity analysis timeframe runs from 2020 through 2048.

This analysis does not include NRC costs to develop the direct final rule and supporting guidance because these costs are considered sunk costs. The NRC estimated the industry's implementation costs and the recurring annual operating costs for the NRC and the industry. The values for annual operating expenses are modeled as a constant expense for each year of the analysis horizon. The NRC performed a discounted cash flow calculation to discount these annual expenses to 2020-dollar values.

5.1.4 Discount Rates

In accordance with NUREG/BR 0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," draft Revision 5 (NRC, 2018b), net present value (NPV) calculations determine how much society would need to invest today to ensure that the designated dollar amount is available in a given year in the future. By using NPVs, costs and benefits are valued to a reference year for comparison, regardless of when the cost or benefit is incurred in time. The choice of a discount rate and its associated conceptual basis is a topic of ongoing discussion within the Federal Government. Based on U.S. Office of Management and Budget (OMB) Circular No. A-4, "Regulatory Analysis," dated September 17, 2003 (OMB, 2003), and consistent with NRC past practice and guidance, present-worth calculations in this analysis use 3-percent and 7-percent real discount rates. A 3-percent discount rate approximates the real rate of return on long-term Government debt, which serves as a proxy for the real rate of return on savings to reflect reliance on a social rate of time preference discounting concept. A 7-percent discount rate approximates the marginal pretax real rate of return on an average investment in the private sector and is the appropriate discount rate whenever the main effect of a regulation is to displace or alter the use of capital in the private sector. A 7-percent rate is consistent with an opportunity cost of capital concept to reflect the time value of resources directed to meet regulatory requirements.

5.1.5 Cost/Benefit Inflators

$$\frac{CPI - U_{2020}}{CPI - U_{Base\ Year}}\ x\ Value_{Base\ Year} = Value_{2020}$$

Table 4 summarizes the consumer price index for all urban consumers (CPI-U) values used in this regulatory analysis.

Table 4 CPI-U Inflator

Base Year	CPI-U Annual Average
2017	245.14
2018	251.38
2019	257.50
2020	263.00

Source: (Statistica, 2019) (http://www.statista.com/statistics/244993/projected-consumer-price-index-in-the-united-states/)

5.1.6 Labor Rates

For the purposes of this regulatory analysis, the NRC developed costs for the various licensee tests that were expected to be eliminated, which included industry labor costs. The NRC estimated the loaded labor rate to be $129 per hour and calculated it based on 2018 NRC labor and benefit cost data.

5.1.7 Sign Conventions

In this analysis, all favorable consequences for the alternative are positive and all adverse consequences for the alternative are negative. Negative values use parentheses (e.g., negative $500 is displayed as ($500)).

5.1.8 Identification of Affected Attributes

The NRC evaluated the following attributes in support of this regulatory basis.

- NRC implementation
- industry implementation
- industry operation
- NRC operation

6.0 Presentation of Results

This section presents the benefits and costs estimated for the regulatory options. To the extent that the NRC could analyze the affected attributes quantitatively, it has calculated the net effect of each option and presented it below. However, it could evaluate some values and impacts only on a qualitative basis.

6.1 NRC Implementation

The NRC's rule development costs are sunk costs and are not included in this analysis.

6.2 Industry Implementation

6.2.1 Alternative 1—No Action: Industry Implementation Costs

This alternative would maintain the current requirements in Appendix H to 10 CFR Part 50 (i.e., status quo) and as such would have no incremental impact on the industry.

Although there is no incremental impact on licensees, this alternative would result in continued expenditures by licensees or future applicants that are associated with testing or examining capsule specimens that do not provide beneficial surveillance data or support direct regulatory needs to assess and monitor embrittlement of the reactor vessel. Furthermore, licensees or future applicants that participate in an integrated surveillance program will likely continue to submit extension requests for submittal of test results within the one-year requirement of the capsule withdrawal because of the significant coordination needed among multiple licensees participating in the integrated surveillance program and with hot-cell laboratories.

6.2.2 Alternative 2—Direct Final Rule Process: Industry Implementation Costs

The NRC assumes that there are little to no industry implementation costs for rulemaking material review and comment because of the noncontroversial nature of the direct final rule changes.

6.3 Industry Operations Cost

The industry would avert costs in Alternative 2, direct final rule process, resulting from the following and as further described below:

- HAZ specimens
  - Eliminate the requirement for inclusion of weld HAZ specimens.
  - Eliminate the requirement for testing weld HAZ specimens.
- Tension specimens
  - Reduce the number of tension specimens included in surveillance capsules (new or reconstituted).
  - Reduce the requirement for testing tension specimens.
  - Specify the required test temperatures for irradiated materials (i.e., at room temperature and service temperature).
- CMM
  - Specify that CMM testing is not required.
- Thermal monitors
  - Eliminate the requirement for inclusion of thermal monitors.
  - Eliminate the requirement for examining thermal monitors.
- Surveillance test results reporting
  - Extend submittal of surveillance capsule reports to 18 months after the withdrawal of the capsule.

Heat-Affected Zone Specimens

Licensees of operating reactor units would realize incremental savings if they were no longer required to test HAZ test specimens upon the withdrawal of each surveillance capsule and if they were no longer required to include HAZ test specimens in reconstituted or new surveillance capsules.

Applicants for a reactor license that will seek NRC review and approval for a reactor vessel material surveillance program would realize incremental savings if they were not required to include HAZ test specimens in new surveillance capsules.

Based on industry input, the NRC estimates that the HAZ specimen testing averted cost is $8,046 per withdrawn capsule.

Tension Specimens

Licensees of operating reactor units and applicants for a reactor license that will seek NRC review and approval for a reactor vessel material surveillance program would realize incremental savings, resulting from the reduction in the number of required tensile tests and tension specimens in surveillance capsules. The disposition of the remaining tension specimens in existing surveillance capsules, if any, would be at the discretion of the licensee.

Specifically, licensees of operating reactor units would only be required to test one tension specimen at room temperature and one tension specimen at service temperature for all materials and irradiation levels. As mentioned above, the disposition of the remaining tension specimens, if any, in existing surveillance capsules would be at the discretion of the licensee. Furthermore, the number of tension specimens required for reconstituted and new surveillance capsules would align with the two test temperatures described above for licensees of operating reactor units and applicants for a reactor license that will seek NRC review and approval for a reactor vessel material surveillance program.

Based on ASTM E 185-82, each capsule is required to contain three tension specimens for each material (i.e., base and weld). The rule would eliminate testing of one of these three specimens for each material. The remaining two specimens for each material would still require testing at the test temperatures specified above. The NRC assumed that the cost to test two tension specimens is two-thirds of the cost to test the three required specimens, for each material (e.g., this would avert one-third of the current tensile test cost).

Based on industry input, the NRC estimates that the tension specimen test averted cost is $2,682 per withdrawn capsule, based on the assumption that one-third of the current tensile test would no longer require testing and that the cost for tensile testing is $8,046 per withdrawn capsule.

Correlation Monitor Materials

Licensees of operating reactor units and applicants for a reactor license that will seek NRC review and approval for a reactor vessel material surveillance program would realize incremental savings. This would result from explicitly specifying that testing of CMM specimens, if included in existing, reconstituted, or new surveillance capsules, is optional upon the withdrawal of a surveillance capsule.

Based on industry input, the NRC estimates that the CMM specimen testing averted cost is $8,046 per withdrawn capsule. However, since CMM specimens are optionally included in surveillance capsules, the NRC assumed that only 40 percent of the remaining surveillance capsules contain CMM specimens. The NRC's assumption is based on a sampling of surveillance capsule reports submitted by the licensees.

Thermal Monitors

Licensees of operating reactor units would realize incremental savings if they were no longer required to (1) examine thermal monitors upon the withdrawal of each surveillance capsule and (2) include thermal monitors in reconstituted or new surveillance capsules.

Applicants for a reactor license that will seek NRC review and approval for a reactor vessel material surveillance program would realize incremental savings if they were not required to include thermal monitors in new surveillance capsules.

Based on industry input, the NRC estimates that the thermal monitor testing averted cost is $2,682 per withdrawn capsule.

Surveillance Test Results Reporting

Appendix H to 10 CFR Part 50 requires light-water nuclear power reactor licensees to have a reactor vessel material surveillance program to monitor changes in the fracture toughness properties of the reactor vessel materials adjacent to the reactor core. The NRC requires licensees to periodically test irradiated material specimens from test capsules in their reactor vessels to evaluate changes in material fracture toughness properties to assess the integrity of the reactor vessel. The program must meet the design, test procedures, and reporting requirements of ASTM E 185-82, or earlier editions. The licensee establishes the number, design, and location of these surveillance capsules within the reactor vessel during the design of the program before initial plant operation.

This direct final rule permits reactor licensees having an NRC-approved reactor vessel material surveillance program an additional 6 months to submit their report of surveillance testing following the withdrawal of each surveillance capsule, compared to the current Appendix H to 10 CFR Part 50 requirements.

Those licensees that participate in an integrated surveillance program—specifically, operating boiling-water reactors—would recognize a substantial reduction in the need to submit extension requests for the report on surveillance testing following each capsule withdrawal to accommodate internal processes established by the BWRVIP (i.e., committee review process). These licensees would be relieved of the administrative and financial burden associated with submitting the extension requests.

Applicants for a reactor license that will seek NRC review and approval for a reactor vessel material surveillance program will have 18 months to submit the report of surveillance testing following the withdrawal of each surveillance capsule.

Based on industry input, the NRC estimates that the cost for a licensee to prepare and submit a schedule extension request for a surveillance capsule test report is $21,457. Based on a review of previously submitted schedule requests and proprietary industry information, the NRC estimates that the direct final rule would avert 18 schedule extension requests during the years 2020 to 2041.

6.3.1 Alternative 1—No Action: Industry Operation Costs

This alternative would have no incremental impact on the industry. However, some reactor licensees would continue to be required to prepare and submit extension requests for submittal of test results within one year of the capsule withdrawal that have generally been associated with licensees participating in integrated surveillance programs.

6.3.2 Alternative 2—Industry Testing Cost

Under this alternative, the operating reactor units with remaining capsules would begin to realize the averted costs in 2020, when the direct final rule becomes effective. Based on Table 1, licensees would withdraw 40 capsules beginning in 2020 and continuing through 2041. Table 5 shows the resulting averted cost savings.

Table 5 Industry Operation Costs (Direct Final Rule)

Years[a]	Description	No. of Units	Unit Cost	Undiscounted	Net Present Value	
					7% Discount Rate	3% Discount Rate
2020–2041	Industry HAZ tests	40	$8,046	$321,857	$193,707	$254,609
	Industry tension specimen tests	40	$2,682	$107,286	$64,569	$84,870
	Industry CMM tests	16	$8,046	$128,743	$77,483	$101,844
	Industry thermal monitor tests	40	$2,682	$107,286	$64,569	$84,870
	Industry report submittal extension	18	$21,457	$386,228	$208,431	$290,012
			Total[b]	**$1,051,399**	**$608,758**	**$816,204**

[a] The year 2020 represents a future year when a new reactor licensee recognizes a cost savings (averted cost) resulting from changes to current requirements.
[b] Totals may not match due to rounding.

6.4 NRC Operation Costs

The following changes to Appendix H to 10 CFR Part 50 through a direct final rule would result in the following changes in incremental operation costs to the NRC:

- Eliminate the NRC review of HAZ specimen test results.
- Reduce the NRC review of tension specimen test results, except at room temperature and service temperature.
- Eliminate the NRC review of CMM test results.
- Eliminate the NRC review of thermal monitor test results.
- Reduce the need for the NRC to review routine licensee schedule extension requests related to the submittal of the surveillance capsule reports.

Heat-Affected Zone Specimens

Because the HAZ testing is eliminated, the NRC would realize averted costs. The NRC estimates that the NRC would save two hours per capsule by eliminating the review of HAZ test result submittals.

Tension Specimens

Because the tensile testing is reduced, the NRC would realize averted costs. The NRC estimates that it would save two hours per eliminated tension test.

Correlation Monitor Material

Because CMM testing is eliminated, the NRC would realize averted costs. The NRC estimates that it would save two hours per eliminated CMM test.

Thermal Monitors

Because the requirement to examine thermal monitors is reduced, the NRC would realize averted costs. The NRC estimates that it would save two hours per eliminated thermal monitor examination.

Surveillance Test Results Reporting

Reducing the need for licensees to request extensions by increasing the allowable time to submit the test results from 1 year to 18 months would realize averted costs for the NRC to review and approve these requests. The NRC estimates that it would save 60 hours per averted schedule extension request.

6.4.1 Alternative 1—No Action: NRC Operation Costs

This alternative would have no incremental impact on the NRC. However, the NRC would continue to be required to review extension requests for submittal of test results within one year of the capsule withdrawal that have generally been associated with licensees participating in integrated surveillance programs.

6.4.2 Alternative 2—Direct Final Rule Process: NRC Operation Costs

Under this alternative, the operating reactor units with remaining capsules would begin to realize the averted cost savings in 2020, when the direct final rule becomes effective. Table 6 shows the resulting NRC cost savings.

Table 6 NRC Operation Costs

Years[a]	Description	No. of Submissions	No. of Review Hours	Hourly Rate	Undiscounted	Net Present Value	
						7% Discount Rate	3% Discount Rate
2020–2041	NRC HAZ tests	40	2	$129	$10,320	$6,211	$8,164
	NRC tension specimen tests	40	2	$129	$10,320	$6,211	$8,164
	NRC CMM tests	16	2	$129	$4,128	$2,484	$3,266
	NRC thermal monitor tests	40	2	$129	$10,320	$6,211	$8,164
	NRC report submittal extension	16	60	$129	$123,840	$71,316	$96,167
				Total[b]	**$158,928**	**$92,433**	**$123,924**

a The year 2020 represents a future year when a new reactor licensee recognizes a cost savings (averted cost) resulting from changes to current requirements.
b Totals may not match due to rounding.

6.5 Uncertainty Analysis

The NRC completed a Monte Carlo sensitivity analysis for this regulatory basis using the specialty software @Risk®.[4] The Monte Carlo approach answers the question, "What distribution of net benefits results from multiple draws of the probability distribution assigned to key variables?"

As this regulatory basis uses estimates of values that are sensitive to plant-specific cost drivers and plant dissimilarities, the NRC provides the following analysis of the variables that have the greatest amount of uncertainty.

Monte Carlo simulations involve introducing uncertainty into the analysis by replacing the point estimates of the variables used to estimate base case costs and benefits with probability distributions. By defining input variables as probability distributions instead of point estimates, the influence of uncertainty on the results of the analysis (in other words, the net benefits) can be effectively modeled.

The probability distributions chosen to represent the different variables in the analysis were bounded by the range-referenced input and the NRC's professional judgment. Defining the probability distributions for use in a Monte Carlo simulation requires summary statistics to characterize the distributions. These summary statistics include the minimum, most likely, and maximum values of a program evaluation and review technique (PERT) distribution.[5] The NRC

4 Information about this software is available at http://www.palisade.com.

5 A PERT distribution is a special form of the beta distribution with specified minimum and maximum values. The shape parameter is calculated from the defined *most likely* value. The PERT distribution is similar to a triangular distribution, in that it has the same set of three parameters. Technically, it is a special case of a scaled beta (or beta general) distribution. The PERT distribution is generally considered superior to the triangular distribution when the parameters result in a skewed distribution, as the smooth shape of the curve places less emphasis in the direction of skew. Similar to the triangular distribution, the PERT distribution is bounded on both sides and, therefore, may not be adequate for some modeling purposes if it is desired to capture tail or extreme events.

used the PERT distribution to reflect the relative spread and skewness of the distribution defined by the three estimates.

The NRC performed the Monte Carlo simulation by repeatedly recalculating the results, 5,000 times. For each iteration, the values were chosen randomly from the probability distributions that define the input variables. The NRC recorded the values of the output variables for each iteration and used these resulting output variable values to define the resultant probability distribution. Figure 1 through Figure 3 display the probability distribution function and the descriptive statistics of the incremental benefits and costs of the direct final rule alternative (Alternative 2), compared to the no-action alternative (Alternative 1). Figure 1 displays the probability distribution function and the descriptive statistics of the industry incremental benefits and costs of the direct final rulemaking alternative (Alternative 2), compared to the no-action alternative (Alternative 1). The analysis shows that, for the industry, if Alternative 2 is selected, the direct final rule process is cost beneficial with a mean value of $608,759.

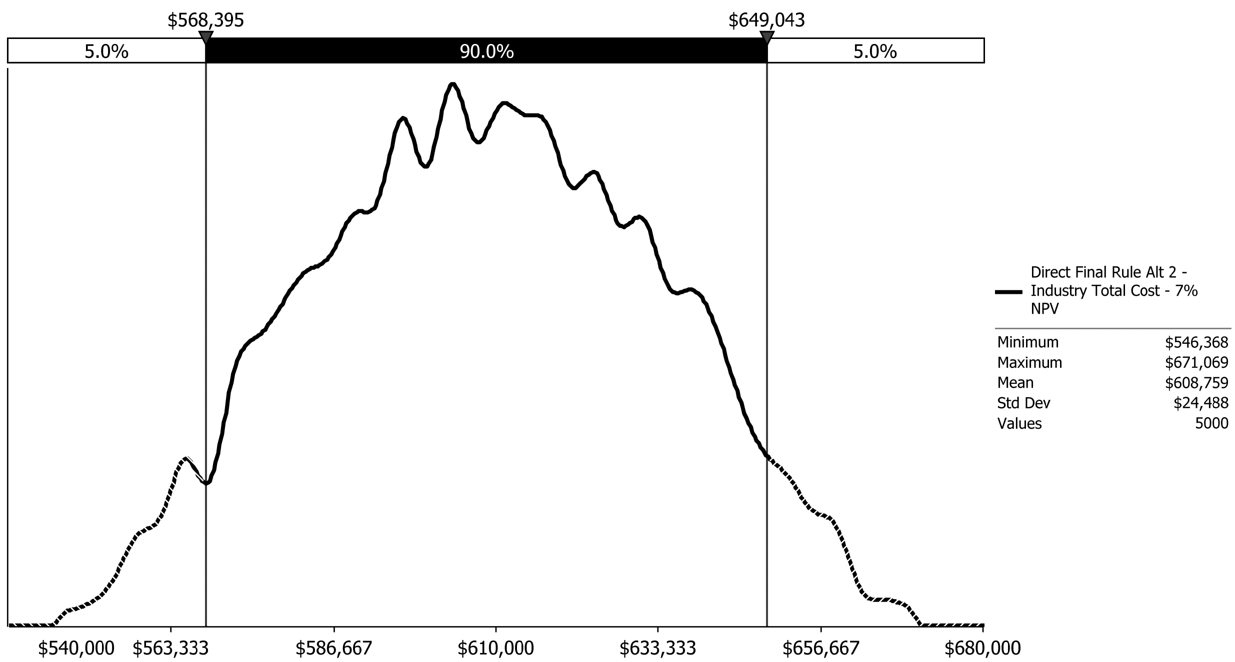

Figure 1 Industry Total Averted Costs—7-Percent Net Present Value (2020$)

Figure 2 displays the probability distribution function and the descriptive statistics of the NRC incremental benefits and costs of the direct final rulemaking alternative (Alternative 2), compared to the no-action alternative (Alternative 1). The analysis shows that, for the NRC, if Alternative 2 is selected, the direct final rule process is cost beneficial with a mean value of $98,155.

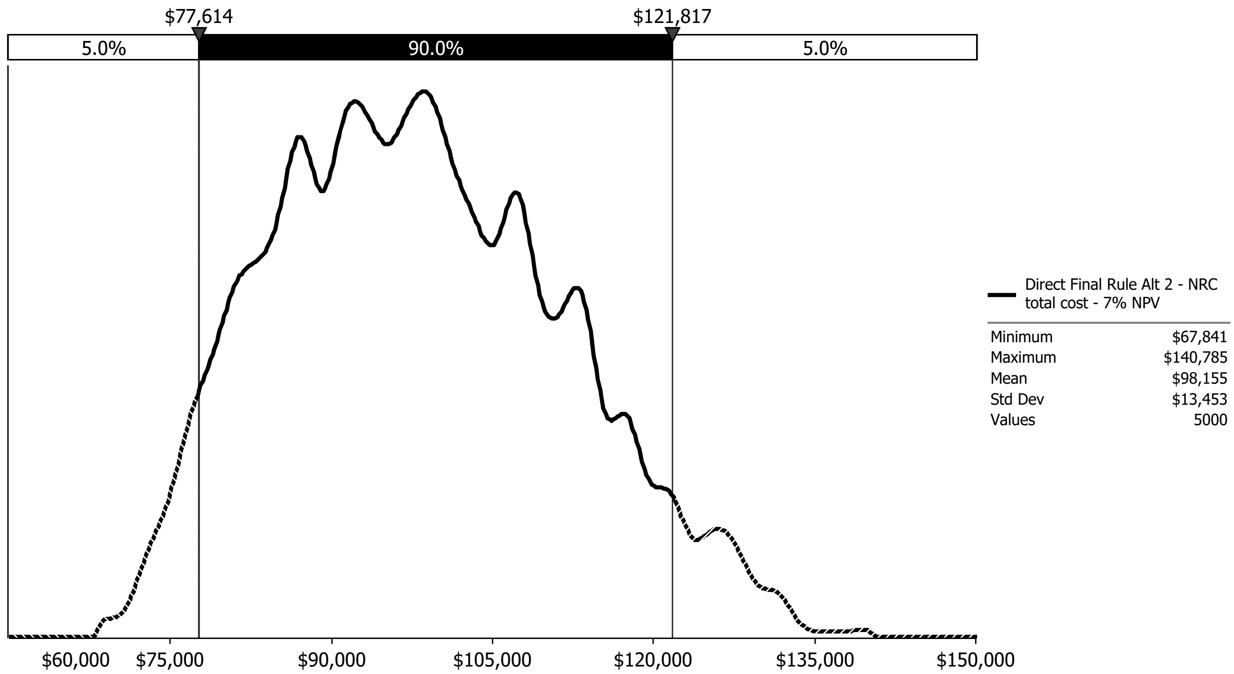

Figure 2 NRC Total Costs—7-Percent Net Present Value (2020$)

Figure 3 displays the probability distribution function and the descriptive statistics of the total incremental benefits and costs of the direct final rule alternative (Alternative 2), compared to the no-action alternative (Alternative 1). The analysis shows that if Alternative 2 is selected, the direct final rule process is cost beneficial with a mean value of $706,913.

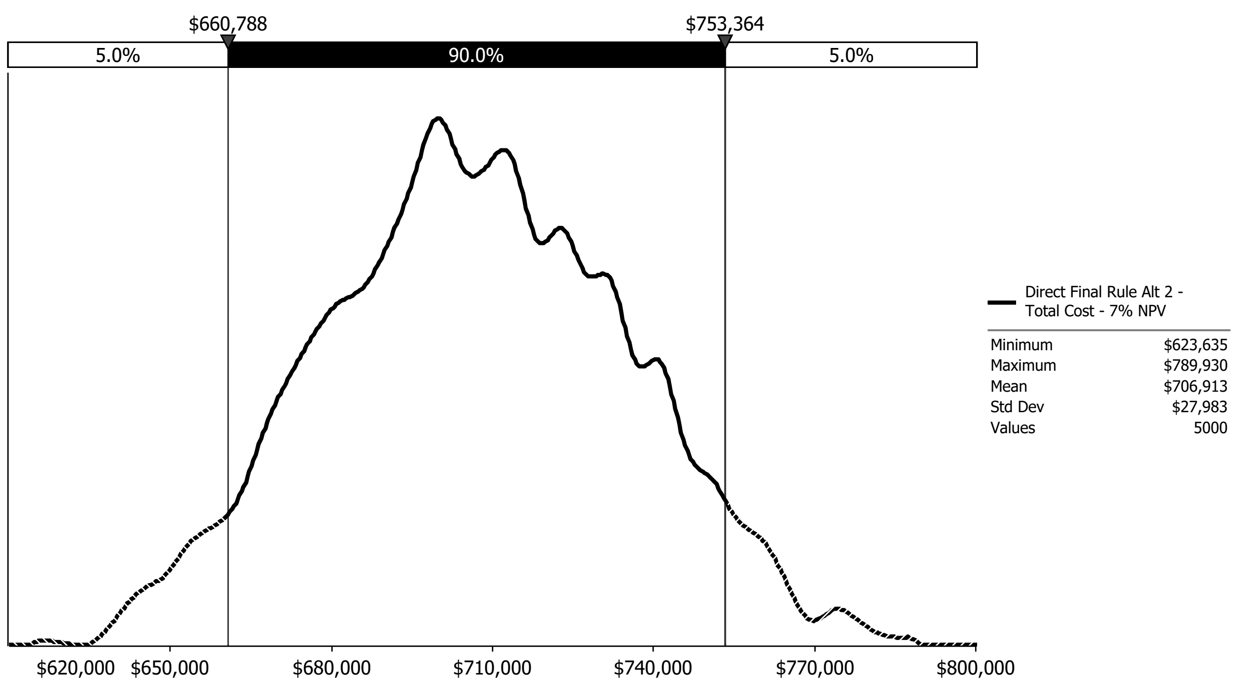

Figure 3 Total Averted Costs—7-Percent Net Present Value (2020$)

6.6 Sensitivity Analysis

In addition to estimating the probability distributions for the net benefits of the direct final rule, the NRC used Monte Carlo simulation to conduct a sensitivity analysis to determine the variables with the greatest impact on the resulting net benefits. Variables shown to have a large effect on the resulting net benefits may deserve more attention and scrutiny than variables shown to have a small or minimal effect.

To estimate the effect of each variable on the net benefits, the NRC performed a regression with the net benefits as the dependent variable and the inputs as the independent variables. The result of this regression, called a tornado diagram, presents in vertical order the variables with the greatest influence on net benefits. Figure 4 displays a tornado diagram for the total costs of the direct final rule and ranks the variables based on their contribution to cost uncertainty on the mean value. Three variables—the industry HAZ specimen testing costs, the NRC costs to review the licensee requests for extensions, and NRC thermal monitor test review costs—cause the greatest uncertainty in the costs.

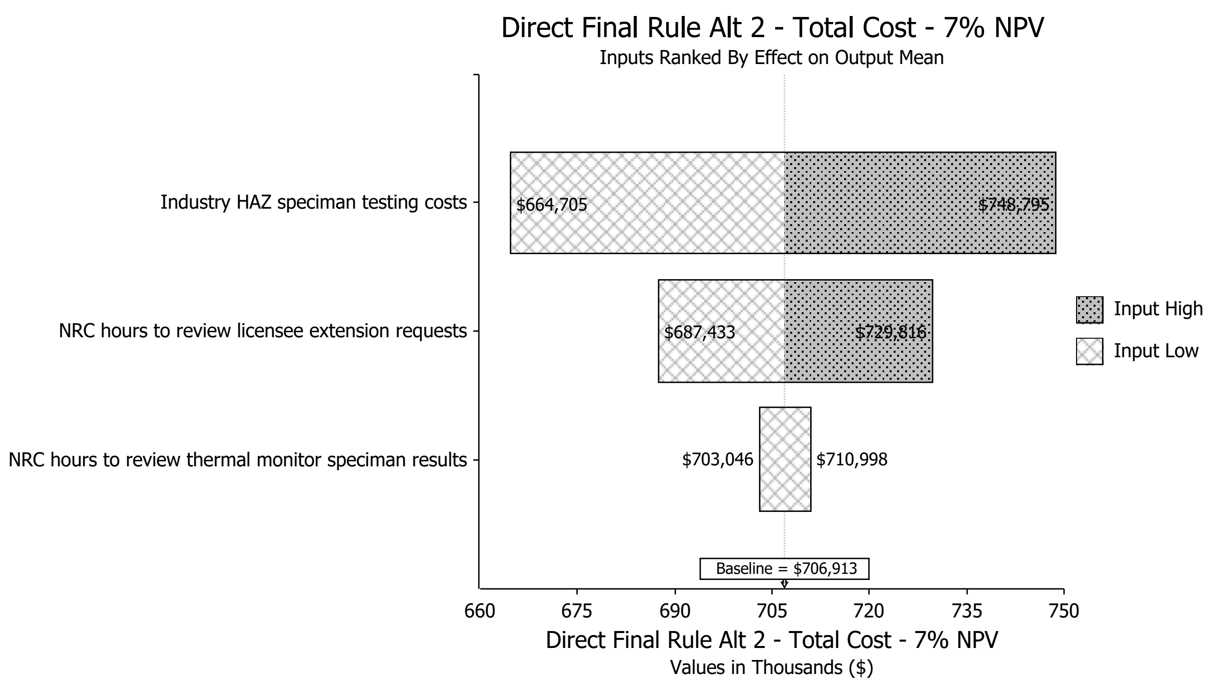

Figure 4 Tornado diagram—Total Averted Costs of the Final Rule at 7-Percent Discounting (2020$)

The NRC performed a sensitivity analysis to evaluate the impact of delaying the removal and testing of the remaining capsules by 7 years to model the possible impact of second license renewals. Table 7 shows the effect of delaying the removal and testing of the remaining capsules results in a reduction in the net benefit of the direct final rule of ($264,525), assuming a 7-percent discount rate.

Table 7 Effect of Delaying Capsule Withdrawals To Account for Second License Renewal Requests

Description	**Base Case (2020 to 2041) (A)**	**Sensitivity Case (2020 to 2049) (B)**	**Difference (C = B – A)**
Discount Rate	**7% NPV**	**7% NPV**	**7% NPV**
Reactor Vessel Pulls	40	40	0
Industry Implementation	$0	$0	$0
Industry Operation	$608,758	$379,104	($229,654)
Industry Total	*$608,758*	*$379,104*	*($229,654)*
NRC Implementation	$0	$0	$0
NRC Operation	$92,433	$57,563	($34,870)
NRC Total	*$92,433*	*$57,563*	*($34,870)*
Total	**$701,191**	**$436,667**	**($264,525)**

6.7 Summary

Table 8 displays the quantified incremental net benefits for the direct final rule alternative as compared to the no-action alternative.

Table 8 Summary Table

Net Monetary Savings (or Costs)	Nonquantified Benefits and Costs
Option 1: No Action $0	*<u>Nonquantified Benefits and Costs</u>* None
Option 2: Direct Final Rule: **Industry Implementation:** $0 **Industry Operation:** $608,758 using a 7-percent discount rate $816,204 using a 3-percent discount rate **Industry Total** $608,758 using a 7-percent discount rate $816,204 using a 3-percent discount rate **NRC Implementation:** $0 **NRC Operations:** $92,433 using a 7-percent discount rate $123,924 using a 3-percent discount rate **NRC Total:** $92,433 using a 7-percent discount rate $123,924 using a 3-percent discount rate **Total Net:** $701,191 using a 7-percent discount rate $940,128 using a 3-percent discount rate	*<u>Nonquantified Benefits:</u>* <u>Independence</u>. Final decisions would be based on objective, unbiased assessments of all information and would be documented with reasons explicitly stated. <u>Openness</u>. The direct final rule would be transacted publicly and candidly. The NRC engaged the regulated community, the public, and other interested stakeholders through public meetings during the early development of the regulatory basis and the direct final rule to ensure that diverse views were considered in the regulatory decision making process. <u>Efficiency</u>. The direct final rule would reduce the regulatory burden on reactor licensees and the NRC that are associated with test specimens contained within surveillance capsules and the reporting of surveillance test results. The direct final rule would continue to ensure protection of public health and safety. <u>Clarity</u>. The direct final rule to revise Appendix H to 10 CFR Part 50 would result in coherent, logical, and practical regulations. The revised requirements would be readily understood and easily applied. <u>Reliability</u>. The direct final rule would result in regulations that are based on the best available knowledge from research and operational experience. This information would be used to revise the requirements in Appendix H to 10 CFR Part 50 to lend stability to the design and implementation of a reactor vessel material surveillance program. *<u>Nonquantified Costs</u>*: None were identified.

[a] Benefits and averted costs are positive. Costs are (negative).
[b] The NRC staff took credit for averted costs beginning in 2020.

As shown in Table 8, Alternative 2 reduces industry costs. Based on this estimate, the Alternative 2 direct final rule would result in estimated averted costs to the industry that range from $608,758 using a 7-percent discount rate to $816,204 using a 3-percent discount rate.

Likewise, the NRC would realize reduced operation costs by eliminating certain testing requirements and reducing the need to review and approve schedule extensions for submitting reactor vessel specimen test results. Based on this estimate, Alternative 2 (direct final rule process) results in estimated averted costs to the NRC that range from $92,433 using a 7-percent discount rate to $123,924 using a 3-percent discount rate. This analysis does not include NRC implementation costs because they are sunk costs.

In addition to the quantified costs discussed in this regulatory analysis, the attributes of public confidence and improvements in knowledge would produce nonquantified benefits for the industry and the NRC. The direct final rule supports the NRC's 2018–2022 Strategic Plan (NRC, 2018a) in relation to the five principles of good regulation as described in Table 8.

7.0 Decision Rationale

The cost-benefit analysis evaluated two alternatives. Alternative 1, the no-action alternative, would maintain the current requirements in Appendix H to 10 CFR Part 50 (i.e., status quo) and, as such, the specimens and testing required by ASTM E 185-73, E 185-79, and E 185-82, as applicable. Alternative 1 avoids the costs that the direct final rule would impose; however, the NRC will continue to require licensees to (1) test Charpy impact specimens for the weld HAZ, (2) test tension specimens for the weld metal and base metal at various temperatures, (3) test CMMs, if they were included, and (4) examine thermal monitors in each surveillance capsule in accordance with ASTM E 185-82, to the extent practicable. Furthermore, licensees that need additional time to submit their surveillance capsule reports would continue to submit extension requests for NRC review and approval.

Under Alternative 2, the NRC would prepare a direct final rule to revise the underlying regulations to alleviate the burden to existing licensees and to future applicants with no reduction to public health and safety. This alternative achieves the objective of maximizing the burden reduction for the reactor vessel material surveillance program while maintaining a comparable level of safety. This alternative also has the advantage of being relatively simple to implement.

Table 8 shows that the direct final rule would result in estimated costs to the NRC that range from $92,433 using a 7-percent discount rate to $123,924 using a 3-percent discount rate.

The NRC also observed that the remaining number of surveillance capsules in the existing fleet of commercial nuclear power reactors is a small fraction of the total number that licensees have already withdrawn and tested, given the maturity of reactor vessel material surveillance programs. Therefore, the opportunity to reduce licensee burden associated with the reactor vessel material surveillance test program can be maximized only if the NRC completes the rulemaking effort under the direct final rule process.

7.1 Backfitting and Issue Finality

The NRC’s backfitting provisions for holders of construction permits, and applicants and holders of operating licenses, appear in § 50.109, “Backfitting” (the Backfit Rule). Issue finality provisions, which are analogous to the backfitting provisions in § 50.109, appear in § 52.63, “Finality of Standard Design Certifications”; § 52.83, “Finality of Referenced NRC Approvals; Partial Initial Decision on Site Suitability”; § 52.98, “Finality of Combined Licenses; Information Requests”; § 52.145, “Finality of Standard Design Approvals, Information Requests”; and § 52.171, “Finality of Manufacturing Licenses; Information Requests.” The backfitting and issue finality considerations, as applied to these entities and regulatory approvals, are considered below.

The alternatives presented would not constitute backfitting under § 50.109 or violate any issue finality provision in 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants.” Alternative 1 would maintain the status quo of the requirements for a reactor vessel material surveillance program under Appendix H to 10 CFR Part 50, thereby imposing no change in requirements or NRC positions. Alternative 2 would (1) provide licensees with a nonmandatory relaxation from the current 1-year period following a capsule withdrawal to 18-months to submit surveillance capsule test results, and (2) reduce testing requirements by amending the NRC’s regulations.

Because this change is not mandatory, the NRC would not require licensees to comply with the regulations that eliminate or reduce testing requirements for specified surveillance capsule specimens or that extend the allowable period for submitting surveillance test results (i.e., licensees can continue to submit surveillance capsule test results one year following a capsule withdrawal); thus, the rule would not constitute backfitting or violate issue finality.

7.2 Regulatory Flexibility Analysis

The Regulatory Flexibility Act, enacted in September 1980, requires agencies to consider the effect of regulatory changes on small entities, analyze alternatives that minimize effects on small entities, and make their analyses available for public comment.

This rule primarily affects the utilities that own light-water nuclear power reactors, and the vendors of those reactors, none of which meet the definition of "small entities" set forth in the size standards established by the NRC in § 2.810, "NRC Size Standards." Therefore, this rule would not have a significant economic effect on a substantial number of small entities.

7.3 Safety Goal Evaluation

Safety goal evaluations are applicable to regulatory initiatives that are generic safety enhancement backfits subject to the substantial additional protection standard in § 50.109(a)(3). This regulatory basis describes potential regulatory changes that would not qualify as generic safety enhancement backfits because the changes under consideration would be as follows:

- Revise requirements to eliminate and reduce the need to test certain surveillance capsule specimens.
- Extend the required submittal period of 1 year to 18 months for reporting surveillance test results.

Therefore, no safety goal evaluation is needed because the potential revisions do not affect one's ability to monitor changes in the fracture toughness properties of the reactor vessel materials and to analyze the integrity of the reactor vessel. These material surveillance programs would continue to be effective at predicting, in advance, the changes in reactor vessel material properties resulting from the cumulative effects of radiation.

7.4 Disaggregation

The NRC performed a screening review to determine whether any of the individual requirements (or set of integrated requirements) of the direct final rule would be unnecessary to achieve the objective of the rulemaking. The objective of this rulemaking is to reduce the regulatory burden on reactor licensees and the NRC that is associated with test specimens contained within surveillance capsules and the reporting of surveillance test results, with no impact on public health and safety. Each change to the regulatory language is in support of this objective. Therefore, the NRC concludes that each of the requirements in the direct final rule would be necessary to achieve the objective of the rulemaking.

8.0 Implementation

The NRC estimates that the direct final rule would be effective in 2020. The NRC assumes that the industry would defer any scheduled capsule withdrawals or tests in 2020 to avoid the cost for these activities.

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APPENDIX
MAJOR ASSUMPTIONS AND INPUT DATA

Description	Mean Estimate	Distribution	Low Estimate	Best Estimate	High Estimate	Source or Basis of Estimate
General Input						
Analysis Base Year	2020			*2020*		NRC assumption
Year Rule Is Effective	2020			2020		The effective year of the direct final rule is 4 years in the future, and averted costs can be credited in 2020.
Alternative A Discount Factor	3%			3%		NRC assumption, OMB guidance
Alternative B Discount Factor	7%			7%		NRC assumption, OMB guidance
NRC Hourly Staff Rate	$129			$129		NRC calculation
HAZ Specimen Testing Averted						
Licensee HAZ Specimen Testing CURRENT PROGRAM						
Cost per HAZ Test—$7,500 ($2017)	$8,046	PERT	$5,364	$8,046	$10,728	Industry input
NRC HAZ Specimen Testing CURRENT PROGRAM						
Hours to Review HAZ Specimen Testing Results	2	PERT	1	2	4	NRC estimate
Optional Tensile Testing						
Licensee Tensile Testing CURRENT PROGRAM						
Cost per Tensile Test—$7,500 ($2017)	$8,046			$8,046		NRC estimate
Residual Percentage of Tensile Testing Cost	33%	PERT	32%	33%	35%	NRC estimate; rule eliminates one-third of the tensile specimens that need to be tested
NRC Tensile Testing CURRENT PROGRAM						
Hours to Review Tensile Testing Results	2	PERT	1	2	4	NRC estimate
Correlation Monitors Testing Averted (Current Program)						
Licensee Correlation Monitor Specimen Testing						
Percentage of Capsules Scheduled for Testing that Include Correlation Monitor Material	40%	PERT	38%	40%	42%	NRC assumption that 40% of remaining capsules contain correlation monitor material
Cost per Correlation Monitor Test—$7,500 ($2017)	$8,046			$8,046		Industry input
NRC Correlation Monitor Specimen Testing						
Hours to Review Correlation Monitor Testing Results	2	PERT	1	2	4	NRC estimate

Description	Mean Estimate	Distribution	Low Estimate	Best Estimate	High Estimate	Source or Basis of Estimate
Thermal Monitor Specimen Examination						
Licensee Thermal Monitor Specimen Examination						
Cost per Thermal Monitor Test—$7,500 ($2017)	$8,046			$8,046		NRC estimate
Residual Percentage of Thermal Monitor Testing Cost	33%	PERT	32%	33%	35%	NRC estimate; rule eliminates one-third of the thermal monitor specimens that need to be tested
NRC Thermal Monitor Examination						
Hours to Review Thermal Monitor Specimen Results	2	PERT	1	2	4	NRC estimate; rule eliminates one-third of the tensile specimens that needed to be tested
Schedule Extension Request Letter to Extend Surveillance Report Submission from 12 to 18 Months						
Licensee Preparation of Extension Request						
Cost to Prepare and Submit Extension Request	$21,457			$21,457		Industry input
NRC review and respond to extension request submission						
NRC Review Licensee Reports Requesting Extension (Hours)	60	PERT	40	60	100	NRC estimate

PERT=program evaluation and review technique

SUBJECT: Regulatory Analysis for the Direct Final Rule: Appendix H to Part 50—Reactor Vessel Material Surveillance Program Requirements. **DATED:**

ADAMS Accession No.: ML19184A625 *concurrence by e-mail

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NAME	CBladey	SSoto-Lugo	JTappert	JDonoghue
DATE	8/15/2019	8/23/2019	9/18/2019	9/30/2019
OFFICE	RES: D*	OGC (NLO)*	NRR: D*	EDO
NAME	RFurstenau	MSpencer (JGillespie for)	HNieh (RTaylor for)	MDoane
DATE	10/18/2019	3/26/2020	4/16/2020	5/5/20

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