## PRELIMINARY DRAFT

*Public availability of this draft document is intended to inform stakeholders of the current status of the NRC staff's preliminary draft final rule package and associated documents for § 50.46c of Title 10 of the Code of Federal Regulations (10 CFR). These preliminary draft documents are in support of an October 22, 2015, Category 3 public meeting, and a November 2, 2015, Advisory Committee on Reactor Safeguards (ACRS) subcommittee meeting.*

*This draft document has not been subject to all levels of NRC management review. Accordingly, it is incomplete and may be in error in one or more respects. The document may be subject to further revision before the staff provides the final draft rule language package to the Commission (currently scheduled to be provided to the Commission in February 2016). In particular, the preliminary draft language in paragraph (e),* ***Alternate risk-informed approach for addressing the effects of debris on long term core cooling****, is under current staff discussion with respect to the manner in which the risk-informed alternative may be used in the initial NRC approval of new reactor designs, and for modifications of both new reactor designs and currently operating nuclear power reactors.*

**[7590-01-P]**

**NUCLEAR REGULATORY COMMISSION**

**10 CFR Parts 50 and 52**

**RIN 3150-AH42**

**[NRC-2008-0332, NRC-2012-0041, NRC-2012-0042, NRC-2012-0043, NRC-2015-0095]**

# Performance-Based Emergency Core Cooling System Requirements And Related Fuel Cladding Acceptance Criteria

**AGENCY:** Nuclear Regulatory Commission.

**ACTION:** Final rule.

**SUMMARY:** The U.S. Nuclear Regulatory Commission (NRC) is amending its regulations to revise the acceptance criteria for emergency core cooling systems (ECCS) for light-water

nuclear power reactors. The revised ECCS acceptance criteria are performance-based and reflect recent research findings that identified new embrittlement mechanisms for fuel rods with zirconium alloy cladding under loss-of-coolant accident (LOCA) conditions. The rule also addresses two petitions for rulemaking (PRMs) by establishing requirements applicable to all fuel types and cladding materials, and requiring the consideration of crud, oxide deposits, and hydrogen content in zirconium-based alloy fuel cladding. Further, the rule contains a provision that allows licensees to use an alternative risk-informed approach to evaluate the effects of debris for long-term cooling. The NRC is publishing four regulatory guides that support the implementation of the final rule.

**DATES:** *Effective Date*: *This* final rule is effective **[INSERT DATE 30 DAYS AFTER THE DATE OF PUBLICATION].**

**ADDRESSES:** Please refer to Docket ID NRC-2008-0332 when contacting the NRC about the availability of information for this final rule. You may obtain publicly-available information related to this final rule by any of the following methods:

- **Federal Rulemaking Web Site:** Go to http://www.regulations.gov and search for Docket ID NRC-2008-0332. Address questions about NRC dockets to Carol Gallagher; telephone: 301-415-3463; e-mail: Carol.Gallagher@nrc.gov. For technical questions, contact the individuals listed in the FOR FURTHER INFORMATION CONTACT section of this document.

- **NRC's Agencywide Documents Access and Management System (ADAMS):** You may obtain publicly-available documents online in the ADAMS Public Documents collection at http://www.nrc.gov/reading-rm/adams.html. To begin the search, select "ADAMS Public Documents" and then select "Begin Web-based ADAMS Search." For problems with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to pdr.resource@nrc.gov. For the convenience of the reader, instructions about obtaining materials referenced in this document are provided in Section XIX, "Availability of Documents," of this document.
- **NRC's PDR:** You may examine and purchase copies of public documents at the NRC's PDR, Room O1-F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852.

**FOR FURTHER INFORMATION CONTACT:** Alysia G. Bone, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, telephone: 301-415-1034, e-mail: Alysia.Bone@nrc.gov; or Paul M. Clifford, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, telephone: 301-415-4043, e-mail: Paul.Clifford@nrc.gov .

**SUPPLEMENTARY INFORMATION:**

**EXECUTIVE SUMMARY:**

*Purpose of the Regulatory Action.*

The final rule adopts performance-based regulatory requirements for determining the acceptability of an ECCS for a nuclear power reactor, including requirements governing the acceptability of the fuel rod cladding performance. (Fuel cladding performance affects the cooling requirements for the ECCS.) The final rule expands the applicability of the rule from uranium oxide pellets within cylindrical zircaloy or ZIRLO™ [1] cladding to any light-water reactor (LWR), regardless of fuel design or cladding material. The final rule also replaces prescriptive requirements with performance-based requirements. The requirements of the performance-based rule also address new technical information on fuel cladding integrity and degradation mechanisms. Performance-based ECCS requirements provide more flexibility for applicants and licensees to meet U.S. Nuclear Regulatory Commission (NRC) requirements for ECCS in a manner that provides reasonable assurance of adequate protection consistent with the requirements of the Atomic Energy Act of 1954, as amended.

The final rule also addresses two petitions for rulemaking (PRMs), PRM-50-71 and PRM-50-84. The PRM-50-71 requests that the NRC expand the applicability of the ECCS rule beyond zircaloy and ZIRLO™ cladding materials. The PRM-50-84 requests, among other items, that the NRC require licensees to consider the thermal effects of crud and oxide layers.

Finally, the final rule allows individual nuclear power plant licensees to resolve issues similar to GSI-191, “Assessment of Debris Accumulation on PWR [Pressurized Water Reactor] Sump Performance,” by using a risk-informed approach for evaluating the effects of debris on long-term cooling. The scope of the rule addresses this issue for both Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs).

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1 ZIRLO is a registered trademark of Westinghouse Electric Company LLC.

*Summary of the Major Provisions of the Final Rule.*

The final rule includes several significant changes to the NRC's existing[2] requirements on the ECCS:

- The final rule replaces prescriptive analytical requirements with performance-based requirements. To demonstrate compliance with the requirements, ECCS performance will be evaluated using fuel-specific performance objectives and associated analytical limits that take into consideration all known degradation mechanisms and unique performance features of the particular fuel system, along with an acceptable ECCS evaluation model.
- The final rule applies to all fuel designs and cladding materials. The final rule defines two principal ECCS performance requirements:
    - Core temperature during and following the postulated LOCA does not exceed the analytical limits for the fuel design used for ensuring acceptable performance.
    - The ECCS provides sufficient coolant so that decay heat will be removed for the extended period of time required by the long-lived radioactivity remaining in the core.
- The final rule also includes specific performance requirements for fuel designs consisting of uranium oxide or mixed uranium-plutonium oxide fuel pellets within cylindrical zirconium-alloy cladding. New performance objectives and analytical limits may be necessary for other fuel designs, as they are developed. These changes address the requests of PRM-50-71.

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2 In this *Federal Register* (FR) notice, "existing" requirements indicate 10 CFR 50.46c requirements that were last revised in 72 FR 49494 on August 28, 2007.

- The final rule incorporates the results of relatively recent research findings. The existing requirement to maintain the calculated total cladding oxidation below 17 percent is replaced with a requirement to establish analytical limits on peak cladding temperature (PCT) and integral time at temperature (ITT) that correspond to the measured ductile-to-brittle transition for the zirconium-alloy cladding material. The final rule also addresses a newly identified phenomenon known as breakaway oxidation by requiring measurement of breakaway oxidation behavior for each cladding allow, and periodic confirmation of those measurements. The final rule will require evaluation against measured breakaway oxidation behavior to demonstrate that ECCS performance precludes breakaway oxidation. Additionally, the final rule requires licensees to consider the effects of oxygen diffusion from the cladding inside surfaces, if an oxygen source is present on the cladding inner surfaces at the onset of the LOCA.
- The final rule requires that licensees evaluate the thermal effects of crud and oxide layers that accumulate on the fuel cladding during plant operation. Crud is defined as any foreign substance deposited on the surface of the fuel cladding prior to initiation of a LOCA. This addition addresses a request of PRM-50-84.
- The final rule requires that calculated core temperature be maintained to prevent further cladding failure during the long-term cooling period of the accident.
- The final rule contains a provision that allows licensees to use an alternative risk-informed approach to evaluate the effects of debris for long-term cooling. The final rule contains acceptance criteria that apply to the risk-informed approach and its required content. Additionally, the final rule adds reporting requirements that pertain to the risk-informed approach.

*Costs and Benefits.*

The final rule, by requiring applicants and licensees to address new technical matters not currently required to be addressed by the NRC's existing ECCS requirements, provides adequate protection to the health and safety of the public by maintaining the level of protection that the NRC previously thought would be achieved by the existing rule. The NRC prepared a regulatory analysis for this final rule (ADAMS Accession No. XXXXXX) to identify the benefits and costs of the particular regulatory approach for addressing ECCS performance. The NRC notes that adequate protection must be assured without regard to cost, but if there is more than one way of achieving that level of protection, then costs may be considered. The regulatory analysis prepared for this rulemaking was used to help the NRC identify the most effective way of achieving reasonable assurance of adequate protection with respect to protection against LOCAs.

*<Placeholder for summary of regulatory analysis>*

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## I. Background.

*Risk-Informed Regulation.*

In SECY-98-300, "Options for Risk-Informed Revisions to 10 CFR Part 50-'Domestic Licensing of Production and Utilization Facilities,'" dated December 23, 1998 (Agencywide Document Access and Management System (ADAMS) Accession No. ML992870048), the U.S. Nuclear Regulatory Commission (NRC) began to explore approaches to risk-informing its regulations for nuclear power reactors. One alternative (termed "Option 3") involved making risk-informed changes to the specific requirements in the body of Title 10 of the Code of Federal Regulations (10 CFR) part 50. As the NRC began to develop its approach to risk-informing these requirements, the agency sought stakeholder input in public meetings. Two of the regulations identified by industry as potentially benefitting from risk-informed changes were §§ 50.44 and 50.46. Section 50.44 specifies the requirements for combustible gas control inside reactor containment structures, and § 50.46 specifies the requirements for light-water power reactor emergency core cooling systems (ECCS). For § 50.46, the potential was identified for making risk-informed changes to requirements for both ECCS cooling performance and ECCS analysis acceptance criteria in § 50.46(b).

*PRM-50-71.*

On March 14, 2000, as amended on April 12, 2000, the Nuclear Energy Institute (NEI) submitted a petition for rulemaking (PRM) (ADAMS Accession No. ML003723791) requesting that the NRC amend its regulations in §§ 50.44 and 50.46 (PRM-50-71). The NEI petition noted that these two regulations apply to only two specific zirconium-alloy fuel cladding materials (zircaloy and ZIRLO™). The NEI petition stated that reactor fuel vendors had subsequently developed new cladding materials other than zircaloy and ZIRLO™ and that in order for licensees to use these new materials under the regulations, licensees needed to request exemptions from §§ 50.44 and 50.46.

On May 31, 2000, the NRC published a notice of receipt in the *Federal Register* (65 FR 34599) and requested public comment. The public comment period ended on August 14, 2000, and the NRC received 11 public comment letters from public citizens and the nuclear industry. Although the majority of the comments generally supported the requests of the PRM, one commenter suggested that the enhanced efficiency of the proposal would be at the expense of public health and safety. The NRC disagrees with that commenter and notes that, while the petition's proposal would remove specific zirconium-alloy names from the regulation, the NRC review and approval of specific zirconium-alloys for use as reactor fuel cladding would be required prior to their use in reactors (with the exception of lead test assemblies permitted in technical specifications). The NRC's detailed discussion of the public comments submitted on PRM-50-71, including a detailed list of commenters, is contained in a separate document, "Section 50.46c and PRM-50-71 Comment Response Document" (ADAMS Accession No. ML12283A213).

After evaluating the petition and public comments received, the NRC decided that PRM-50-71 should be considered in the rulemaking process. The NRC's determination was published in the *Federal Register* on November 6, 2008 (73 FR 66000). Because most of the issues raised in this PRM pertain to § 50.46, the PRM is addressed in this final rule.[3]

*Staff Requirements Memorandum Direction.*

On March 31, 2003, in response to SECY-02-0057, "Update to SECY-01-0133, 'Fourth Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.46 (ECCS Acceptance Criteria)'" (ADAMS Accession No. ML020660607), the Commission issued a staff requirements memorandum (SRM) (ADAMS Accession No. ML030910476) directing the NRC staff to move forward to risk-inform its regulations in a number of specific areas. In addition, this SRM directed the staff to modify the ECCS acceptance criteria to provide a more performance-based approach to the ECCS requirements in § 50.46.

*PRM-50-84.*

On March 15, 2007, Mr. Mark Leyse (the petitioner) submitted a PRM to the NRC (ADAMS Accession No. ML070871368) requesting that all holders of operating licenses for nuclear power plants be required to operate such plants at operating conditions (e.g., levels of power production and light-water coolant chemistries) necessary to effectively limit the thickness

[3] PRM-50-71 also requested changes to § 50.44. Those changes were addressed in a rulemaking that revised that section (68 FR 54123; September 16, 2003) to include risk-informed requirements for combustible gas control. That regulation was also modified to be applicable to all boiling or pressurized water reactors regardless of type of fuel cladding material used.

of crud[4] and/or oxide layers on fuel rod cladding surfaces. The petitioner requested that the NRC conduct rulemaking in the following three specific areas:

1) Establish regulations that require licensees to operate light-water power reactors under conditions that are effective in limiting the thickness of crud and/or oxide layers on zirconium-clad fuel in order to ensure compliance with § 50.46(b) ECCS acceptance criteria;

2) Amend Appendix K to 10 CFR part 50 to explicitly require that steady-state temperature distribution and stored energy in the reactor fuel at the onset of a postulated loss-of-coolant accident (LOCA) be calculated by factoring in the role that the thermal resistance of crud deposits and/or oxide layers plays in increasing the stored energy in the fuel (these requirements also be applied to any acceptable, best-estimate ECCS evaluation models used in lieu of Appendix K to 10 CFR part 50 calculations); and

3) Amend § 50.46 to specify a maximum allowable percentage of hydrogen content in (fuel rod) cladding.

On May 23, 2007, the NRC published a notice of receipt for this petition in the *Federal Register* (72 FR 28902) and requested public comment. The public comment period ended on August 6, 2007. Comments in support of PRM-50-84 were provided by the Union of Concerned Scientists, two individuals, and the petitioner. The NEI and Strategic Teaming and Resource Sharing organization submitted comments in opposition to the petition. After evaluating the public comments, the NRC resolved PRM-50-84 by deciding that each of the petitioner's issues should be considered in the rulemaking process. The NRC's determination, including the

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[4] For the purpose of this discussion, the NRC defines "crud" as any foreign substance deposited on the surface of the fuel cladding prior to the initiation of a LOCA. It is known that this layer can impede the transfer of heat.

NRC's response to public comments received on the petition, was published in the *Federal Register* on November 25, 2008 (73 FR 71564). Although there is no direct relationship between the subject of crud and the revised ECCS acceptance criteria requirements, the petition relates to the NRC's requirements on ECCS performance in § 50.46. Given the comprehensive changes to § 50.46 being addressed in this rulemaking, the NRC is addressing the petitioner's proposed changes in this rulemaking.

**A. Emergency Core Cooling System: Embrittlement Research Findings.**

Separate from the effort to modify the regulations to provide a more risk-informed, performance-based regulatory approach, the NRC had also undertaken a fuel cladding research program to investigate the behavior of high-exposure fuel cladding under accident conditions. This research program included an extensive LOCA research and testing program at Argonne National Laboratory (ANL), as well as jointly-funded programs at the Kurchatov Institute (supported by the French Institute for Radiological Protection and Nuclear Safety and the NRC) and the Halden Reactor project (a jointly-funded program under the auspices of the Organization for Economic Cooperative Development – Nuclear Energy Agency, sponsored by national organizations in 18 countries). The effects of both alloy composition and fuel burnup (the extent to which fuel is used in a reactor) on cladding embrittlement (e.g., loss of ductility) under accident conditions were studied in these research programs. The research programs identified new cladding embrittlement mechanisms and expanded the NRC's knowledge of previously identified mechanisms. Major research findings are summarized as follows.

Hydrogen-Enhanced Beta-Layer Embrittlement:

In the existing regulations, the preservation of cladding ductility, via compliance with regulatory criteria on peak cladding temperature (§ 50.46(b)(1)) and local cladding oxidation (§ 50.46(b)(2)), provides a level of assurance that fuel cladding will not experience gross failure and that the fuel rods will remain within their coolable lattice arrays. The recent LOCA research program identified new cladding embrittlement mechanisms that demonstrated that the current combination of peak cladding temperature (2200 degrees Fahrenheit (°F) (1204 degrees Celsius (°C))) and local cladding oxidation (17 percent equivalent cladding reacted (ECR)) criteria may not always ensure post-quench ductility (PQD). As explained in Section 1.4 of NUREG/CR-6967, oxygen diffusion into the base metal under LOCA conditions promotes a reduction in the size (referred to as beta-layer thinning) and ductility (referred to as beta-layer embrittlement) of the metallurgical structure within the cladding that provides its macroscopic mechanical behavior. The presence of hydrogen within the cladding accelerates this embrittlement process.

The NRC's LOCA research program did not investigate cladding degradation mechanisms or develop the technical basis for performance-based requirements beyond the existing 2200 °F peak cladding temperature criterion. Examples of degradation mechanisms beyond cladding embrittlement (via oxygen diffusion) include excessive exothermic metal-water reaction, alloy-specific eutectics, and loss of fuel rod geometry due to plastic deformation. As a result, the existing 2200 °F limit (specified in § 50.46c(g)(1)(i) of the final rule) remains an absolute upper limit for zirconium-based alloys on peak cladding temperature (PCT). However, as reflected in this embrittlement requirement (§ 50.46c(g)(1)(ii) of the final rule) and the results of the fuel cladding research program, a lower PCT may be required to preserve ductility.

Oxygen Ingress From Cladding Inside Diameter:

As explained in Section 1.4.6 of NUREG/CR-6967, oxygen sources may be present on the inner surface of irradiated cladding due to gas-phase $UO_3$ transport prior to gap closure, fuel-cladding-bond formation (uranium dioxide in solid solution with zirconium dioxide), and the fuel bonded to this layer. Under LOCA conditions, this available oxygen may diffuse into the base metal of the cladding, effectively reducing the integral time-at-temperature to nil ductility.

Breakaway Oxidation:

As explained in Section 1.4.5 of NUREG/CR-6967, zirconium dioxide can exist in several crystallographic forms (allotropes). The normal tetragonal oxide that develops under LOCA conditions is dense, adherent, and protective with respect to hydrogen pickup. However, there are conditions that promote a transformation to the monoclinic phase (i.e., the phase that is grown during normal operation), which is neither fully dense nor protective. The tetragonal-to-monoclinic transformation is an instability that initiates at local regions of the metal-oxide interface and grows rapidly throughout the oxide layer. Because this transformation results in an increase in oxidation rate, it is referred to as breakaway oxidation. Along with this increase in oxidation rate resulting from cracks in the monoclinic oxide, significant hydrogen pickup also occurs. Hydrogen that enters in this manner during a LOCA transient promotes rapid embrittlement of the cladding.

While all zirconium alloys will eventually experience breakaway oxidation when exposed to long enough durations of high-temperature steam oxidation, the fuel cladding research program demonstrated that alloying composition and manufacturing process (e.g., surface roughness) influence the timing of this phenomenon.

Applicability of Ductility-Based Analytical Limits to Burst Region:

During a postulated LOCA, a portion of the fuel rod population may be predicted to experience fuel rod ballooning and cladding rupture as a result of rapid depressurization of the reactor coolant system in combination with elevated cladding temperature. The number of burst rods depends on several variables including initial conditions (e.g., fuel rod design, rod internal pressure, rod power) and accident conditions (e.g., LOCA break size, cladding temperature). This flawed section of the fuel rod may experience degradation mechanisms beyond oxygen diffusion embrittlement encountered in the remaining portions of the fuel rod, including significant amounts of hydrogen uptake from steam entering the fuel rod through the rupture.

To investigate the mechanical behavior of ruptured fuel rods, the NRC conducted integral LOCA testing, designed to exhibit ballooning and burst, on as-fabricated and hydrogen-charged cladding specimens and high-burnup fuel rod segments exposed to high-temperature steam oxidation followed by quench. The research results and conclusions are documented in the NUREG-2119, "Mechanical Behavior of Ballooned and Ruptured Cladding" (ADAMS Accession No. ML12048A475). The integral LOCA testing confirms that continued exposure to a high-temperature steam environment weakens the already flawed region of the fuel rod surrounding the cladding rupture. Hence, limitations on PCT and integral time-at-temperature are necessary to preserve an acceptable amount of mechanical strength and fracture toughness to maintain a coolable fuel geometry. In addition, the research demonstrated that the degradation in strength and fracture toughness with prolonged exposure to steam oxidation was enhanced with pre-existing cladding hydrogen content.

The research findings from the integral LOCA research presented the NRC with two options for revising the fuel performance requirements: 1) establish a separate performance requirement within the burst region (i.e., analytical limits that preserve sufficient fracture toughness to ensure burst region survival): or 2) apply the hydrogen-based embrittlement analytical limits to the entire fuel rod.

In the absence of a credible analysis of loads, cladding stresses, and cladding strains for a core degraded by LOCA conditions, there are no absolute metrics to determine how much ductility or strength would be needed to provide absolute assurance that fuel-rod cladding would maintain its geometry during and following post-LOCA quench. It is also not clear what impact severance of some fuel rods into two pieces, owing to potential loads following a hypothetical LOCA, would have on core coolability. Fragmentation of fuel rod cladding would be more detrimental to core coolability than severance of rods into two pieces. Even minimal ductility ensures that cladding will have high strength and toughness and therefore, high resistance to fracturing. Brittle cladding, on the other hand, might fail at low strength and shatter. Therefore, the intent to maintain ductility is beneficial even without adequate knowledge of LOCA loads. The research documented in NUREG-2119 showed that if wall thinning and double-sided oxidation are accounted for, then hydrogen-based embrittlement limits, such as the limit provided in Figure 2 of Regulatory Guide (RG) 1.224, are sufficient to ensure reasonable behavior of the ballooned and ruptured region.

Therefore, the NRC elected the second regulatory approach to apply a single performance-based requirement to the entire fuel rod. This decision recognizes that portions of the cladding within the burst region may not maintain ductility. This position is reflected in RG 1.224 and supported by the technical basis documented in NUREG-2119.

These research findings have been summarized in Research Information Letter (RIL)-0801, “Technical Basis for Revision of Embrittlement Criteria in 10 CFR 50.46” (ADAMS Accession No. ML081350225), and the detailed experimental results from the program at ANL are contained in NUREG/CR-6967, “Cladding Embrittlement during Postulated Loss-of-Coolant Accidents” (ADAMS Accession No. ML082130389). Since the publication of NUREG/CR-6967 and RIL-0801, additional testing was conducted related to the embrittlement phenomenon, which has been documented in supplemental reports. Where the additional testing relates to conclusions and recommendations in RIL-0801, RIL-0801 has been supplemented to reference the additional reports and incorporate findings (“Update to Research Information on Cladding Embrittlement Criteria in 10 CFR 50.46,” dated December 29, 2011 (ADAMS Accession No. ML113050484)).

The NRC publicly released the technical basis information in RIL-0801 on May 30, 2008, and NUREG/CR-6967 on July 31, 2008. Also on July 31, 2008, the NRC published in the *Federal Register* a notice of availability of the RIL and NUREG/CR-6967, together with a request for comments (73 FR 44778). Further discussion is provided in Section IV of this document, “Opportunities for Public Participation.”

**B. Generic Safety Issue (GSI)-191 and Long-Term Cooling.**

As a result of evolving staff concerns related to the adequacy of pressurized power reactor (PWR) recirculation sump designs, the NRC opened Unresolved Safety Issue (USI) A-43, "Containment Emergency Sump Performance." The resolution of USI A-43 was subsequently documented in Generic Letter (GL) 1985-022, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," dated December 3, 1985 (ADAMS Accession No. ML031150731). The NRC staff found in GL 1985-022 that the 50 percent blockage assumption, identified in RG 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems," Revision 0 (ADAMS Accession No. ML111680318), should be replaced with a more comprehensive requirement to assess debris effects on a plant-specific basis. Following the resolution of USI A-43, industry events at Barsebeck and Limerick Generating Station challenged the conclusion that no new requirements were necessary to prevent the clogging of ECCS strainers at operating BWRs.

As described in NRC Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode," dated October 7, 1995 (ADAMS Accession No. ML082490807), a safety relief valve at the Limerick Generating Station inadvertently opened and could not be closed, the plant was manually scrammed, and the residual heal removal (RHR) system was started in the suppression pool cooling mode to remove the heat added by the open relief valve. The A train of the RHR exhibited signs of pump cavitation and was secured. The B train of the RHR was then started to remove the heat from the relief valve discharge. After the plant was stabilized, a diver inspected the pump suction strainers and found a mat of fibers and sludge covering them. The licensee determined that the discharge from the relief valve did not contribute debris to the suppression pool.

As described in NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling

Suction Strainers by Debris in Boiling-Water Reactors," dated May 6, 1996 (ADAMS Accession No. ML082401219), a Swedish BWR, Barseback Unit 2, experienced plugging of two containment vessel spray system (CVSS) suction strainers. The strainers were partially plugged with mineral wool (a fibrous insulation) that was dislodged by a steam jet from an open pilot operated relief valve. The operators noticed an indication of high-differential pressure across the strainers and were able to back flush them to keep the CVSS operating.

Also described in NRC Bulletin 96-03 are two ECCS suction strainer plugging events that occurred at the Perry Nuclear Power Plant, a BWR located in the United States. The first event resulted from general maintenance material and dirt in the suppression pool collecting on the RHR suction strainers. The differential pressure caused by the debris resulted in deformation of the suction strainers. After the suppression pool was cleaned and the suction strainers replaced, a second event occurred when several safety relief valves lifted. The RHR system was used to cool the suppression pool after the steam discharge. The suction strainers were inspected and found to be covered with fibrous debris and corrosion products. A test of the system found that the B train pump suction pressure dropped to zero. The fibrous debris originated from temporary drywell cooling filter media that was accidentally dropped into the suppression pool and not retrieved. The fibers created a filtering bed on which particles collected, resulting in a high-resistance debris bed.

In response to these events, the NRC issued generic communications requesting that BWR licensees take appropriate actions to minimize the potential for the clogging of ECCS suction strainers by debris accumulation following a LOCA. The NRC staff concluded that all BWR licensees have sufficiently addressed these bulletins in a memorandum, "Completion of

Staff Reviews of NRC Bulletin 96-03, 'Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors,' and NRC Bulletin 95-02, 'Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode',” dated October 18, 2001 (ADAMS Accession No. ML012970229).

The findings regarding BWR strainers prompted the NRC to open GSI-191, “Assessment of Debris Accumulation on PWR Sump Performance,” to ensure that post-accident debris effects would not impede long-term core cooling at PWRs. After completing its technical assessment of GSI-191, the NRC issued Bulletin 2003-01, “Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors,” dated June 9, 2003 (ADAMS Accession No. ML031600259). This bulletin did not require licensees to immediately perform deterministic evaluations for debris effects, but it requested that plants take compensatory measures to reduce risk or otherwise enhance the capability of the ECCS and containment spray system (CSS) recirculation functions. The bulletin also informed licensees that the staff was preparing a GL that would request that plants demonstrate through deterministic methods that long-term core cooling would not be compromised by debris effects.

Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors,” dated September 13, 2004 (ADAMS Accession No. ML042360586), was issued to all operating PWRs requesting that they perform a mechanistic evaluation of the effects of debris on the ECCS and CSS recirculation functions. The affected plants are currently working to address the issues identified by the GL. All operating PWRs have installed larger strainers and taken other actions toward the final resolution of the issue. Final closure of the generic letter has been delayed to allow industry and the NRC staff to develop appropriate methodologies for evaluation of debris related issues that were identified after the issuance of the GL. The staff generated two SECY papers on this issue to provide options and solicit feedback from the NRC Commissioners. On December 14, 2012, the Commission issued an SRM (ADAMS Accession No. ML12349A378) for SECY-12-0093, "Closure Options for Generic Safety Issue – 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance" (ADAMS Accession No. ML121320270). In this SRM, the Commission directed the following:

> The forthcoming § 50.46c proposed rulemaking should contain a provision allowing NRC licensees on a case-by-case basis, to use risk-informed alternatives. The license amendment process would be used to reconstitute the long-term core cooling licensing basis. Stakeholder comments should be solicited on the proposed provision.

Consistent with this SRM, the final rule includes a provision that allows licensees to use an alternate risk-informed approach for addressing the effects of debris on long term core cooling. The risk-informed guidance was developed using lessons learned from a PWR plant pilot application for implementing a risk-informed alternative to evaluate long-term core cooling.

## II. Operating Plant Safety.

### A. Cladding Embrittlement.

In response to the research findings in RIL-0801, the NRC performed a preliminary safety assessment of currently operating reactors ("Plant Safety Assessment of RIL-0801 (non-proprietary)," dated February 23, 2009 (ADAMS Accession No. ML090340073)). This assessment found that, due to realistic fuel rod power history, calculated cladding performance under LOCA conditions, and current analytical conservatisms, sufficient safety margin relative to the research findings exists for operating reactors. Therefore, the NRC staff determined that immediate regulatory action was not required and that changes to the ECCS acceptance criteria to account for these new findings could reasonably be addressed through the rulemaking process.

Recognizing that finalization and implementation of the new ECCS requirements would take several years, the NRC decided that a more detailed safety assessment was necessary. As a voluntary industry effort, the PWR Owners Group (OG) ("Letter Report: OG-11-143 PWROG 50.46(b) Margin Assessment," dated April 29, 2011 (ADAMS Accession No. ML11139A309)) and BWROG ("BWROG-TP-11-010 (Rev. 1) Evaluation of BWR LOCA Analyses and Margins Against High Burnup Fuel Research Findings," dated June 2011 (ADAMS Accession No. ML111950139)), under the auspices of NEI, submitted ECCS margin assessment reports. After grouping plants based on similar design features, cladding alloys, or ECCS evaluation models and defining cladding alloy-specific analytical limits, the OG reports identified analytical credits or performed new LOCA analyses necessary to demonstrate that the limiting plant within each grouping had positive margin relative to the research findings. The NRC conducted an audit of the OG reports and supporting General Electric - Hitachi (GEH), AREVA, and Westinghouse engineering calculations. Based on the OG reports and supplemental information collected during the audits, the NRC was able to confirm and document , for every operating reactor, current safe operation. In other words, in the unlikely event that an actual LOCA had occurred at any operating reactor, there is a level of assurance that the ECCS would have performed in an acceptable manner (relative to the new requirements) and a coolable core geometry would have been maintained. This conclusion is partly based on analyses that may not contain the level of conservatism or precision inherent in currently approved models and methods.

As documented in the audit report and safety assessment ("ECCS Performance Safety Assessment and Audit Report," dated February 10, 2012 (ADAMS Accession No. ML12041A078)), the NRC intends to verify, on an annual basis, continued safe operation until each licensee has implemented the new ECCS requirements. Updates to the ECCS safety assessment for 2013, 2014, and 2015 are available in ADAMS Accession Nos. ML14022A161, ML14358A493, and MLXXXXXX respectively.

While the updated safety assessment provides a level of assurance that no imminent safety concern exists for operating reactors, placing the burden of demonstrating plant safety on the staff, licensees should be required by regulation to demonstrate acceptable ECCS performance.

**B. GSI-191 and Long-Term Core Cooling.**

This section includes information on action taken by the NRC and licensees to address the potential effects of debris on long-term cooling. These actions have contributed significantly to the safety of operating plants. The NRC staff provided information to the Commission in two SECY papers: SECY-10-0113, "Closure Options for Generic Safety Issue – 191, Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance," dated August 26, 2010 (ADAMS Accession No. ML101820296); and SECY-12-0093, "Closure Options for Generic Safety Issue – 191, Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance," dated July 9, 2012 (ADAMS Accession No. ML12130270).

The Commission issued guidance for the closure of the issue in two SRMs associated with each SECY paper. The SRM-SECY-10-0113 ("Staff Requirements – SECY-10-0113 – Closure Options for Generic Safety Issue – 191, Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance" (ADAMS Accession No. ML103570354)) was issued on December 23, 2010. With respect to operating plant safety the SRM stated:

> The staff should take the time needed to consider all options to a risk-informed, safety conscious resolution to GSI-191. While they have not fully resolved this issue, the measures taken thus far in response to the sump-clogging issue have contributed greatly to the safety of U.S. nuclear power plants. Given the vastly enlarged advanced strainers installed, compensatory measures already taken, and the low probability of challenging pipe breaks, adequate defense-in-depth is currently being maintained.

On December 14, 2012, the Commission issued the SRM-SECY-12-0093 (ADAMS Accession No. ML12349A378). With respect to operating plant safety, the SRM reiterated the direction in SRM-SECY-10-0113.

As directed by the Commission, the NRC staff is currently working with licensees to assure adequate safety by closing the issue and updating plant licensing bases to reflect full compliance on a schedule consistent with Commission direction.

## III. Discussion: Requirements for ECCS Performance during LOCAs.

The final rule establishes a general, performance-based rule governing ECCS performance for light-water reactor (LWRs), regardless of fuel design or cladding material. This represents a significant change from the existing ECCS regulations, which apply to "uranium oxide pellets within cylindrical zircaloy or ZIRLO$^{TM}$ cladding." Because ECCS requirements must be expressed independent of fuel type, and because ECCS performance ultimately must be based upon maintaining the fuel in the reactor in a safe (analyzed) condition, the final rule separates the ECCS requirements from the need for the applicant/licensee to establish the fuel system design performance criteria constituting a safe condition.

In the final § 50.46c, the specified performance objectives of the systems, structures, and components of the ECCS are to provide residual heat removal during and following a postulated LOCA. As with the existing regulations, the ECCS performance is demonstrated by acceptable ECCS evaluation models in the final § 50.46c. Specific performance requirements and analytical limits have been established for fuel designs consisting of uranium oxide or mixed uranium-plutonium oxide pellets within zirconium cladding alloys that account for relatively recent research findings. For other fuel designs, new performance objectives and analytical limits may be necessary. Such objectives and limits would need to take into consideration all degradation mechanisms and any unique performance features of the particular fuel system.

The final rule follows the general regulatory approach of the existing regulations, yet it establishes non-prescriptive, performance-based regulatory language for demonstrating acceptable ECCS performance and determining fuel performance characteristics. The organization and 10 CFR designations of the NRC's requirements governing emergency core cooling (currently in § 50.46) and reactor cooling venting systems (currently in § 50.46a) will change in two steps. . A detailed description of the transition of 10 CFR designations is provided in Section VI, "Section-by-Section Analysis," of this document.

**A. Applicability.**

The NRC is expanding the applicability of the rule from "uranium oxide pellets within cylindrical zircaloy or ZIRLO™ cladding" to any LWR, regardless of fuel design or cladding material. The final rule is applicable to the following entities: applicants for and holders of construction permits, operating licenses, combined licenses, standard design approvals and to applicants for certified designs (design certification rules) and for manufacturing licenses. The rule does not apply to any licensee that has submitted certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel, in accordance with § 50.82(a)(1) or § 52.11(a)(1).

Over the past 10 years, the NRC has granted exemptions from the requirements of § 50.46 (in accordance with § 50.12(a)) to licensees utilizing approved fuel designs with M5®[5] zirconium-based alloy cladding and, more recently, to licensees using approved fuel designs with Optimized ZIRLO™ zirconium-based alloy cladding. The final rule includes general performance requirements for future LWR fuel designs and specific performance requirements for the current generation of LWR fuel designs with zirconium-based alloy claddings. As such, it is anticipated that future exemption requests would not be necessary for loading an advanced fuel design or cladding material approved by the NRC. However, the licensee would still need to submit a license amendment request. During this approval process the NRC would determine whether, either: 1) specified and NRC-approved analytical limits have been established, along with an acceptable ECCS evaluation model, which satisfy the specific performance-based requirements for fuel designs consisting of uranium oxide or mixed uranium-plutonium oxide pellets within zirconium-based alloy cladding material or 2) specified

5 M5 is a registered trademark of AREVA.

performance objectives and associated analytical limits which take into consideration all degradation mechanisms and any unique performance features of the particular fuel system have been established, along with an acceptable ECCS evaluation model, by which to judge the ECCS performance for new fuel designs.

The NRC recognizes that a small number of fuel rods may experience cladding failure (e.g., small perforation) during normal operation due to manufacturing defects, debris fretting, grid-to-rod fretting, etc. The allowable number of fuel rod failures during normal operation is not governed by ECCS performance requirements, but limited by 10 CFR part 20, "Standards for Protection against Radiation," and plant Technical Specifications, which limit reactor coolant activity level to maintain on-site and off-site dose during normal operation, anticipated operational occurrences, and postulated accidents to within prescribed limits. In addition to Technical Specifications limitations, plant administrative limits on reactor coolant activity level further reduce the potential number of failed fuel rods within an operating core.

Due to secondary degradation effects, the performance of these limited failed fuel rods during a postulated LOCA may be difficult to predict and would most likely be outside the experimental database used to set the NRC-approved analytical limits for coolable geometry (i.e., cladding embrittlement for zirconium-based alloys). However, due to their limited number relative to the total core population, any unforeseen degradation or performance during a postulated LOCA would not challenge the general performance requirements. As such, compliance with ECCS performance requirements of § 50.46c is not required for this limited number of failed fuel rods.

This extension to all LWR fuel types addresses PRM-50-71, which requested that the applicable regulations be amended to allow for the introduction of advanced zirconium-based alloy claddings, thus eliminating the need for a licensee to pursue an exemption for alloys which did not meet the definition of "zircaloy or ZIRLO$^{TM}$." As a result of adopting this provision, PRM-50-71 is considered granted and resolved.

**B. ECCS Performance and Analytical Requirements.**

The systems, structures, and components of the ECCS are designed to provide residual heat removal during and following a postulated LOCA. Failure of the ECCS to perform its intended function would result in a loss of coolable geometry followed by core reconfiguration. While the principal ECCS performance requirements are simple in nature (i.e., remove residual heat and maintain core temperatures at acceptable levels), the system must be designed to achieve specified performance objectives, taking into consideration all degradation mechanisms and any unique performance features of the particular fuel system that the ECCS is intended to cool. Sufficient empirical data must be available for the particular fuel system to identify all degradation mechanisms (e.g., embrittlement, loss of structural integrity) and any unique performance features (e.g., eutectic or exothermic reactions, combustible gas generation) to specify both acceptable core temperatures and the duration for which the ECCS must remove residual heat. In addition, fuel-specific analytical requirements may be necessary to accurately or conservatively model unique phenomena that impact the ECCS performance demonstration (e.g., fuel rod balloon and burst, cladding inside-diameter oxygen ingress).

To achieve the NRC's goal of a more performance-based rule, significant changes in format and structure are being made relative to § 50.46. In place of the existing prescriptive

§ 50.46(b) analytical limits, the rule defines the following principal ECCS performance requirements:

- Core temperature during and following the LOCA event does not exceed the analytical limits for the fuel design used for ensuring acceptable performance. This ensures that the fuel maintains a coolable geometry.
- Sufficient cooling so that decay heat will be removed for the extended period of time required by the long-lived radioactivity remaining in the core so that long-term cooling is ensured.

Compliance with these performance requirements provides reasonable assurance that the overall objective of maintaining a coolable core geometry in the event of a LOCA is met. In addition, the rule dictates specific analytical requirements for demonstrating compliance with the ECCS performance requirements. For instance, to demonstrate compliance with these system performance requirements, ECCS performance will be evaluated using fuel-specific performance objectives and associated analytical limits that take into consideration all degradation mechanisms and unique performance features of the particular fuel system, along with an acceptable evaluation model.

**C. Fuel-Specific Performance and Analytical Requirements.**

The rule includes specific performance requirements for fuel designs consisting of uranium oxide or mixed uranium-plutonium oxide fuel pellets within cylindrical zirconium-alloy cladding. These performance requirements incorporate the findings of the NRC LOCA research program. New performance objectives and analytical limits may be necessary for other fuel designs.

For uranium oxide or mixed uranium-plutonium oxide fuel pellets within cylindrical zirconium-alloy cladding, all known degradation mechanisms and unique performance features have been identified, specific performance objectives have been defined, and fuel design-specific performance requirements have been established and included in the final rule. For this fuel system design, the performance objective is to maintain the rod bundle array in a coolable configuration. In other words, the objective is to maintain fuel pellets within the cladding, and fuel rods within the fuel bundle lattice. Existing ECCS evaluation models and methods are capable of accurately predicting core temperatures and demonstrating ECCS performance, provided this core configuration is maintained. To achieve this performance objective, the ECCS must limit core temperatures to prevent high-temperature cladding failure, prevent brittle cladding failure (i.e., maintain PQD and prevent breakaway oxidation), minimize hydrogen gas generation, and provide for long-term residual heat removal for the long-lived fission decay products associated with uranium oxide or uranium-plutonium oxide fuel.

The § 50.46(b) requirement concerning peak cladding temperature, specifically that the calculated maximum fuel element cladding temperature does not exceed 2200 °F, remains unchanged in the final § 50.46c.

The peak cladding temperature requirement currently in § 50.46(b)(1) is moved to § 50.46c(g)(1)(i). This requirement is unchanged from the proposed rule. In the proposed rule FRN, the NRC requested comment on the retention of a prescriptive peak cladding temperature criterion, rather than adoption of a performance-based requirement (Question 1). All comments received on Question 1 expressed views that maintaining the existing prescriptive limit was inconsistent with the objective of a performance-based rule. Many industry commenters suggested that the peak cladding temperature limit be moved to regulatory guidance, allowing applicants wishing to request PCT limits greater than 2200 °F to address relevant degradation mechanisms. NEI noted that at present, there are no industry plans to immediately seek a PCT acceptance criterion greater than 2200 °F. The NRC position is that there is not an adequate technical basis to extend peak cladding temperature beyond 2200 °F. Further, the NRC does not believe there a strong demand from licensees to outline a methodology to extend PCT beyond 2200 °F. The NRC did not change the rule due to these comments.

The § 50.46(b) requirement concerning maximum hydrogen generation remains unchanged in the final § 50.46c. Specifically, this provision requires that the calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

The maximum hydrogen generation limits currently in § 50.46(b)(3) are moved to § 50.46c(g)(1)(iv). This requirement is unchanged from the proposed rule.

The following requirements in the final § 50.46c rule represent changes from § 50.46(b):

<u>Post-Quench Ductility Performance Requirement</u>:

In the existing regulations, the preservation of cladding ductility, via compliance with regulatory criteria on PCT (§ 50.46(b)(1)) and local cladding oxidation (§ 50.46(b)(2)), provided a level of assurance that fuel cladding will not experience gross failure and that the fuel rods will remain within their coolable lattice arrays. The recent LOCA research program identified new cladding embrittlement mechanisms that demonstrated that the existing combination of PCT (2200 °F (1204 °C)) and local cladding oxidation (17 percent equivalent cladding reacted (ECR)) criteria may not always ensure PQD. To address this phenomenon (as well as to achieve a more performance-based rule), the NRC has replaced the existing prescriptive analytical limit on ECR with a performance-based requirement that requires licensees to establish specified and NRC approved analytical limits to preserve cladding PQD. These limits should correspond to the measured ductile-to-brittle transition for the zirconium based alloy cladding using an NRC approved experimental technique. This requirement is substantively unchanged from the proposed rule. There were few comments on this part of the proposed rule and no comments that the NRC accepted.

It is important to recognize that the embrittlement of the cladding is the result of oxygen diffusion into the base metal and not directly related to the rate of growth or overall thickness of a zirconium dioxide layer on the outside cladding diameter. In combination with a limit on peak cladding temperature, the existing regulation limits maximum local oxidation to preserve cladding ductility. Maximum local oxidation is used as a surrogate to limit the time-at-temperature and associated oxygen diffusion. This surrogate approach is possible because both the rate of oxidation and rate of oxygen diffusion share strong temperature dependence. In an attempt to more accurately characterize the degrading phenomenon, the final rule replaces the term maximum local oxidation with integral time-at-temperature, which more directly relates to the parameter of interest (i.e., embrittlement due to oxygen diffusion).

The LOCA research used to support the 1973 rulemaking employed the Baker-Just weight gain correlation to integrate time-at-temperature for the PQD experiments. To maintain consistency with the original experimental bases, 10 CFR part 50 appendix K required the use of Baker-Just to predict maximum local oxidation (i.e., integrate time-at-temperature). The recent NRC LOCA research program employed the Cathcart-Pawel weight gain correlation to integrate time-at-temperature for the PQD experiments. As such, future LOCA calculations using the generic analytical limits developed from these tests (i.e. Figure 2 of RG 1.224) must also use the Cathcart-Pawell correlation. This should clarify the need to have a consistent analytical technique to integrate time-at-temperature in both the empirical database (e.g., allowable CP-ECR) and evaluation model (e.g., predicted CP-ECR). Independent of the time-at-temperature integration correlation, the EM must employ an accurate or conservative weight gain correlation based on measured oxidation for estimating the rate of energy release and hydrogen generation from the metal/water reaction.

<u>Breakaway Oxidation Performance Requirement</u>:

Any fuel rod that experiences breakaway oxidation during a postulated LOCA will rapidly become brittle and more susceptible to gross failure and hence, is no longer in compliance with the General Design Criteria (GDC)-35 requirements for coolable core geometry. To address this phenomenon, the NRC has added a performance-based requirement that the entity establish an analytical time limit to preclude breakaway oxidation. This requirement, along with a periodic test requirement, will confirm that slight composition changes or manufacturing changes have not inadvertently altered the cladding's susceptibility to breakaway oxidation. The NRC is issuing RG 1.222, which will provide licensees with an acceptable experimental technique for conducting breakaway oxidation measurements as well as guidance regarding periodic test programs. The NRC is issuing RG 1.224, which includes guidance to establish an analytical limit to preclude breakaway oxidation. Licensees may also propose alternative approaches to those proposed in the regulatory guides.

The requirement to establish a breakaway oxidation analytical limit has not changed from the proposed rule; however, the testing and reporting requirements have changed. In the proposed rule, there was a requirement to report the results of periodic testing to the NRC on an annual basis. In the proposed rule FRN, the NRC requested comment on the type of data that should be reported and the required frequency of testing for breakaway oxidation. The NRC received many comments on this part of the rule. The industry commenters generally expressed views that the sample frequency should be reduced and be more flexible. The industry commenters also expressed views that requiring licensees to report breakaway oxidation results was unnecessary and that the fuel cladding vendors should address the concerns regarding breakaway oxidation with their quality assurance programs. A few fuel cladding vendors proposed that periodic test program plans could be developed by the fuel cladding vendors and approved by the NRC. The NRC agreed that the periodic testing and reporting requirements could be revised in a way that adds flexibility, decreases cost and burden of breakaway oxidation testing, and still achieves the safety objective. The NRC agreed that the objective of the rule can be achieved with rule language that requires a fuel vendor to submit a breakaway oxidation testing program for NRC review and approval and that the requirement for licensees to report breakaway oxidation results could be removed. The NRC changed the rule and associated regulatory guidance accordingly due to these comments.

Application of Ductility-Based Analytical Limits in the Burst Region:

During a postulated LOCA, a portion of the fuel rod population may be predicted to experience fuel rod ballooning and cladding rupture as a result of rapid depressurization of the reactor coolant system in combination with elevated cladding temperature. The existing rule explicitly prescribed how to calculate the ECR in the ballooned and ruptured region of the fuel rod. In the proposed rule, this prescription was removed from the rule language and an acceptable approach for evaluating post-quench ductility for the ballooned region was proposed to be provided in draft regulatory guidance. The NRC did not receive any comments on this part of the proposed rule. Therefore, this approach is reflected in the final rule and final regulatory guidance.

Long-Term Cooling Performance Requirement:

The demarcation between short-term and long-term ECCS performance demonstration depends on many variables including reactor coolant system design, ECCS design, break size, and break location. For example, a given PWR large-break loss-of-coolant accident (LBLOCA) evaluation may define entry into long-term cooling once the core is quenched, liquid level in the core is re-established, and ECCS coolant pumps draw suction from the containment sump. As a result of these design dependencies, the applicable entity must define the demarcation between short-term and long-term ECCS performance demonstration, as well as applicable evaluation models, analytical requirements, and, if necessary, analytical limits.

The existing regulation in § 50.46(b)(5) required that, for long-term cooling, the calculated core temperature be maintained at an acceptably low value following any calculated successful initial operation of the ECCS. It also required that decay heat be removed for the extended period of time required by the long-lived radioactivity remaining in the core. No explicit performance objective was defined for judging an acceptably low core temperature.

The final rule requires that the long-term ECCS recirculation coolant delivery to the core exceed the minimum flow necessary to remove decay heat loads such that core temperature tends to be stable or decreasing, or for cases where debris loading interferes with coolant delivery and prompts a post-quench reheat transient, the entity must demonstrate that no further fuel cladding failure occurs. If the entity predicts a debris-induced, post-quench reheat transient, which could reasonably result in further cladding failure, the entity would need to conduct research on post-quench fuel specimens to: 1) identify all degradation mechanisms, cladding failure modes, and any unique features of fuel rod performance during the predicted long-term temperature history; and 2) establish analytical limits and analytical requirements that demonstrate no further fuel cladding failure occurs.

The final rule requirement has evolved relative to the proposed rule. The proposed rule introduced a new requirement in § 50.46c(g)(1)(v) that stipulated that a LTC peak cladding temperature analytical limit be established which preserved cladding ductility based upon an NRC-approved test program. In the proposed rule, the NRC requested input regarding this new performance requirement to determine: 1) if cladding ductility was the most suitable performance-based metric; 2) if peak cladding temperature was the most suitable analytical limit; and 3) if a technical basis existed for long-term cladding performance. No commenter supported the proposed new requirement. Several commenters questioned whether cladding ductility was the most appropriate performance-based metric. These commenters noted that different cladding degradation mechanisms may exist at different post-quench temperature regimes. Several commenters questioned the use of a single analytical limit on PCT noting that time-at-temperature may be more appropriate to capture the degradation mechanisms. No commenter identified an existing technical basis for long-term, post-quench fuel performance. Several commenters requested that the existing§ 50.46 rule language be maintained.

The following provides a short history of regulatory considerations for long-term cooling and the basis for the final rule language.

Background:

General Design Criteria 35, “Emergency core cooling,” established the following principal ECCS design requirements:

> The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

The first requirement limits the degree of core damage to ensure “continued effective core cooling” is maintained for the long-term period required to remove decay heat. This requirement stems from an Advisory Task Force on Power Reactor Emergency Cooling, which in 1967 concluded that:

> The analysis of [a LOCA] requires that the core be maintained in place and essentially intact to preserve the heat-transfer area and coolant-flow geometry. Without preservation of heat-transfer area and coolant-flow geometry, fuel-element melting and core disassembly would be expected... Continuity of emergency core cooling must be maintained after termination of the temperature transient for an indefinite period until the heat generation decays to an

insignificant level, or until disposition of the core is made.[6]

This conclusion highlights the importance of preserving the heat-transfer area and coolant-flow geometry over the extended period of the accident.

During the 1973 rulemaking hearing, the AEC wrote:

> In view of the fundamental and historical importance of maintaining core coolability, we retain this criterion as a basic objective, in a more general form than it appeared in the Interim Acceptance Criteria. It is not controversial as a criterion... Although most of the attention of the ECCS hearings has been focused on the events of the first few minutes after a postulated major cooling line break, up to the time that the cladding would be cooled to a temperature of 300°F or less, the long term maintenance of cooling would be equally important.[7]

The result of the hearing was the existing regulation in § 50.46(b)(5) that requires that for long-term cooling the calculated core temperature be maintained at an acceptably low value following any calculated successful initial operation of the ECCS. It also requires that decay heat be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Commercial nuclear power plants have demonstrated compliance with these generalized requirements using numerous analytical techniques to evaluate core thermal and hydraulic behavior (e.g., core remains covered with a two-phase mixture while predicted recirculation flow exceeds heat removal flow requirements for decay heat) and identified timing for boric acid precipitation to establish timely operator actions to preclude such precipitation (e.g., redirect injection to prevent boron precipitation for large breaks and initiate cooldown to control boric

6 Report of Advisory Task Force on Power Reactor Emergency Cooling, TID-24226, 1967.
7 Atomic Energy Commission Rule-Making Hearing, Opinion of the Commission, Docket RM-50-1, December 28, 1973.

acid for small breaks). The staff has found these approaches acceptable for decades. In the absence of a debris-induced, post-quench reheat transient, the staff has determined that: 1) currently approved analytical models and methods continue to be acceptable and 2) no further fuel testing and analysis is required to satisfy the more explicit performance requirement discussed below.

Consideration of Debris:

Limitations in ECCS capability or phenomena interfering with ECCS coolant delivery challenge the long-term performance goal of continued effective core cooling. The possibility that debris may interfere with ECCS coolant delivery and promote a temporary, post-quench reheat transient necessitates more explicit LTC fuel performance goals than the existing generalized requirements.

Consideration of debris effects in demonstrating compliance with long-term cooling requirements will introduce further complexity in the simulation of accident progression and may require additional analytical rigor and operator actions. In addition to showing adequate ECCS coolant delivery to the core, the entity must be able to demonstrate that debris deposition does not significantly impair fuel rod heat transfer or block coolant flow.

In addition to explicitly requiring coolable geometry, the existing regulation in § 50.46(b)(5) also implicitly limits radiological releases. Maintaining long-term ECCS recirculation coolant delivery provides assurance that core geometry remains stable and further fuel cladding damage (beyond that experienced during the initial period of accident) would be minimal. Fuel rods that retained their integrity as a fission product barrier during the initial transient would likely continue to retain their integrity during the long term recirculation phase. The NRC considers any debris-induced, post-quench reheat transient resulting in fuel cladding failures as a loss of safety margin, and as such, inconsistent with the defense-in-depth concept. The final LTC rule language allows an applicant to predict a debris-induced, post-quench reheat transient during a postulated LOCA, provided the overarching goals are satisfied: 1) preserving a coolable geometry, 2) removing decay heat for the extended period of time required by the long-lived radioactivity remaining in the core, and 3) minimizing the release of fission products are satisfied. The performance metric that satisfies all of these safety goals is the preservation of fuel cladding integrity.

If an entity predicts a debris-induced, post-quench reheat, the entity should demonstrate that no further fuel cladding failure occurs, beyond that associated with the short-term period. Fuel cladding failure includes, but is not limited to, perforations due to excessive local oxidation, ruptures due to differential pressure, and cladding fragmentation due to loss of ductility.

As described in Section I, “Background,” a comprehensive research program was completed that identified all known degradation mechanisms and unique features of fuel rod performance under accident conditions, and it helped establish performance-based objectives and analytical requirements for current fuel designs (i.e., $UO_2$ pellets within cylindrical zirconium alloy cladding). This research program only addressed fuel performance during the initial, short-term period of the accident and focused on preserving cladding ductility. No technical basis was developed to establish metrics or objectives for cladding performance during a postulated post-quench reheat transient. Based upon the duration and magnitude of the post-quench reheat transient, several different degradation mechanisms may exist, which could challenge cladding integrity. The dominant failure mode will depend on the temperature history.

If an entity determines that a debris-induced, post-quench reheat could reasonably result in further cladding failure, then that entity would need to conduct research on post-quench fuel specimens to:

1) identify any potential degradation mechanisms, cladding failure modes, and any unique features of fuel rod performance during the predicted long-term temperature history, and
2) establish analytical limits and analytical requirements that demonstrate no further fuel cladding failure could occur in the long-term period.

<u>Cladding Inner Surface Oxygen Ingress Analytical Requirement</u>:

To address oxygen ingress to the cladding inner surface, the NRC has added an analytical requirement to the ECCS evaluation model. If an oxygen source is present on the inside surfaces of the cladding at the onset of a LOCA, a licensee or applicant must consider the effects of oxygen diffusion from the cladding inside surfaces in the ECCS evaluation model. This requirement is unchanged from the proposed rule. The NRC did not receive any comments on this part of the proposed rule.

The NRC recognizes that the availability of a cladding inner surface oxygen source and its diffusion into the base metal during a postulated LOCA may depend on several factors (e.g., rod design, power history). As such, applicants are responsible for determining when the fuel-cladding bonding layer is strong enough to allow the diffusion of oxygen from the uranium-oxide fuel to the zirconium cladding and, therefore, must be included in the ECCS evaluation model.

It is anticipated that identifying the magnitude and onset of oxygen diffusion on the cladding inner surface would be part of the NRC's review and approval of ECCS evaluation models or vendor fuel designs. A conservative analytical limit is provided in RG 1.224.

<u>Crud and Oxide Layer Analytical Requirement:</u>

As discussed in Section I, "Background," on March 15, 2007, Mr. Mark Leyse submitted a PRM to the NRC requesting that all holders of operating licenses for nuclear power plants be required to operate such plants at operating conditions (e.g., levels of power production and light-water coolant chemistries) necessary to effectively limit the thickness of crud and/or oxide layers on fuel rod cladding surfaces. The petitioner requested that the NRC conduct rulemaking in the following three specific areas:

1) Establish regulations that require licensees to operate light-water power reactors under conditions that are effective in limiting the thickness of crud and/or oxide layers on zirconium-clad fuel in order to ensure compliance with § 50.46(b) ECCS acceptance criteria;

2) Amend Appendix K to 10 CFR part 50 to explicitly require that steady-state temperature distribution and stored energy in the reactor fuel at the onset of a postulated loss-of-coolant accident (LOCA) be calculated by factoring in the role that the thermal resistance of crud deposits and/or oxide layers plays in increasing the stored energy in the fuel (these requirements would also apply to any acceptable, best-estimate ECCS evaluation models used in lieu of Appendix K to 10 CFR part 50 calculations); and

3) Amend § 50.46 to specify a maximum allowable percentage of hydrogen content in (fuel rod) cladding.

Licensees use approved fuel performance models to determine fuel conditions at the start of a LOCA, and the impact of crud and oxidation on fuel temperatures and rod internal pressures may be determined explicitly or implicitly by the system of models used. With the addition of an unambiguous regulatory requirement to address the accumulation of crud and oxide during plant operation, the NRC believes that fuel performance and LOCA evaluation models must include the thermal effects of both crud and oxidation whenever their accumulation would significantly affect the calculated results. The NRC notes that licensees are required to operate their facilities within the boundary conditions of the calculated ECCS performance. During or immediately after plant operation, if actual crud layers on reactor fuel are implicitly determined or visually observed after shutdown to be greater than the levels predicted by or assumed in the ECCS evaluation model, licensees will be required to determine the effects of the increased crud on the calculated results. In many cases, engineering judgment or simple calculations could be used to evaluate the effects of increased crud levels; therefore, detailed LOCA reanalysis may not be required. In other cases, new analyses would be performed to determine the effects that the new crud conditions have on the final calculated results (i.e., PCT and integral time-at-temperature). If unanticipated or unanalyzed levels of crud are discovered, then the licensee must determine if correct consideration of crud levels would result in a reportable condition as provided in the relevant reporting paragraphs. The NRC believes this regulatory approach to address crud and oxide accumulation during plant operation satisfactorily addresses the issues raised by the petitioner's first request.

The formation of cladding crud and oxide layers is an expected condition at nuclear power plants. Although the thickness of these layers is usually limited, the amount of accumulated crud and oxidation varies from plant to plant and from one fuel cycle to another. Intended or inadvertent changes to plant operational practices may result in unanticipated levels of crud deposition. The NRC agrees with the petitioner (the petitioner's second request) that crud and/or oxide layers may directly increase the stored energy in reactor fuel by increasing the thermal resistance of cladding-to-coolant heat transfer, and may also indirectly increase the stored energy through an increase in the fuel rod internal pressure. As such, to ensure that licensee ECCS evaluation models properly account for the thermal effects of crud and/or oxide layers that have accumulated during operations at power, the final rule adds a new requirement to evaluate the thermal effects of crud and oxide layers that may have accumulated on the fuel cladding during plant operation. Therefore, the NRC believes this regulatory approach resolves the second request of PRM-50-84.

The petitioner's third request is for the NRC to establish a maximum allowable percentage of hydrogen content in fuel rod cladding. The purpose of this request is to prevent embrittlement of fuel cladding during a LOCA. Although the NRC decided not to propose the specific rule language recommended by the petitioner, the proposed new zirconium-specific requirements address the petitioner's third request by considering cladding hydrogen content in the development of analytical limits on integral time at temperature.

The NRC believes that this final rule addresses each of the three issues raised in PRM-50-84. Therefore, the NRC considers PRM-50-84 to be granted in part and resolved.

## D. Risk-Informed Alternative to Address Debris for Long-Term Cooling

The rule allows an entity to use the alternative risk-informed approach to demonstrate compliance with the requirements in paragraph (d)(2)(iii) of the rule, “Core geometry and coolant flow,” if requested by the entity and approved by the NRC. NRC approval of an entity’s risk-informed approach allows the entity to exclude the effects of debris in its analysis of long-term cooling that is required in paragraph (d)(2)(i), “Realistic ECCS evaluation model,” or (d)(2)(ii), “Appendix K model,” of the rule. However, it is not the NRC’s intent that this approach be used to justify the use of problematic insulation or other debris sources in new reactor designs or for plant modifications. Similar to GSI-191, the NRC intends to approve use of the alternative risk-informed approach only in cases where an issue emerges that could not have been readily foreseen or addressed during the design or implementation processes and where removal of such debris could pose an undue burden not justified by the risk posed by the debris.

For the purpose of § 50.46c provisions on the risk-informed alternative to long-term cooling, debris is material in any location that may be transported to a location that could degrade long-term core cooling; e.g. to the suction strainer(s) for the ECCS and CSS in the case of a PWR. Debris includes (but is not limited to) loose materials that may transport and materials that may be damaged by a LOCA jet to the extent that they become transportable. Debris sources of interest typically include insulation, coatings, dust, dirt, concrete, fire barrier material, signs and tags, and materials left in containment; however, debris may originate from other sources. Debris may also result from chemical interactions that cause precipitation of materials. Debris may cause increased head loss across the strainer and restrict the flow of water to the ECCS and CSS pumps. Debris may also pass through the strainer and cause blockage of components or the core, or damage to components downstream of the strainer.

The § 50.46c provisions allowing a risk-informed approach for evaluating the effects of debris on long-term cooling performance require that sufficient defense-in-depth and safety margins be maintained and, as a result, defense-in-depth and safety margins must be explicitly considered. This consideration of defense-in-depth and safety margins is consistent with the NRC's general guidance regarding risk-informed decisionmaking contained in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk Informed Decisions on Plant Specific Changes in the Licensing Basis," Revision 2, dated May 2011 (ADAMS Accession No. ML100910006). However, the terms defense-in-depth and safety margins, while frequently used, can have different definitions and considerations for different stakeholders. Therefore, the NRC intends to ensure that entities using the risk-informed approach to debris include in their application, final safety analysis report (FSAR), and/or updated safety analysis report (UFSAR), as appropriate, a list of applicable design, plant, and operational capabilities of defense-in-depth and safety margins with respect to the implementation of the risk-informed alternative.

The RG 1.174 guidance provides an acceptable approach to risk-informed decision-making related to changes to the licensing basis, consistent with the Commission's Policy Statement on the Use of Probabilistic Risk Assessment (PRA) dated August 16, 1995 (60 FR 42622). The RG sets forth a set of five key principles, four of which are relevant to the proposed rule[8]:

- Maintain sufficient defense-in-depth;
- Maintain sufficient safety margins;
- Result in no more than a small increase in risk and be consistent with the Commission's

Safety Goal Policy Statement; and

8 The first key principle of risk-informed decisionmaking is that the existing regulations (or existing licensing basis) are met, unless the proposed change is related to a specific exemption under 10 CFR 50.12. In this context, the first key principle is met when an entity complies with the subject final rule.

- Implement performance measurement strategies to assure continued validity of the risk-informed approach.

Each of these key principles for risk-informed decisionmaking are discussed further below and in RG 1.229, "Risk-Informed Approach for Addressing the Effects of Debris on Post Accident Long-Term Core Cooling." Defense-in-depth has been interpreted in many different ways and has traditionally been applied in reactor design and operation to provide multiple means of accomplishing safety functions and to prevent the release of radioactive material. The requesting licensee or applicant must address the underlying purpose of the general design criteria (or similar licensing basis design criteria), national standards, and engineering principles (e.g., single failure criterion) in evaluating the impact of the alternative approach on defense-in-depth. The current guidance in RG 1.174 on defense-in-depth states that it is considered sufficiently maintained if the following attributes are achieved:

- Reasonable balance is preserved among prevention of core damage, prevention of containment failure or bypass, and mitigation of consequences of an offsite release.
- There is not an over-reliance on programmatic activities to compensate for weaknesses in plant design.
- System redundancy, independence, and diversity are sufficiently preserved commensurate with the expected frequency of challenges, consequences of failure of the system, and the associated uncertainties in determining these parameters.
- Defenses against potential common cause failures are sufficiently preserved and the potential for the introduction of new common cause failure mechanisms are assessed and addressed.

- Independence of barriers is not significantly degraded.
- Defenses against human errors are sufficiently preserved.
- The underlying purpose of the plant's design criteria is sufficiently maintained.

Regarding the maintenance of sufficient safety margins, the entity needs to address the impact of implementing the alternate approach on current safety margins. Consistent with the current guidance in RG 1.174, sufficient safety margins are considered to be maintained when:

- Codes and standards, or their alternatives approved for use by the NRC, are met.
- Safety analysis acceptance criteria in the licensing basis, or alternative acceptance criteria approved by the NRC, are met.

The risk-informed provisions for considering the effects of debris on long-term cooling also require that any potential increase in risk from implementation of the risk-informed approach be assessed and that reasonable confidence be provided that this increase in risk is small. In addressing the increase in risk, the NRC generally considers two metrics in considering licensing basis changes: core damage frequency (CDF) and large early release frequency (LERF). For 10 CFR part 52 design certification applicants and holders, part 52 combined license applicants, and part 52 combined license holders before the Commission finding under § 52.103(g) there is another risk metric, large release frequency (LRF), that is used to demonstrate, in part, the achievement of the Commission expectation that new reactor designs have incorporated improved safety capabilities. This design-related expectation metric, which is a LRF of less than $10^{-6}$ per year, must still be demonstrated even if the risk-informed alternative approach is pursued by one of these part 52 entities. In the context of this rule, the calculated increases in CDF ($\Delta$CDF) and LERF ($\Delta$LERF) represent the difference between the as-built, as-operated plant (or as-to-be-built, as-to-be-operated plant for part 52 applicants and holders prior to the Commission finding under § 52.103(g)) accounting for the effects of debris and the "baseline" plant where the effects of debris are assumed to be negligible. For plants with a total CDF of $10^{-4}$ per year or less, the NRC regards as "small" a CDF increase of up to $10^{-5}$ per year, and for plants with a total CDF that is greater than $10^{-4}$ per year, "small" is considered to be a CDF increase of up to $10^{-6}$ per year. However, if there is an indication that the total CDF may be considerably higher than $10^{-4}$ per year, the focus of the entity should be on finding ways to decrease rather than increase CDF and the entity may be required to present arguments as to why steps should not be taken to reduce CDF in order for the alternate approach to be considered. In addition, for plants with a total LERF of $10^{-5}$ per year or less, "small" is considered to be a LERF increase of up to $10^{-6}$ per year, and for plants with a total LERF that is greater than $10^{-5}$ per year, "small" is considered to be a LERF increase of up to $10^{-7}$ per year. Similar to the CDF metric, if there is an indication that the total LERF may be considerably higher than $10^{-5}$ per year, the focus of the entity should be on finding ways to decrease rather than increase LERF and the entity may be required to present arguments as to why steps should not be taken to reduce LERF in order for the alternate approach to be considered. This perspective is consistent with the current guidance in RG 1.174.

If an entity elects to implement the alternative risk-informed approach, § 50.46c requires it to utilize systematic processes that evaluate the risk of debris for internal and external events initiated during full power, low power, and shutdown operation. At a minimum, an internal-events, at-power PRA capable of estimating CDF and LERF is required. This PRA must reasonably reflect the current plant configuration and operating practices and use applicable plant and industry operating experience. It must also be of the appropriate scope, level of detail, and technical adequacy needed for this alternative process. The technical adequacy can be determined as set forth in the latest version of RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," (the current version is dated March 2009, ADAMS Accession No. ML090410014). In addition to the PRA, other assessment methods and techniques may be used to address the hazards and plant operational modes that are not covered in the PRA. These include margin-based methods, bounding risk assessments, or other approaches. Engineering calculations and analyses, tests, and other supporting information are also considered part of the systematic process that evaluates the risk of debris. The above aspects are collectively referred to as the "systematic risk assessment."

The rule also requires that systematic risk assessment be performed under a quality assurance program, that a monitoring program be utilized to ensure the continued validity of that assessment and its results, and that the systematic risk assessment be updated at least every four years. In addition, § 50.46c contains requirements for corrective action and reporting for conditions when the established risk-informed approach results exceed established thresholds for risk, defense-in-depth, and safety margins. Together, these requirements maintain the validity of the risk-informed approach such that the risk-informed decisionmaking principles continue to be satisfied over the life of the facility.

In as much as § 50.46c contains requirements that: 1) ensure any net risk increase from implementation of its requirements is small, 2) maintain sufficient defense-in-depth, 3) maintain sufficient safety margins, and 4) require the use of monitoring and performance measurement strategies, the final rule is consistent with the Commission's policy on the use of probabilistic risk assessment (PRA) for risk-informed decision-making and, more importantly, would maintain adequate protection of public health and safety.

*Development of Regulatory Guidance for the Risk-Informed Alternative.*

South Texas Project Nuclear Operating Company (STPNOC) submitted a letter of intent to pilot a risk-informed approach for addressing GSI-191 (ADAMS Accession No. ML103481027) in December 2010. Subsequently, the NRC received a pilot submittal from STPNOC on January 31, 2013 (ADAMS Accession No. ML13043A013), as supplemented on June 19, 2013 (ADAMS Accession No. ML131750250), and supplemented again on November 13, 2013 (ML13323A128). After significant staff review and discussions with the licensee STPNOC changed their methodology to a more simplified approach and submitted another supplement to its submittal on August 20, 2015 (ML15246A125). The NRC concluded that the methodology used in the final submittal is acceptable. The NRC developed guidance for the risk-informed alternative to address the effects of debris on long-term cooling, building off lessons learned while conducting the review of the application and through a number of public meetings to obtain stakeholder input. The draft regulatory Guide, DG-1322, ''Risk-Informed Approach for Addressing the Effects of Debris On Post-Accident Long-Term Core Cooling,'' was issued for public comment on April 20, 2015. Comments on the draft guide were incorporated into the final guidance, as appropriate. For more information on this regulatory guide, see Section XVIII, "Availability of Guidance."

**E. Corrective Actions and Reporting Requirements.**

*1. Deterministic ECCS Performance.*

The final rule clarifies existing reporting and corrective action requirements in order to resolve recurring issues involving the interpretation of the current regulations' requirements. The draft final rule distinguishes three possible combinations of reporting criteria based upon predicted response, level of significance (i.e., significant or not significant, as defined by the proposed rule), and whether the error, change or operation would result in any exceeded acceptance criteria. For each scenario, the proposed rule provides the required actions, reports, and a time frame for providing the necessary reports.

Presently, the reporting requirements in 10 CFR 50.46(a)(3) require that licensees report changes to or errors in an ECCS evaluation model, or in the application of the evaluation model, and the estimated effect of the changes or errors on predicted peak cladding temperature. The final rule expands the definition of a significant change or error to include integral time-at-temperature. The NRC made this change to improve the content and communications of reports submitted to the NRC. The NRC also made this change to inform the staff's response to future changes to or errors discovered in ECCS evaluation models, or in the applications thereof.

Many comments were received on Paragraph (m) corrective actions and reporting requirements. Based on these comments, the rule was modified to improve clarity and allow 60 days for reporting significant errors or changes that do not cause acceptance criteria to be exceeded.

The rule adds a reporting requirement and definition of significant change or error based on predicted changes in integral time-at-temperature (i.e., ECR) and reformats the reporting section to clarify existing requirements. Any changes or errors that prolong the temperature transient may further challenge the the post-quench ductility analytical limits; however, they may not significantly change the predicted PCT. As such, this change or error would not be captured in the existing reporting requirements. To improve the reporting and evaluation of changes or errors of this type, the NRC is expanding the definition of significant change or error to include integral time-at-temperature. The threshold for a significant change or error, 0.4 percent ECR, is equivalent to a change in calculated ECR for a 50 °F change in cladding temperature.

The definition of a significant change or error (i.e., 50 °F PCT, 0.4 percent ECR) is specific to zirconium-alloy cladding. A new definition of significant change or error may be necessary for other cladding materials. In addition, the rule requires the use of maximum local oxidation (i.e., percent ECR) to evaluate the impact of a change or error on the predicted integral time-at-temperature.

Existing reporting requirements § 50.46(a)(3) with respect to any "change to or error discovered in an acceptable evaluation model or in the application of such a model" have been a source of confusion. Two areas of common misconceptions are related to: 1) the baseline PCT and integral time-at-temperatrue values when estimating a significant change or error (i.e., greater than 50 °F), and 2) the 30-day reporting requirement including "a proposed schedule for providing a reanalysis or taking other action as may be needed to show compliance with § 50.46 requirements." In the final rule, the NRC has revised the reporting requirements to: 1) identify more clearly the baseline values to be used in reporting pursuant to the requirements of this section, and 2) distinguish between the requirements for proposing a reanalysis schedule, and for taking other action as may be needed to show compliance with § 50.46c requirements. When estimating the effect of a significant change or error, the rule provides threshold values for both PCT and integral time-at-temperature. The baseline predictions used to assess a significant change or error should be the PCT and integral time-at-temperature values documented in a plant's updated final safety analysis report (UFSAR). These values should represent the latest LOCA analyses that were submitted and reviewed by the NRC staff as part of a license amendment request (e.g., power uprate, fuel transition), or as incorporated into the facility licensing basis in accordance with NRC-approved reload licensing methods, as amended by prior annual reports. The following example illustrates the NRC's position:

> In 2007, a licensee submits new LOCA analyses as part of an extended power uprate license amendment request with a predicted PCT of 1900 °F and maximum local oxidation (MLO) of 2.4 percent ECR. The 2008 and 2009 annual reports identify no changes or errors. In 2010, two errors in the ECCS evaluation model are discovered and documented in the annual report with an estimated impact on PCT of +25 °F and -20 °F and estimated impact on MLO of +0.08 percent ECR and -0.01 percent ECR. A 30-day notification was not required since the estimated impact was below the threshold for a significant change or error. At this point, the licensee was required update the UFSAR, document the error notification, and identify the baseline for judging future changes or errors as 1905 °F PCT and 2.5 percent ECR.

In the existing rule language, there is a requirement to include a "proposed schedule for providing a re-analysis or taking other action as may be needed to show compliance with § 50.46 requirements" in a report describing the nature of a significant error or change, and its estimated effect on the predicted PCT. This language has led to a misconception that, when a significant error is reported that does not cause the predicted PCT to exceed its 2200 °F acceptance criterion, a proposed schedule for providing a reanalysis is not required and taking other action is not needed to show compliance with the requirements. It has long been the NRC's determination that facility operation in excess of the § 50.46 acceptance criteria is an immediate safety concern that requires prompt corrective action, and the "other action" language in the rule was intended to address that concern. This concept is further underscored by the final sentence of the existing paragraph at § 50.46(a)(3)(iii): "The affected applicant or holder shall propose immediate steps to demonstrate compliance or bring plant design into compliance with § 50.46 requirements." Therefore, the reporting and reanalysis requirements have been further separated to distinguish the requirements applicable when a significant change or error is identified that results in facility operation in excess of the § 50.46 acceptance criteria, from those that apply when a significant change or error is identified that does not result in facility operation in excess of the § 50.46 acceptance criteria.

When a change to or error in an ECCS evaluation model, or in the application of such a model, is discovered, the licensee would be responsible for estimating the magnitude of changes in predicted results to: 1) determine if immediate steps are necessary to demonstrate compliance or bring plant design or operation into compliance with § 50.46c requirements, and 2) identify reporting requirements. Under the final rule, a licensee's obligation to report and take corrective action varies depending upon whether the licensee's situation falls into one of three possible scenarios, as described below:

1. *Change or error that does not result in any predicted response that exceeds any acceptance criteria and is itself not significant*.

The licensee must:

a. Submit an annual report documenting the change(s) and/or, error(s), along with the estimated magnitudes of changes in predicted results.
b. Revise the UFSAR in accordance with § 50.71(e).
c. Use the revised UFSAR PCT/ECR predictions as a baseline for future evaluations.

2. *Change or error that does not result in any predicted response that exceeds any acceptance criteria but is significant*.

The licensee must:

a. Submit a 60-day report documenting the change(s) and/or error(s), estimated magnitudes of changes in predicted results, and the schedule for providing a new analysis of record (AOR).
b. In accordance with the schedule proposed in a), provide the re-analysis to the NRC. This may be accomplished by submitting a subsequent report, pursuant to § 50.46(m)(1), describing the re-analysis and providing the updated results. The re-analysis shall be performed using an acceptable evaluation model.
c. Revise the UFSAR to include new evaluation model results in accordance with § 50.71(e).

d. Use the revised UFSAR evaluation model results as a baseline for the future evaluations.

3. *Change or error that results in any predicted response that exceeds any of the five acceptance criteria established at § 50.46c(g)(1)(i-v).*

The licensee must:

a. Take immediate actions to bring the plant into compliance with the acceptance criteria.

b. Report the change(s) and/or error(s) under §§ 50.55(e), 50.72, and 50.73, as applicable.

c. Submit a 30-day report documenting the change(s) and/or error(s), estimated magnitudes of changes in predicted results, description of corrective actions and/or compensatory measures, and the schedule for providing a new AOR.

d. In accordance with the schedule proposed in (c), provide the re-analysis to the NRC. This may be accomplished by submitting a subsequent report, pursuant to § 50.46(m)(1), describing the re-analysis and providing the updated results. The re-analysis shall be performed using an acceptable evaluation model. Revise the UFSAR to include new evaluation model results in accordance with § 50.71(e).

e. Use the revised UFSAR evaluation model results as the baseline for future evaluations.

The reporting requirements in § 50.46c(m) reflect reformatting of the existing reporting provisions in order to separately identify these three scenarios and clarify their respective requirements.

Explicit reporting requirements have not been established for breakaway oxidation, maximum hydrogen generation, or long-term cooling. However, the language in § 50.46c(m)(1)(iii) and § 50.46c(m)(2)(i) are intended to apply to changes to, or errors in, ECCS evaluation models, or the applications thereof, that affect the predicted performance relative to all of the acceptance criteria contained in § 50.46c. For example, an error or change in a PWR boron precipitation calculation that invalidates the timing for emergency operating procedures is considered a condition in which a plant's conformance to § 50.46c(g)(1)(v) is uncertain. In this circumstance, the NRC would consider this a potentially serious safety issue in need of immediate attention and potential correction.

2. *Risk-Informed Alternative to Address Debris for Long-Term Cooling.*

After NRC approval of an entity's request to use the risk-informed, alternative approach, changes to the baseline CDF and LERF values as well as $\Delta$CDF and $\Delta$LERF, or the defense-in-depth or safety margins attributes credited by the plant in the risk-informed analysis may occur. The changes in these criteria may result from changes to the design or physical plant, changes to the systematic risk assessment, including the PRA model, or may be recognized by identifying an error in the systematic risk assessment. Because the NRC staff's original decision is based in part on these metrics and requirements, subsequent changes need to be assessed to ensure that the basis for the NRC staff's original decision remains valid. Therefore, the rule contains reporting, corrective action, and update requirements that ensure that changes and errors are evaluated, reported, and corrected in a timely manner, as appropriate, consistent with the requirements specific to each. The cumulative impact of changes on risk is addressed by requiring that the total CDF and LERF resulting from debris, as compared to a debris-free plant, remain within the rule's acceptance criteria.

The reporting, corrective action, and update requirements associated with the risk-informed, alternative approach of § 50.46c(e) are established in § 50.46c(m)(6) through (m)(8). The rule recognizes that there are different requirements for different entities, as depicted in Table 1, Corrective Actions and Reporting: Risk-Informed Approach, and discussed below.

Table 1 - Corrective Actions and Reporting: Risk-Informed Approach

| Entity | Periodic Update (§ 50.46c(m)(8))? | Reporting (§ 50.46c(m)(6))? | Corrective Action (§ 50.46c(m)(7))? |
|---|---|---|---|
| Design certification applicant before issuance of final design certification rule | No | Yes (in accordance with Part 21) | Yes (ammendment to application) |
| Design certification applicant during the period of validity under § 52.55(a) and (b) – not currently referenced in any combined operating license (COL) application or COL | No | No | No |
| Design certification applicant during the period of validity under § 52.55(a) and (b) – once referenced in a COL application or COL | No | Yes (in accordance with Part 21) | Yes (amendment to the design certification) |
| Design certification renewal applicant | No | Yes (in accordance with Part 21) | Yes (amendment to the application) |
| Part 52 combined license applicant or Part 50 license applicant | No | Yes (report concerning compliance with § 50.46c(e)(1)) | Yes (amendment to the application) |
| Combined license holder before finding under § 52.103(g) | No | Yes (in accordance with § 52.99) | Yes (amendment to the combined license) |
| Operating license holder or combined license holder after finding under § 52.103(g) | Yes (at least every 48 months after initial | Yes (in accordance with §§ 50.72 or | Yes (correct condition to achieve |

| | | | |
|---|---|---|---|
| | NRC approval and subsequently after the latest update; Part 52 combined license holder also perform an update prior to initial fuel load) | 73) | compliance to § 50.46c(e)(1) and may need amendment to license if means of compliance changes) |

If an entity that implements the risk-informed approach to address debris effects, at any time, determines that the risk of debris is greater than the acceptance criteria in the rule or that defense-in-depth or safety margins have decreased from the NRC-approved analysis, then the entity must take the reporting actions required by § 50.46c(m)(6).

For design certification applicants (i.e., prior to issuance of the final design certification rule), this rule requires that, if any errors are discovered, the applicant must submit a report to the NRC within an amended application. That amended application would describe any changes to the certified design and/or changes in the analyses, evaluations, and modeling (including the debris evaluation model and the PRA and its supporting analyses); and would demonstrate that the acceptance criteria in § 50.46c(e)(1) are met.

For design certification applicants during the period of validity under § 52.55(a) and (b) that are not currently referenced in any COL application or COL, there would be no evaluation, reporting, or change requirement. However, once the design certification is referenced by a COL applicant, any information regarding compliance with § 50.46c(e)(1) must be reported in accordance with the requirements in 10 CFR part 21.

If an entity that implements the risk-informed approach to address debris effects, at any time, determines that either the acceptance criteria of § 50.46c(e)(1)(i) have been exceeded or the requirements of § 50.46c(e)(1)(ii) are no longer met, then the entity must take the applicable corrective actions required by § 50.46c(m)(7) in a timely manner.

Periodic update requirements are established in § 50.46c(m)(8) for both Part 50 and Part 52 licensees that implement the risk-informed approach to address debris effects. Part 52 combined license holders that are approved for the risk-informed approach prior to entering operations must update of their analyses, evaluations, and modeling performed under § 50.46c(e) prior to initial fuel load. This update must correct any identified errors and incorporate any licensee-adopted changes to the: plant design, proposed operational practices, or applicable industry operational experience known to the licensee. As appropriate, the licensee shall also update the systematic risk assessment, including the PRA and supporting analyses, and re-perform the evaluations of risk, defense-in-depth, and safety margins to confirm that the acceptance criteria and requirements in § 50.46c(e)(1) of the rule continue to be met. In addition, part 50 and part 52 licensees that implement the risk-informed approach are required to periodically update their systematic risk assessment to confirm that the acceptance criteria and requirements in § 50.46c(e)(1) of the rule continue to be met. The rule establishes the frequency of the periodic update to be no greater than 48 months after the initial NRC approval for the entity to use the risk-informed approach and then periodically thereafter no later than 48 months since the latest update. The rule recognizes that the update requirements do not apply to design certification entities or combine license holder entities before the Commission finding under § 52.103(g).

In addition to the reporting, corrective action, and update requirements identified in § 50.46c(m)(6) through (8) related to the risk-informed approach, the NRC's approval under

§ 50.46c(e)(3) will specify the circumstances under which the entity is required to notify and seek additional approval from the NRC of changes or errors in the risk-informed approach. This requirement ensures that if errors are identified subsequent to the NRC approval or if the entity seeks to change specific aspects of their approach that are determined by the NRC to be important to the NRC approval, such as the scope or level of detail of the PRA, these circumstances will be clearly identified in the NRC's approval of the entity's original or subsequent request to implement the risk-informed approach.

**F. Implementation.**

Paragraph (p), *Implementation*, of the final rule specifies when each entities to whom this rule applies (as specified in paragraph (a) of § 50.46) must be in compliance with the requirements of the final rule for ECCS system and fuel system design. The rule also allows entities to voluntarily seek to meet the long-term cooling requirements in paragraph (e) of the final rule (and other changes as permitted by the risk-informed alternative and noted in the application) using a risk-informed approach at any time after the effective date of the final rule (*i.e.*, 30 days after publication in the *Federal Register*). Because the risk-informed alternative in paragraph (e) is voluntary, paragraph (p) does not provide for an implementation schedule for the use of the risk-infomed alternative.

*Implementation of ECCS system design and fuel system design requirements for existing operating reactors*

For existing operating nuclear power reactors, the final rule requires each licensee to submit an implementation plan and schedule for achieving compliance with the provisions of this regulation with the exception of the consideration of debris effects under § 50.46c(d)(2)(iii) of this section. As described in Section I.B, "Background, Generic Safety Issue (GSI)-191 and Long-Term Cooling," consideration of debris is not a new requirement and is being addressed in separate programs by the industry. The implementation plan must be submitted within 6 months of the effective date of the rule and include details needed by the "staff to understand the scope of the compliance demonstration in order to manage the overall implementation schedule and their work load. Specifically, the implementation plan needs to identify the evaluation model(s), fuel design(s) and cladding alloy(s), and analytical limits to be used in the ECCS performance demonstration, along with the relative level of effort needed to complete the performance demonstration. The following chart should be used for establishing the relative level of effort:

| Level of Effort | Scope of Performance Demonstration |
|---|---|
| Level I | • Maintain existing LOCA evaluation model(s)[9]<br>• Minimal calculations (e.g., post-processing, integrate time-at-temperature with Cathcart-Pawel correlation) |
| Level II | • Maintain existing LOCA evaluation model(s)[10]<br>• Partial reanalysis with limited number of new LOCA simulations (e.g., burnup/rod power-dependent cases) |
| Level III | • New LOCA evaluation model(s)<br>• Complete reanalysis with new break spectrum |

9 As modified to comply with new § 50.46c requirements.
10 As modified to comply with new § 50.46c requirements.

| | |
|---|---|
| | - or -<br>• New PQD testing to develop alloy-specific or PCT-specific analytical limits. |

The schedule must identify, for each element of the ECCS performance demonstration required to be submitted to the NRC for review (e.g., evaluation model, hydrogen uptake model, cladding alloy), the earliest possible date for submission and the expected date of submission. To maintain an up-to-date schedule, the final rule requires each licensee to provide an update to the implementation plan and schedule every 12 months until the license amendment request has been submitted and accepted for review.

The final rule requires that compliance demonstration be in the form of a license amendment application under § 50.90 and must be submitted no later than 60 months after the effective date of the rule. Furthermore, the final rule requires licensees to be in compliance with the requirements of § 50.46c no later than 84 months after the effective date of the rule.

Upon receipt of the implementation plans, the NRC staff will develop an integrated schedule to manage the implementation for the existing fleet. The staff may request that certain schedules be adjusted in order to balance work load and provide confidence that reviews will be completed before the 84 month compliance deadline.

This implementation approach differs from the proposed rule, in that plant-specific compliance dates have been removed from the rule. Instead, the implementation approach relies upon NRC approval of implementation plan submitted by each currently-operating licensee as described above. The NRC has decided to adopt the new implementation approach based on an alternative implementation approach suggested by the industry, because the NRC believes that the approach in the final rule is easier to implement by the licensees while meeting the NRC's regulatory objectives.

*Implementation of ECCS system design and fuel system design requirements for design approvals, design certifications, combined licenses and manufacturing licenses*

For applicants and holders of standard design approvals, design certifications, combined licenses, and manufacturing licenses issued under 10 CFR part 52., paragraph (p) specifies when compliance with the final rule must be achieved. The rule reflects the NRC's determination that reactor designs reviewed and approved under part 52 should have the same constraints as the reactors operating under part 50 with respect to development, submittal, and approval of ECCS performance models necessary to demonstrate compliance with this rule. Alloy-specific hydrogen uptake models and all ECCS performance model updates would be expected to be submitted in a timely manner for NRC review and approval so that demonstration of the ECCS performance with respect to the analytical limits would not impact plant operation more than is necessary. The rule also reflects the NRC's expectation that, for new reactors licensed to operate prior to the effective date of the rule, operation for at least the initial fuel cycle using fuel that has not been analyzed under the rule's provisions accounting for burn-up effects does not present an adequate protection concern. During the initial fuel cycle, the NRC believes that burn-up effects would not be limiting, and the existing ECCS rule's acceptance criteria are sufficient during the initial fuel cycle to provide reasonable assurance of adequate protection with respect to overall ECCS performance.

*Implementation of risk-informed alternative to address debris effects during long-term cooling.* Implementation of the risk-informed alternative approach to addressing the impact of debris on long-term cooling is independent from implementation of the requirements related to the embrittlement research findings. A licensee or applicant may elect to submit its risk-informed alternative under § 50.46c(e) prior to demonstrating compliance with the other requirements of § 50.46c. The NRC is allowing early implementation because the NRC encourages entities to complete resolution of GSI-191 and this risk-informed alternative is one way of resolving that generic issue.

The NRC has determined that an entity's decision to use a risk-informed approach to evaluate the effects of debris on ECCS and containment spray system (CSS) with respect to long-term cooling following a LOCA should be reviewed and approved by the NRC prior to implementation. The ECCS and CSS are significant safety systems. The design bases for the ECCS are of high regulatory significance to the NRC, as reflected in the detailed requirements applicable to the ECCS (and the associated fuel system) in § 50.46 and appendix K to 10 CFR part 50. In addition, the design bases for the ECCS and the CSS affect the design bases for many other SSCs throughout the nuclear power plant. Therefore, changes to the design assumptions for the ECCS and CSS may have significant effects on the design bases for other SSCs throughout the plant. These potential effects include changes in the consequences of postulated accidents, safety margins, and defense-in-depth.

The NRC also determined that an entity could not use a risk-informed approach for evaluating the effects of debris on long-term without prior NRC review and approval. 10 CFR

50.59 would not allow such a change to the facility as described in the FSAR, as updated. The NRC considered a risk-informed approach for addressing the effects of debris on long-term cooling to be a departure from a method of evaluation described in the FSAR, as updated, used in establishing the design bases in the safety analysis as defined in § 50.59(a)(2). Hence, under § 50.59(c)(2)(viii), a licensee's departure from the existing methodology for evaluating long-term cooling must be reviewed and approved by the NRC as a license amendment.

In sum, given the importance of the ECCS and CSS, the "cascading" effects of changes in ECCS and CSS design on the design bases of other structures, systems, or components (SSCs) of a nuclear power plant, the NRC has determined that a licensee's decision to use a risk-informed approach to evaluate the effects of debris on ECCS with respect to long-term cooling must be reviewed and approved by the NRC. The NRC's review and approval is accomplished through the license amendment process in accordance with §§ 50.90 through 50.92 or as part of the application process for a design certification under part 52 or an original license application under part 50 or part 52.

## IV. Opportunities for Public Participation

### A. Technical Basis for Rulemaking.

As discussed in Section I, "Background," the NRC publicly released the technical basis information for this rulemaking in RIL-0801 on May 30, 2008, and NUREG/CR-6967 on July 31, 2008. Also on July 31, 2008, the NRC published in the *Federal Register* a notice of availability of the RIL and NUREG/CR-6967, together with a request for comments (73 FR 44778). In that notice, the NRC stated that these documents and comments on the documents would be discussed at a public workshop to be scheduled in September 2008. The public workshop was held on September 24, 2008, and included presentations and open discussion between representatives of the NRC, international regulatory and research agencies, domestic and international commercial power plants, fuel vendors, and the general public. A summary of the workshop, including a list of attendees and presentations, is available in ADAMS under Accession No. ML083010496. The NRC has not prepared responses to comments received on the technical basis information as a result of the July 31, 2008, *Federal Register* notice (including comments received at the September 2008 public workshop), because: 1) the public workshop was held, in part, to discuss public comments on the technical basis information, and 2) further opportunity to comment was available during the proposed rule's formal public comment period.

**B. Advance Notice of Proposed Rulemaking.**

On August 13, 2009, the NRC published an Advance Notice of Proposed Rulemaking (ANPR) (74 FR 40767) to obtain stakeholder views on issues associated with amending § 50.46(b). The ANPR indicated that the proposed scope of the rulemaking included four major objectives: 1) expand the applicability of § 50.46 to include any light-water reactor fuel cladding material, 2) establish performance-based requirements and acceptance criteria specific to zirconium-based cladding materials that reflect research findings, 3) revise the LOCA reporting requirements, and 4) address the issues raised in PRM-50-84 that relate to crud deposits and hydrogen content in fuel cladding. The ANPR provided interested stakeholders an opportunity to comment on the options under consideration by the NRC during a 75-day public comment period. In addition, the NRC asked 12 specific questions in the following categories: Applicability Considerations, New Embrittlement Criteria Considerations, Testing Considerations, Revised Reporting Requirements Considerations, Crud Analysis Considerations, and Cost Considerations. The public comment period ended on October 27, 2009.

The NRC received a total of 19 comment letters during the ANPR's public comment period; these letters were sent from a variety of entities, including one comment from a private citizen, 15 comments from the nuclear industry, one comment from a non-governmental organization, and two comments from the international community. The NRC held a public meeting on April 28-29, 2010, to discuss, among other things, the public comments received on the ANPR. No additional public comments were accepted at this public meeting. The meeting summary is available in ADAMS under Accession No. ML101300490.

The NRC considered the comments when preparing the proposed rule. A detailed discussion of the public comments submitted on the ANPR, including a detailed list of commenters, is contained in a separate document, "Section 50.46c and PRM-50-71 Comment Response Document" (ADAMS Accession No. ML12283A213).

**C. Proposed Rulemaking.**

On March 24, 2014, the NRC published the proposed rule and associated draft regulatory guides, DG-1261, DG-1262, and DG-1263, for comment (79 FR 16106). The proposed rule indicated that the proposed scope of the rulemaking included five major objectives: 1) expand the applicability of § 50.46 to include any LWR fuel cladding material; 2) establish performance-based requirements and acceptance criteria specific to zirconium-based cladding materials that reflect research findings; 3) incorporate recent research findings associated with embrittlement mechanisms; 4) address the issues raised in PRM-50-84 that relate to crud deposits and hydrogen content in fuel cladding; and 5) allow licensees to use an alternative risk-informed approach to evaluate the effects of debris for long-term cooling. The public comment period provided interested stakeholders an opportunity to comment on the draft preliminary provisions under consideration by the NRC during a 150-day public comment period. In addition, the NRC asked 12 specific questions in the following categories: Fuel Performance Criteria, Risk-Informed Alternative to Address the Effects of Debris, Implementation, Restructuring 10 CFR Chapter I, and Cumulative Effects of Regulation. In response to multiple extension requests, the NRC extended the public comment, which closed on August 21, 2015. Discussion of the public comments are found in Section V, "Public Comment Analysis."

On April 20, 2015, the NRC published an additional DG for comment, DG-1322 (75 FR 21658) for a 75-day comment period. This DG was associated with the alternative risk-informed approach for addressing the effects of debris on long-term cooling. The NRC considered public comments received on this DG while developing the final regulatory guide. Further discussion of this regulatory guide is found in Section XVIII, "Availability of Guidance," and a reference to the NRC's detailed comment responses for DG-1322 is found in Section XIX, "Availability of Documents."

During the public comment period for the rule, the NRC hosted several public meetings to facilitate development of public comments. On April 29-30, 2014, the NRC conducted a public meeting to provide an overview of the proposed rule. On June 24-26, 2014, the NRC conducted an additional public meeting to further discuss the proposed rule and draft regulatory guides. During this meeting, members of the nuclear industry provided presentations on various aspects of the rule. Additionally, Mr. Mark Leyse provided a presentation related to PRM-50-84, which the proposed rule addressed. On July 23, 2014, the NRC conducted a meeting to discuss topics specific to BWRs. During this meeting, members of the BWROG provided a presentation. References to the summaries for these public meetings can be found in Section XIX, "Availability of Documents."

### D. Public Meetings after Close of Public Comment Period.

After the closure of the formal period for submission of public comments on the proposed rule, the NRC conducted several additional public meetings to clarify comments received, continue dialogue on key areas with stakeholders, and facilitate development of the final rule. No additional public comments were accepted at these public meetings.

On March 17-19, 2015, the NRC conducted a public meeting to seek clarification regarding comments previously received on implementation and the regulatory analysis associated with the proposed rule. As a result of this public meeting, the NRC held a series of three follow-on public meetings to further discuss a draft preliminary implementation plan that would represent an alternative to that in the proposed rule. These follow-on meetings were held on April 23, 2015, May 7, 2015, and June 4, 2015. Additionally, on April 29-30, 2015, the NRC conducted a public meeting at Oak Ridge National Laboratory to discuss specific comments received on draft regulatory guides (DG) DG-1261, "Conducting Periodic Testing for Breakaway

Oxidation Behavior;" DG-1262, "Testing for Post Quench Ductility;" and DG-1263, "Establishing Analytical Limits for Zirconium-Based Alloy Cladding." On June 9, 2015, the NRC conducted a public meeting to discuss the long-term cooling provision in the proposed rule. References to the summaries for these public meetings can be found in Section XIX "Availability of Documents."

**E. Interactions with the Advisory Committee on Reactor Safeguards**

Since 2002, the NRC has met with the Advisory Committee on Reactor Safeguards (ACRS) multiple times to discuss the progress of the LOCA research program and rulemaking proposals. Provided in the following table are the dates and ADAMS accession numbers of the relevant ACRS meetings and associated correspondence.

Table 2 – ACRS Interactions Associated with 10 CFR 50.46c Rulemaking

| Date | Meeting/Letter | ADAMS |
|---|---|---|
| October 9, 2002 | Subcommittee Meeting | ML023030246* |
| October 10, 2002 | Full Committee Meeting | ML022980190* |
| October 17, 2002 | Letter from ACRS to NRC staff | ML022960640 |
| December 9, 2002 | Response letter from NRC staff to ACRS | ML023260357 |
| September 29, 2003 | Subcommittee Meeting | ML032940296* |
| July 27, 2005 | Subcommittee Meeting | ML052230093* |
| September 8, 2005 | Full Committee Meeting | ML052710235* |
| January 19, 2007 | Subcommittee Meeting | ML070390301* |
| February 2, 2007 | Full Committee Meeting | ML070430485 |
| May 23, 2007 | Letter from ACRS to NRC Staff | ML071430639 |
| July 11, 2007 | Response letter from NRC staff to ACRS | ML071640115 |
| December 2, 2008 | Subcommittee Meeting | ML083520501*<br>ML083530449* |
| December 4, 2008 | Full Committee Meeting | ML083540616* |
| December 18, 2008 | Letter from ACRS to NRC staff | ML083460310 |
| January 23, 2009 | Response letter from NRC staff to ACRS | ML083640532 |
| May 10, 2011 | Subcommittee Meeting | ML111450409 |
| June 8, 2011 | Full Committee Meeting | ML11166A181 |
| June 22, 2011 | Letter from ACRS to NRC staff | ML11164A048 |
| June 23, 2011 | Subcommittee Meeting | ML11193A035 |
| July 13, 2011 | Full Committee Meeting | ML11221A059 |
| July 21, 2011 | Response letter from NRC staff to ACRS | ML111861706 |
| December 15, 2011 | Subcommittee Meeting | ML120100268 |
| January 19, 2012 | Full Committee Meeting | ML12032A048 |
| January 26, 2012 | Letter from ACRS to NRC Staff | ML12023A089 |
| February 17, 2012 | Response Letter from NRC staff to ACRS | ML120260893 |
| December 2, 2014 | Subcommittee Meeting | ML14351A368 |
| November X, 2015 | Subcommittee Meeting | MLXXXXX |
| December X, 2015 | Full Committee Meeting | MLXXXXX |

*ADAMS file is a transcript of the ACRS meeting.

## V. Public Comment Analysis.

During the public comment period for the proposed rule, the NRC received 36 comment submissions from 23 commenters. In addition, the NRC received multiple requests to extend the public comment period. These extension requests were from Ms. Gretel Johnston, Ms. Ruth Thomas, and NEI, which was endorsed by Indiana Michigan Power. The NRC accommodated these requests by providing a 75-day extension to the public comment period. Of the 36 comment submissions, 17 letters were from private citizens and 19 were from the nuclear industry. This section describes the general comments received in response to the NRC's specific questions provided in the FRN for the proposed rule. In addition, the NRC has prepared a separate document that provides a detailed comprehensive breakdown and analysis of the comments. This document may be accessed as described in Section XIX, "Availability of Documents."

NRC Question 1. *Performance-Based Peak Cladding Temperature Limit.* The NRC requested comment on the retention of the prescriptive peak cladding temperature (PCT) criterion, rather than adopt a performance-based requirement. The NRC specifically asked whether established test procedures were available for demonstrating high temperature cladding performance. The NRC received specific responses regarding NRC Question 1 from AREVA, GEH, NEI, and Westinghouse Electric Company (WEC). All comments received on Question 1 expressed views that maintaining the existing prescriptive limit was inconsistent with the objective of a performance-based rule. Many industry commenters suggested that the PCT limit be moved to regulatory guidance, allowing applicants wishing to request PCT limits greater than 2200 °F to address relevant degradation mechanisms. NEI noted that at present, there are no industry plans to immediately seek a PCT greater than 2200 °F. No commenter identified test procedures that could be used to demonstrate high temperature cladding performance. The NRC's position is that there is not an adequate technical basis to extend peak cladding temperature beyond 2200 °F. Further, the NRC does not believe there is a strong demand from licensees to outline a methodology to extend peak cladding temperature beyond 2200 °F. The NRC did not change the rule as a result of these comments.

NRC Question 2. *Periodic Breakaway Testing*. The NRC requested comment on the type of data that should be reported and the required frequency of testing for breakaway oxidation. The NRC expressed that the objective of periodic testing was to prevent affected fuel from being loaded into a reactor, without adding ineffective requirements or unnecessary burden. The NRC received specific responses regarding NRC Question 2 from AREVA, GEH, NEI, and WEC. The industry commenters generally expressed that the sample frequency should be reduced and be more flexible. The industry commenters also expressed that requiring licensees to report breakaway oxidation results was unnecessary and that the fuel cladding vendors should address the concerns regarding breakaway oxidation with their quality assurance programs. A few fuel cladding vendors proposed that periodic test program plans could be developed by the fuel cladding vendors and approved by the NRC. The NRC agreed that the periodic testing and reporting requirements could be revised in a way that adds flexibility, decreases cost and burden of breakaway oxidation testing, and still achieves the safety objective. The NRC agreed that the objective of the rule could be achieved with rule language that requires a fuel vendor to submit breakaway oxidation testing program for NRC review and approval and that the requirement for licensees to report breakaway oxidation results could be removed. The breakaway oxidation testing program would be outlined in the as part of the documentation supporting the staff's review and approval of the new fuel design or as part of the topical report that details the intial breakaway oxidation behavior characterization for alloys already approved. The NRC changed the rule and associated regulatory guidance accordingly as a result of these comments.

NRC Question 3. *Analytical Long-Term Peak Cladding Temperature Limit*. The NRC sought comment on the proposed requirement in § 50.46c(g)(1)(v) of the proposed rule, which stipulated that a long-term cooling PCT analytical limit be established, which preserved cladding ductility based upon an NRC approved test program. Specifically, the NRC requested input regarding this new performance requirement to determine whether: 1) cladding ductility was the most suitable performance-based metric, 2) peak cladding temperature was the most suitable analytical limit, and 3) a technical basis existed for long-term cladding performance. No commenter supported the proposed new requirement. Several commenters questioned whether cladding ductility was the most appropriate performance-based metric. These commenters noted that different cladding degradation mechanisms may exist at different post-quench temperature regimes. Several commenters questioned the use of a single analytical limit on PCT noting that time-at-temperature may be more appropriate to capture the degradation mechanisms. No commenter identified an existing technical basis for long-term, post-quench fuel performance. Several commenters requested that the existing § 50.46 rule language be maintained.

As a result of the comments received, the NRC decided to revise the LTC performance requirement. Specifically, if debris considerations prompt a post-quench reheat transient, then the applicant would need to demonstrate, using an NRC approved analytical limit and experimental procedure, that no further cladding failure is predicted to occur. See Section III.C, "Fuel-Specific Performance and Analytical Requirements," for further details.

NRC Question 4. *Acceptance Criteria for Risk-Informed Alternative.* The NRC sought comment on whether the detailed acceptance criteria should be set forth in § 50.46c, or in the associated regulatory guidance. All commenters on this question stated that the detailed acceptance criteria should be in a RG and not the rule. Several commenters said that the guidance should refer to and be consistent with RG 1.174. One commenter suggested that industry should develop a guidance document for NRC endorsement and said that the guidance should be piloted before final rulemaking.

The NRC position is that the rule is written at a high level and that the details are contained in the associated regulatory guide. As no industry guidance document was received by NRC for possible endorsement, this suggestion has not been adopted. A pilot application of the alternative risk-informed approach was underway during this rulemaking activity and greatly influenced the development of the associated regulatory guide. The NRC did not change the rule due to these comments.

NRC Question 5. *Regulatory Approach for Risk-Informed Regulation*. The NRC sought comment on whether the risk-informed alternative offered by this regulation should require meeting numeric-risk acceptance criteria as a matter of compliance (similar to § 50.48c); or whether other risk-informed approaches that use risk-importance insights to establish measurable criteria or performance objectives, such as those in use by §§ 50.62, 50.63, and 50.65; or approaches using both risk importance and numeric-risk acceptance criteria, such as those in use by § 50.69, would be preferable. All of those commenting agreed that a "single selection" approach should not be specified; but rather, the rule should allow use of CDF and LERF, importance measures, or some other metrics. All stated that the rule should contain performance objectives rather than meeting specific numeric risk acceptance criteria. Two commenters suggested that § 50.48(c) was a model for how this rule should be written. One commenter said the guidance, including the risk acceptance criteria, should be piloted before final rulemaking.

The NRC position is that numeric risk acceptance criteria for CDF and LERF are consistent with the current approach to risk-informed regulation as set forth in RG 1.174. These metrics are approved surrogates for the Commission's quantitative health objectives. The NRC did not find adequate basis in the comment submissions to justify the use of risk importance measures or other metrics in lieu of the accepted surrogates (CDF and LERF). The commenters did not provide examples of performance objectives for staff consideration. The NRC notes that the § 50.46c section on the alternative risk informed approach contains less detail than § 50.48(c) but is consistent in terms of the risk-informed metrics, maintenance of defense-in-depth and safety margins, and other aspects. A separate pilot application of the risk acceptance criteria is not necessary, as a pilot application was underway during this rulemaking activity and because the risk-informed approach in the rule is consistent with RG 1.174 and past use of risk information. The NRC did not change the rule due to these comments.

NRC Question 6. *Operational Modes Considered in Risk-Informed Alternative*. The

NRC sought comment on whether the risk-informed approach provided in § 50.46(e) could generically exclude any plant operational modes (e.g., low power or shutdown) from consideration. If so, an operational mode was suggested to be excluded from the risk-informed approach, the NRC also sought the bases for excluding these operational modes from consideration. Two commenters stated that the risk assessment should be limited to modes where Technical Specifications require recirculation and should be limited to design-basis accidents, not severe accidents. Two other commenters stated that the modes should be limited to those where high pressure jets could result in debris sufficient to impact the recirculation function. One commenter noted that there may be, on a case-by-case basis, some modes other than at-power where high pressure jets cold result in a debris issue, and noted the need to pilot the guidance on the alternate risk-informed approach.

The NRC position is that the risk assessment must consider all hazards and all operating modes. Some hazards and certain plant operating modes may be screened from further consideration using approaches as set forth in the American Society for Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA Standard as endorsed in RG 1.200. Restricting the plant operating mode based on Technical Specifications or limiting the analysis to design-bases accidents is inconsistent with the risk-informed approach, which considers the risk from severe accidents that are beyond the design basis. In addition, the NRC does not believe an additional pilot effort is necessary, as the regulatory guide was developed using insights from the pilot application underway during the rulemaking activity. The NRC did not change the rule as a result of these comments.

NRC Question 7. *Reporting Criteria for the Risk-Informed Alternative*. The NRC sought specific comment on the reporting criteria for the risk-informed approach. Alternatively, the NRC also sought comment on whether the reporting criteria for the risk-informed approach should be more prescriptive and establish requirements similar to those for the ECCS model (i.e., § 50.46c(m)(1) through (m)(3)). The commenters generally stated that the rule should be very high level regarding monitoring and reporting, and that the details should be in a regulatory guide. By "high level," some commenters stated that the rule might only contain a requirement for a monitoring program and reporting, but no specific requirements. Two commenters stated that the monitoring and reporting should be plant-specific – each licensee would propose its program in the license amendment request and NRC would review and approve. These commenters suggested that thresholds for reporting would be graded – plants with higher risk could have more stringent reporting requirements than lower-risk plants, for example. One commenter said that corrective action and reporting should be no different than for current risk-informed applications. Regarding PRA updates, a commenter said that these should be event-triggered rather than calendar-based. In other words, PRA update would result from plant or procedural modifications independent of the time since the last update. A commenter stated that the guidance on monitoring and reporting should be reviewed by industry and piloted.

The NRC position is that the rule must contain requirements for performance monitoring, reporting, and corrective action. However, the NRC's intent was to write the rule at a high level and to include detailed guidance in the regulatory guides. The reporting and corrective action criteria in the rule have been modified and reorganized based on comments regarding the complexity of the proposed rule. In addition, the NRC does not believe an additional pilot effort is necessary, as the regulatory guide was developed using insights from the pilot application underway during the rulemaking activity

NRC Question 8. *Exemptions Needed to Implement the Risk-Informed Alternative.* The NRC sought input on whether conforming changes to other regulations would be necessary or desirable. Three commenters stated that they had identified no additional regulations for which exemption requests are expected to be necessary to support the implementation of the alternate risk-informed approach for addressing the effects of debris on core cooling. One commenter questioned whether an exemption from GDC 19, “Control Room,” would be needed, as the other regulatory criteria are dependent upon success of the long-term recirculation criteria. There were other commenters who provided comments on this concept.

*[Placeholder for summary of NRC response]*

NRC Question 9. *Staged Implementation*. The NRC sought comment on the staged implementation plan in § 50.46c(g)(1)(v) of the proposed rule, which divided the existing fleet among three implementation tracks based upon existing margin to the revised requirements and anticipated level of effort to demonstrate compliance. No commenter supported the Table 1 plant-specific compliance dates from the proposed rule. Several commenters identified that ongoing licensing activities, planned plant modifications, potential fuel vendor and/or fuel cladding alloy changes, and other plant-specific activities may necessitate exemption requests due to the Table 1 assignments. A series of public workshops and webinars were held in 2015 to improve the implementation plan. The industry provided a comprehensive, integrated schedule to illustrate the magnitude of effort and parallel and series work activities. This information informed a revision to the rule language. The rule was modified and Table 1 was removed and replaced with a requirement for licensees to submit an implementation plan within

6 months. Schedule requirements were established within the rule for: 1) submitting a license amendment request documenting compliance, and 2) complying with the rule. The revised implementation plan is based upon an alternative implementation plan provided by several commenters.

NRC Question 10. *New Reactor Implementation*. The NRC received specific responses regarding Question 10 from three submissions by the nuclear industry. The commenters raised concerns about new reactors potentially loading fuel and coming on-line while being required to comply with the new rule at a faster pace than the operating reactor fleet. The NRC has rewritten § 50.46c(o)(9) of the proposed rule to place the combined license holders on a time schedule for compliance that is consistent with the approach taken for the operating fleet. That is, a new reactor that is in the startup sequence aligned with the effective date of the rule must now comply with the requirements of the rule by the initial fuel loading or 84 months from the effective date of the rule, whichever is later. This change to the proposed rule makes the compliance timing of a new reactor consistent with that of the operating reactors.

NRC Question 11. *Re-structuring 10 CFR Chapter I with respect to ECCS Regulations*. The NRC sought comment on the administrative changes of restructuring 10 CFR chapter I with respect to ECCS regulations. In response to this request, the NRC received specific responses from three submissions by the nuclear industry. All of these comments stated that the industry agrees with the NRC that there will be large costs to revise industry documentation to reflect the proposed restructuring discussed in this question. These costs include the complete renumbering of many licensing basis documents. The benefits of the proposed restructuring to be small or non-existent from the standpoint of safety. The comments stated that, in light of the lack of perceived safety or other benefits of significance, the industry did not develop a cost estimate. The NRC did not change the rule as a result of these comments.

NRC Question 12. *Cumulative Effects of Regulation.* The NRC sought comment on the rule's implementation schedule in light of any existing cumulative effect of regulation challenges. In response to this request, the NRC received comments from three submissions by the nuclear industry. The comments generally stated that the industry is concerned with the adequacy of industry resources that will be required to perform the work activities resulting from the rule. Further, comments stated that the rule's effective date, compliance date, and submittal dates do not provide sufficient time to implement the new requirements. To cope with the aforementioned challenges, comments suggested greater schedule flexibility. As a result of these public comments, the NRC hosted multiple public meetings in 2015 to discuss comments on implementation of the rule. The industry and NRC staff discussed the most effective and efficient means to implement § 50.46c. The industry provided a comprehensive, integrated schedule to illustrate the magnitude of effort and parallel and series work activities. This information informed a revision to the rule language. The revised implementation approach provides flexibility to address any plant-specific issues, such as pursuing advanced cladding. The rule was modified with a requirement for licensees to submit an implementation plan within 180 days. Schedule requirements will be established within the rule for: 1) submitting a license amendment request documenting compliance, and 2) complying with the rule.

## VI. Section-by-Section Analysis.

The organization and 10 CFR designations of the NRC's requirements governing emergency core cooling (currently in § 50.46) and reactor cooling venting systems (currently in § 50.46a) will change in two steps. These are presented in Table 3. The NRC believes implementing the changes in two steps minimizes licensee revision of their internal documents to reflect the implementation of the final rule, as established in paragraph (p) of the final rule.

Table 3 – Reorganization of 10 CFR 50.46

| **Existing NRC Requirements and Final New Regulations (Bolded rules are currently in effect)** | **Rulemaking and Implementation Activities** | |
|---|---|---|
| | **Initial Codification of Final Performance-Based Fuel Cladding Requirements** | **End of Phased Implementation Period for Performance-Based Cladding Requirements** |
| **§ 50.46 ECCS** | § 50.46 ECCS Acceptance Criteria (*existing deterministic requirements remain in effect*) | Removed<br>*(Administrative rulemaking would:*<br>▪ *remove deterministic requirements in (existing) § 50.46*<br>▪ *redesignate § 50.46c as § 50.46.* |
| **§ 50.46a Reactor Coolant Venting Systems** | Maintain as § 50.46a | *Maintain as § 50.46a* |
| **§ 50.46c final rule** | § 50.46c Performance During LOCAs | Re-designated as § 50.46<br>*(Administrative rulemaking would:*<br>*i) remove superseded fuel cladding requirements in § 50.46, and*<br>*ii) redesignate § 50.46c as § 50.46.)* |

In "SRM-SECY-10-0161- 'Final Rule: Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements (10 CFR 50.46a)'," dated April 26, 2012 (ADAMS Accession No. ML12117A121), the Commission approved the NRC staff's request to withdraw SECY-10-0161, "Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements (10 CFR 50.46a)," from Commission consideration (ADAMS Accession No. ML121500380). The Commission also directed the staff, as part of its May 19, 2014 SRM (ADAMS Accession No. ML14139A104) on the staff's recommendation on Near Term Task Force (NTTF)

Recommendation 1 (SECY-13-0132, ADAMS Accession No. ML13277A413) to re-evaluate the staff's proposed Improvement Activities 1 and 2 for Recommendation 1 (i.e., beyond-design-basis events extension and defense-in-depth) in the staff's Risk Management Regulatory Framework (RMRF) activities. Following Commission direction on RMRF, the staff will re-evaluate its plans for whether and how/when to resubmit § 50.46a. If the staff decides to recommend Commission approval of a final § 50.46a rule, the *Federal Register* notice of final rulemaking for that rule will include an updated Table 2.

**A. Section 50.46c – Heading**

A new section, § 50.46c, is created in 10 CFR part 50 by this rulemaking. The heading of § 50.46c is "Emergency core cooling system performance during loss-of-coolant accidents."

**B. Section 50.46c(a) – Applicability.**

Paragraph (a) defines the applicability of the final rule, which remains limited to LWRs, but is expanded beyond fuel designs consisting of uranium oxide pellets within cylindrical zircaloy or ZIRLO™ cladding. The rule is applicable to applicants for and holders of construction permits, operating licenses, combined licenses, and standard design approvals, and also to applicants for standard design certifications and for manufacturing licenses (including applicants for renewal of standard design certifications and manufacturing licenses. The NRC notes that standard design approvals may not be rewed, as provided for in 10 CFR 52.147. As stated above, however, it is not the NRC's intent that the alternate risk-informed approach of paragraph (e) be used to justify the use of problematic insulation or other debris sources in new designs, including new reactors, or plant modifications made by licensees. Similar to GSI-191, the NRC intends to approve use of the alternative risk-informed approach only in cases where an issue emerges that could not have been readily foreseen during the design or implementation processes and where removal of such debris could pose an undue burden not justified by the risk posed by the debris.

**C. Section 50.46c(b) – Definitions.**

Paragraph (b) provides definitions for terms used in this section. The definitions of *Loss-of-coolant accident* and *Evaluation model (re-titled as "ECCS evaluation model")* remain unchanged from those currently located in § 50.46(c)(1) and (c)(2), respectively. Definition of *Breakaway oxidation, Cladding, and Crud* are added.

**D. Section 50.46c(c) – Relationship to Other NRC Regulations.**

Paragraph (c) identifies other regulations which relate, in a direct or indict manner, to § 50.46c. The regulations identified in paragraph (c) remain largely unchanged from that of the existing § 50.46(d). However, the discussion is revised to make clear that an approach approved by the NRC under § 50.46c(e) may also be used when evaluating the effects of debris to demonstrate compliance with other requirements of this part, including GDC-35, GDC-38, and GDC-41 (as allowed by § 50.46c and requested in the application).

**E. Section 50.46c(d) – Emergency Core Cooling System Performance.**

Paragraph (d)(1) defines performance-based requirements for the ECCS.

Paragraph (d)(2) requires that ECCS performance be demonstrated using an acceptable ECCS evaluation model meeting specific requirements for a range of postulated LOCAs of different sizes, locations, and other properties, sufficient to provide assurance that the most severe postulated LOCA has been identified. The provisions for a realistic ECCS model or appendix K to 10 CFR part 50 model remains unchanged from the existing regulation found in § 50.46(a)(1)(i) and (ii), respectively. Similarly, the model requirement that calculated changes in core geometry must be addressed remains unchanged from the existing regulation found in § 50.46(b)(4). Paragraph (d)(2)(iii) explicitly requires that the ECCS evaluation model address calculated changes in core geometry, and consider factors that may alter localized coolant flow or inhibit delivery of coolant to the core. Demonstration of ECCS performance is expected to consider inhibition of core flow that can result from such factors as, but not limited to, pump damage, piping damage, boron precipitation, and deposition of debris and/or chemicals associated with the long-term cooling mode of recirculation from the long-term water source (e.g. reactor building sump, torus). Consideration of debris and/or chemical deposition is already required by the existing rule, and the new rule does not alter the current efforts to address such factors under programs such as GSI-191. Demonstration of consideration of such factors may also be achieved through analytical models that adequately represent the empirical data obtained regarding debris deposition. The final rule alternatively allows the use of risk-informed approaches to evaluate the effects of debris on localized coolant flow and delivery of coolant to the core during the long-term cooling (post-accident recovery) period.

In addition, paragraph (d)(2)(iv) of the rule specifically requires that ECCS performance be demonstrated for both the accident and the post-accident recovery and recirculation period.

Paragraph (d)(2)(v) requires that the ECCS model address the fuel system modeling requirements in paragraph (g)(2) if the reactor uses uranium oxide or mixed uranium-plutonium oxide pellets within zirconium cladding (e.g., currently operating reactors).

Paragraph (d)(3) provides the ECCS evaluation model documentation requirements currently provided in appendix K, Section II, "Required Documentation."

**F. Section 50.46c(e) – Alternate Risk-Informed Approach for Addressing the Effects of Debris on Long-Term Core Cooling.**

Paragraphs (d)(2)(iii) and (e) allow entities to use a risk-informed approach for addressing the effects of debris on long-term core cooling. Paragraphs (e)(1)(i) through (e)(1)(v) provide the attributes of an acceptable alternative risk-informed approach for addressing the effects of debris on long-term core cooling and establishes minimum requirements for the systematic risk assessment, including the PRA, other methods and techniques, and other supporting engineering analyses, calculations, and tests. These requirements are intended to ensure that the implementation of the alternate risk-informed approach to address debris effects on long-term cooling demonstrates that any resulting increase in CDF and LERF will be small, that sufficient defense-in-depth and safety margins are maintained, and that the validity of the risk-informed approach will be maintained. These requirements are consistent with the key principles of risk-informed decisionmaking described in RG 1.174.

Paragraph (e)(1)(i) of the rule requires demonstration that any potential risk increase be small. Paragraph (e)(1)(ii) requires that sufficient defense-in-depth and safety margins be maintained. Paragraph (e)(1)(iii) requires the use of an internal-events, at-power PRA, and allows other methods and techniques to address other hazards and plant operational modes, and recognizes that there are supporting engineering calculations and analyses, tests and other information used in the systematic process. Collectively, the elements in paragraph (e)(1)(iii) comprise the systematic risk assessment. Paragraph (e)(1)(iv) requires the systematic risk assessment to have acceptable scope, level of detail, and technical adequacy and be performed under a quality assurance (QA) program. Paragraph (e)(1)(v) requires that a monitoring program be established. Paragraph (e)(1)(vi) requires the systematic risk assessment to be updated at least every four years after being approved by the NRC and requires a specific updated by a combined license holder prior to initial fuel load.

Paragraph (e)(2) requires those entities seeking to use the alternative risk-informed approach under paragraph (e)(1) to submit an application that contains the information provided in paragraphs (e)(2)(i) through (e)(2)(v).

Paragraph (e)(2)(i) requires entities to provide a description of the alternative risk-informed approach. The RG for the risk-informed alternative approach provides one acceptable approach that can be referenced in describing this approach.

Paragraph (e)(2)(ii) requires that initiating events from sources both internal and external to the plant and for all modes of operation, including low power and shutdown modes, be considered when evaluating the effects of debris on long-term core cooling using the alternate approach. This aspect of the rule recognizes that the minimum PRA that would be required by Paragraph (e)(1)(iii) may not address all sources of initiating events and modes of operations; and as such, other approaches may be used as indicated in Paragraph (e)(1)(iii)(B). Therefore, the application would need to describe the measures taken to assure the scope, level of detail, and technical adequacy of all the analyses (i.e., the systematic risk assessment) performed to address severe accidents are sufficient for this application and address the full spectrum of initiating events and modes of operation. This includes the results of the PRA review process for those aspects addressed by a PRA. This specific aspect includes such items as any peer reviews performed, any actions taken to address peer review findings, and any efforts to compare the plant-specific PRA to the ASME/ANS PRA standard, as endorsed by the NRC in the latest version of RG 1.200.

In paragraph (e)(2)(iii), the entity is required to include information about the evaluations they performed to demonstrate that acceptance criteria in (e)(1)(i) and the requirements in (e)(1)(ii) are met. This description includes the determination that any increase in risk is small and that sufficient defense-in-depth and safety margins are maintained. The entity is required to provide sufficient information to the NRC, describing the evaluations and the basis for their acceptability, to demonstrate compliance with the requirements in this rule.

Paragraph (e)(3) provides that the NRC may approve an application to implement the alternative risk-informed approach if it determines that the proposed approach satisfies the requirements of paragraph (e)(1). The NRC staff would review the description of the alternative risk-informed approach set forth in the application, and the associated evaluations, to confirm that it contains the elements required by the rule. The NRC staff would also review the information provided about the design/plant-specific PRA and other systematic evaluations used to evaluate severe accidents in support of the application to assure that the scope, level of detail, and technical adequacy of the analyses are commensurate with the reliance on the risk information. This aspect of the review would involve the NRC assessment of the information provided about: 1) the peer review process to which the design/plant-specific PRA was subjected, 2) the reliance on other systematic evaluations to address areas not covered by the design/plant-specific PRA, and 3) the approach for demonstrating sufficient defense-in-depth and safety margins are maintained. The NRC's approval of the use of the risk-informed approach to address long-term cooling would specify the circumstances under which the entity would be required to notify and seek additional approval by the NRC of changes or errors in the risk-informed approach. Depending upon the nature of the underlying application (e.g., existing license, operating or combined license application, design certification application, or design approval), the notification and approval requirement will be implemented through a license condition, a provision in the design certification rule, or a condition of the design approval, as applicable.

Paragraph (f) is reserved for future amendments to § 50.46c.

**G. Section 50.46c(g) – Fuel System Designs: Uranium Oxide or Mixed Uranium-Plutonium Oxide Pellets Within Cylindrical Zirconium-Alloy Cladding.**

This section is added to set forth fuel design specific analytical limits and performance-based requirements by which to judge the overall ECCS performance in accordance with paragraph (d)(1) for LWRs using uranium oxide or mixed uranium-plutonium oxide pellets within cylindrical zirconium alloy cladding. The fuel performance criteria in paragraph (g)(1) and fuel system modeling requirements in paragraph (g)(2) are based on the established degradation mechanisms and performance objectives for this specific fuel type.

Paragraph (g)(1)(i) establishes an analytical limit on peak cladding temperature to avoid cladding embrittlement, high temperature failure modes, and run-away exothermic oxidation. Except as calculated in paragraph (g)(1)(ii), the calculated maximum fuel element cladding temperature should not exceed 2200 °F. This requirement remains unchanged from the existing requirement at § 50.46(b)(1).

Paragraph (g)(1)(ii) requires that the zirconium alloy cladding maintains sufficient post-quench ductility in order to avoid gross brittle failure. This requirement replaces the existing prescriptive analytical limit, 17 percent ECR, in § 50.46(b)(2).

Paragraph (g)(1)(iii) is added added to establish a performance-based requirement to preclude breakaway oxidation in order to avoid cladding embrittlement and gross failure. Breakaway oxidation is a new requirement relative to § 50.46(b).

Paragraph (g)(1)(iv) establishes an analytical limit on maximum hydrogen generation to avoid an explosive concentration of hydrogen gas. This requirement is the same as that of the existing regulation in § 50.46(b)(3).

Paragraph (g)(1)(v) requires that the applicant or licensee demonstrate that effective core cooling is maintained for the long-term period required to remove decay heat. Under the language of the final rule, effective core cooling exists when the calculated core temperatures do not result in further cladding failure. This performance requirement is consistent with the existing requirement to “maintain the calculated core temperature at an acceptably low value” in § 50.46(b)(5).

Paragraph (g)(2) establishes fuel design specific modeling requirements that are needed in addition to the generic ECCS evaluation model requirements in paragraph (d)(2). Paragraph (g)(2)(i) requires consideration of oxygen diffusion from the cladding inside surface. This is a new ECCS evaluation model requirement.

Paragraph (g)(2)(ii) is added to include a requirement to evaluate the thermal effects of crud and oxide layers that may have accumulated on the fuel cladding during plant operation.

Paragraphs (h) through (j) are added to reserve rulemaking space for future amendments to § 50.46c, including any changes that stem from using newly designed fuel and cladding materials.

**H. Section 50.46c(k) – Use of NRC-Approved Fuel in Reactor.**

Paragraph (k) prohibits licensees from loading fuel into a reactor or operating the reactor unless the licensee either determines that the fuel meets the requirements in paragraph (d) or complies with technical specifications governing lead test assemblies in its license.

**I. Section 50.46c(l) – Authority to Impose Restrictions on Operation.**

Paragraph (l) provides that the Director of the Office of Nuclear Reactor Regulation or the Director of the Office of New Reactors may impose restrictions on reactor operation if it is found that the evaluations of ECCS cooling performance submitted are not consistent with the requirements of this section. The authority to impose restrictions is expanded, relative to the authority currently granted in § 50.46(a)(2), to address licenses issued under 10 CFR part 52.

**J. Section 50.46c(m) – Reporting, Corrective Actions, and Updates**

Paragraph (m) provides reporting requirements applicable to the ECCS evaluation model and reporting requirements applicable to entities that elect to use the risk-informed alternative to address the effects of debris on long-term cooling. Paragraphs (m)(1) and (m)(2) apply to all entities subject to § 50.46c; paragraphs (m)(6) through (m)(8) apply to those entities demonstrating acceptable long-term core cooling under the provisions of paragraph (e).

Paragraph (m)(1) establishes reporting requirements if an entity identifies any change to, or error in, an ECCS evaluation model or the application of such a model. For clarity, this paragraph was divided into three categories of changes or errors, each with its own reporting requirements. These requirements are unchanged from the existing § 50.46(a)(3), with the exception of conforming to analytical limits established in the rule.

Paragraph (m)(1)(i) establishes reporting requirements if an entity identifies any change to, or error in, an ECCS evaluation model or the application of such a model, that does not result in any predicted response that exceeds any acceptance criteria and is itself not significant.

Paragraph (m)(1)(ii) establishes reporting requirements if a licensee identifies any change to, or error in, an ECCS evaluation model or the application of such a model, that does not result in any predicted response that exceeds any acceptance criteria but is significant (as defined in paragraph (n)).

Paragraph (m)(1)(iii) establishes reporting requirements for an entity who identifies any change to, or error in, an ECCS evaluation model or the application of such a model, that results in any of the acceptance criteria to be exceeded.

Paragraph (m)(2)(i) establishes corrective action requirements if an entity identifies any change to, or error in, an ECCS evaluation model or the application of such a model, that results in any of the acceptance criteria to be exceeded.

Paragraph (m)(2)(ii) requires an amendment to a design certification application reflecting any reanalysis required by paragraph (m)(1)(iii) to be submitted by the applicant in concert with the reanalysis.

Paragraphs (m)(3) through (m)(5) are reserved for future rulemaking. Paragraph (m)(6) through (m)(8) establish requirements for reporting, corrective actions, and periodic updates for entities choosing to implement the alternative risk-informed approach for addressing the effects of debris on long-term core cooling. Paragraph (m)(6) specifics the reporting requirements specific to each entity. Paragraph (m)(7) specifies the corrective action requirements specific to each entity. Paragraph (m)(8) specifies the periodic update requirements specific to each entity.

**K. Section 50.46(n) – Significant change or error in the ECCS evaluation model.**

Paragraph (n) provides the definition of a significant change or error. For uranium and mixed uranium-plutonium oxide fuel within cylindrical zirconium-alloy cladding, the definition in Paragraph (n)(1) is expanded, relative to the 50 °F change in calculated PCT in the current § 50.46(a)(3)(i), to include a 0.4 percent ECR. Paragraph (n)(2) is reserved for future fuel types.

Paragraph (o) is added to reserve rulemaking space for future amendments to § 50.46c.

**L. Section 50.46(p) – Implementation.**

This section establishes the implementation requirements and schedule for the existing fleet and for new reactors. Paragraph (p)(1) requires construction permits under 10 CFR part 50 issued after the effective date of the rule to comply with the requirements of § 50.46c.

Paragraph (p)(2) requires each holder of an operating license issued under 10 CFR part 50 and each holder of an operating license issued under this part which is based upon a construction permit in effect as of the effective date of the rule (including deferred and reinstated construction permits) to submit an implementation plan and schedule for achieving compliance with the provisions of this regulation with the exception of the consideration of debris effects under paragraph (d)(2)(iii) of this section. The rule stipulates that compliance demonstration be in the form of a license amendment application under § 50.90 and must be submitted by no later than 60 months after the effective date of the rule. Furthermore, the rule requires licensees be in compliance with the requirements of this section no later than 84 months after the effective date of the rule.

Paragraph (p)(3) requires operating licenses under 10 CFR part 50 issued after the effective date of the rule to comply with the requirements of § 50.46c no later than 84 months after the effective date of the rule.

Paragraph (p)(4) requires standard design certifications, standard design approvals, and manufacturing licenses under 10 CFR part 52, whose applications (including applications for amendment) are docketed after the effective date of the rule (including branches of these certifications whose applications are docketed after the effective date of the rule), to comply with the provisions of the rule by the time of certifications. Applicants submitting after the rule has been adopted should have had ample time to develop and receive approval for the analysis methods necessary to comply with the provisions of the rule.

Paragraph (p)(5) requires standard design certifications under 10 CFR part 52 issued before the effective date of the rule to comply no later than the time of renewal of certification.

Similar to the requirements of paragraph (p)(5), such applicants will have had ample time necessary to comply with the provisions of the rule.

Paragraph (p)(6) requires standard design certifications, standard design approvals, and manufacturing licenses, along with new branches of certifications under 10 CFR part 52 whose applications are pending as of the effective date of the rule to comply with § 50.46c no later than the time of renewal. Applications for design certifications that are in the approval process at the time the rule becomes effective will be required to comply at the time of renewal in response to the renewal application. This will provide ample time to develop and receive approval for the methodologies necessary to comply with the rule.

Paragraph (p)(7) requires combined license applications under 10 CFR Part 52 that are docketed after the effective date of the rule to comply with the provisions of the rule.

Paragraph (p)(8) requires combined licenses under 10 CFR Part 52 issued before the effective date of the rule to comply with § 50.46c no later than initial fuel load or 84 months after the publication date of the final § 50.46c in the Federal Register. This provision applies to the combined licenses which the NRC has already issued for *Vogtle* Units 3 and 4, *V.C. Summer* Units 2 and 3, and *Fermi Unit 3*. It affords these licensees ample time to develop and submit the necessary methodologies to the NRC for approval. Paragraph (p)(8) also applies the same requirement to entities whose combined licenses are issued after the effective date of the rule but whose applications were docketed before the effective date of the final § 50.46c (*i.e.*, the 30 days after the publication date of the final rule in the *Federal Register*). For those combined license holders, compliance with the rule must be achieved no later than initial fuel load or 84 months after the publication date of the final § 50.46c in the *Federal Register*.

Entities that elect to use the voluntary alternative to the long-term cooling requirements of the rule using a risk-informed approach can do so in advance of the date for compliance with the other aspects of the rule. In this case, the entity has to receive NRC approval on its risk-informed submittal prior to using the risk-informed approach.

**M. Appendix K to Part 50 of Title 10 of the *Code of Federal* Regulations (10 CFR) ECCS Evaluation Models.**

In appendix K, a new paragraph II.6 is added to specify that, for those entities that have implemented § 50.46c, the requirements for documentation are located within § 50.46c(d)(3) rather than in paragraph II of appendix K.

**N. Redesignation of Venting Requirements in § 50.46a; reservation of § 50.46a.**

Existing § 50.46a, "Acceptance criteria for reactor coolant system venting systems," is redesignated as § 50.46b. Section 50.46a is reserved for future use.

**O. Conforming Changes Throughout 10 CFR Parts 50 and 52.**

Several changes are made throughout 10 CFR parts 50 and 52 in order to conform with the final rule and the redesignation of the venting requirements from existing § 50.46a to new § 50.46b. Section 50.8 is amended to add the final rule to the list of approved information collections. Where §§ 50.34(a)(4), 50.34(b)(4), 52.47(a)(4), 52.79(a)(5), 52.137(a)(4), and 52.157(f)(1) currently refer only to § 50.46, they are amended to refer to "§ 50.46 and, § 50.46c, as applicable." Where §§ 50.34(a)(4), 52.47(a)(4), 52.79(a)(5), 52.137(a)(4), and 52.157(f)(1) currently refer only to § 50.46a, they are amended to refer to § 50.46b.

The GDC-35, GDC-38, and GDC-41 in appendix A to 10 CFR part 50 are amended to expressly address the acceptability of using a risk-informed alternative for long-term cooling when demonstrating compliance with these regulations, as allowed by § 50.46c. However, each entity wishing to use the risk informed alternative for long term cooling when demonstrating compliance with any of the three GDCs must request NRC approval in the form of an application for a license amendment.

## VII. Regulatory Flexibility Certification.

Under the Regulatory Flexibility Act (5 U.S.C. 605(b)), the NRC certifies that this rule does not have a significant economic impact on a substantial number of small entities. This final rule affects only the licensing and operation of nuclear power plants. The companies that own these plants do not fall within the scope of the definition of “small entities" set forth in the Regulatory Flexibility Act or the size standards established by the NRC (10 CFR 2.810).

## VIII. Regulatory Analysis.

The NRC has prepared a final regulatory analysis on this regulation. The analysis examines the costs and benefits of the alternatives considered by the NRC. The regulatory analysis is available as indicated in the “Availability of Documents” section of this document.

## IX. Backfitting and Issue Finality.

**PRELIMINARY DRAFT**

***This is a placeholder. A preliminary draft version of this section may be made publicly available in support of the November 2, 2015, ACRS subcommittee meeting and placed on regulations.gov, Docket ID NRC-2008-0332.***

## X. Cumulative Effects of Regulation.

As discussed in Section IV, "Public Comment Analysis," the NRC provided a specific question in the FRN for the proposed rule regarding cumulative effects of regulation (CER). The NRC sought comment on the rule's implementation schedule in light of any existing CER challenges. In response to this request, the NRC received comments from the nuclear industry. The comments generally stated that the industry is concerned with the adequacy of industry resources that will be required to perform the work activities that would have resulted from the rule. Further, comments stated that the proposed rule's effective date, compliance date, and submittal dates did not provide sufficient time to implement the new requirements. To cope with the aforementioned challenges, comments suggested greater schedule flexibility. As a result of these public comments, the NRC hosted multiple public meetings in 2015 to discuss comments on implementation of the rule. The industry and NRC staff discussed the most effective and efficient means to implement § 50.46c. The industry provided a comprehensive, integrated schedule to illustrate the magnitude of effort and parallel and series work activities. This information informed a revision to the rule language. The revised implementation approach provides flexibility to address any plant-specific issues, such as pursuing advanced cladding. The rule was modified with a requirement for licensees to submit an implementation plan within 180 days. Schedule requirements will be established within the rule for: 1) submitting a license amendment request documenting compliance, and 2) deadline for compliance. This revised implementation approach provides greater flexibility while managing workload with consideration of available ECCS performance margin.

## XI. Plain Writing.

The Plain Writing Act of 2010 (Pub. L. 111-274) requires Federal agencies to write documents in a clear, concise, and well-organized manner. The NRC has written this document to be consistent with the Plain Writing Act as well as the Presidential Memorandum, "Plain Language in Government Writing," published June 10, 1998 (63 FR 31883).

## XII. Finding of No Significant Environmental Impact: Environmental Assessment.

The Commission has determined under the National Environmental Policy Act of 1969, as amended, and the Commission's regulations in subpart A of 10 CFR part 51, that this rule is not a major Federal action significantly affecting the quality of the human environment and, therefore, an environmental impact statement is not required. Further, initial implementation of these amendments require licensees, in some cases, to submit an additional license amendment. The NRC's consideration of these license amendments would each contain an environmental assessment of the proposed licensee-specific action. The basis for this determination is as follows:

<u>Identification of the Action</u>:

The action is the amendment of 10 CFR part 50 by adding a new § 50.46c which contains the NRC's requirements for ECCSs for LWRs (that are currently contained in § 50.46). The amendment establishes performance-based requirements and also accounts for the new research information, as discussed in Section II, "Background," of this document. This research identified previously unknown embrittlement mechanisms. The research indicated that the current combination of peak cladding temperature (2200 °F (1204 °C)) and local cladding oxidation criteria do not always ensure PQD. Further, the amendment expands the applicability of § 50.46 to all fuel design and fuel cladding materials. In addition, this rule addresses the issues raised in two PRMs (docketed as PRM-50-71 and PRM-50-84). The rule also contains a provision that allows licensees to use an alternative risk-informed approach to evaluate the effects of debris for long-term cooling.

<u>The Need for Action</u>:

The action is needed in response to recent research into the behavior of fuel cladding under LOCA conditions. This research, as discussed in Section II, "Background," of this document, indicated that the current combination of peak cladding temperature (2200 °F (1204 °C)) and local cladding oxidation criteria do not always ensure PQD. The research also identified previously unknown embrittlement mechanisms. The action replaces the limits on peak cladding temperature and local oxidation with specific cladding performance requirements and acceptance criteria that ensure that an adequate level of cladding ductility is maintained throughout the postulated LOCA.

The provision to expand applicability to all light-water nuclear power reactors, regardless of fuel design or cladding material used, allows for the development and use of cladding materials other than zircaloy and ZIRLO™. Under the existing § 50.46, licensees that use different types of cladding material were required to request NRC approval for an exemption from the rule, in accordance with § 50.12.

The rule requires licensees to take into account the deposition of crud on the fuel cladding during plant operation. This change addresses PRM-50-84.

The NRC identified the need for an approach that would allow entities to address the effects of debris on long-term cooling in a manner that would be more timely and cost-effective than the current use of deterministic methods.

Environmental Impacts of the Action:

This environmental assessment focuses on those aspects of the rulemaking through which the revised requirements could potentially affect the environment. The NRC has concluded that there will be no significant radiological environmental impacts associated with the implementation of the rule requirements for the following reasons:

1) The amendments to the ECCS requirements of § 50.46 are unrelated to the integrity of reactor coolant system piping whose sudden failure would initiate a LOCA. Therefore, the rule does not affect the probability of an accident.

2) The amendments to the 10 CFR part 50 ECCS requirements are unrelated to the physical make-up of the systems, structures, and components that mitigate the consequences of a LOCA. These amendments revise and expand the performance requirements for which the ECCS response is judged. With these enhancements, the reactor core remains coolable because, by addressing previously unknown degradation mechanisms, cladding ductility is

preserved following a postulated LOCA. Therefore, the consequences of a postulated LOCA are not adversely changed by the rule.

3) The amendments to the 10 CFR part 50 ECCS requirements do not impact a facility's release of radiological effluents during and following a postulated LOCA. Therefore, the rule does not affect the amount of effluent released as a result of a possible accident.

4) The rule allows an entity to address the effects of debris on long-term cooling using a risk-informed approach. The effects of debris are currently addressed using deterministic methods. Any change in CDF and LERF allowed by a risk-informed approach would be small and within criteria already established in the current RG 1.174 , for making risk-informed changes to plant licensing bases.

This rulemaking amends calculated ECCS evaluation models used to assess the emergency core cooling system's response to a postulated LOCA. The rulemaking does not affect any other procedures used to operate the plant, nor alter the plant's geometry or construction. Further, the amendments ensure post quench ductility and core coolability following a postulated LOCA, and as such, do not affect the dose to any plant workers following postulated accidents. Similarly, dose to any individual member of the public are not affected.

For the reasons discussed, the action does not significantly increase the probability or consequences of accidents, nor result in changes being made in the types of any effluents that may be released off-site, and there is no increase in occupational or public radiation exposure.

With regard to potential non-radiological impacts, the rule has no significant impact on the environment. The rule to revise and expand the ECCS performance requirements is applied by an NRC nuclear reactor power plant licensee to the restricted area of its facility only, and in many cases does not result in any physical changes to the plant. Restricted areas of nuclear power plants are industrial portions of the facility constructed upon previously disturbed land, to which access is limited to authorized personnel. As such, it is extremely unlikely that the amendments will create any significant impact on any aquatic or terrestrial habitat in the vicinity of the plant, or to any threatened, endangered, or protected species under the Endangered Species Act, or have any impacts to essential fish habitat covered by the Magnuson-Stevens Act. Similarly, it is extremely unlikely that there will be any impacts to socioeconomic, or to historic properties and cultural resources. Therefore, there are no significant non-radiological environmental impacts associated with the action.

Licensee compliance with the amendments require an additional license amendment. A National Environmental Policy Act analysis will be conducted for each licensee-specific license amendment review.

Alternatives to the Action:

As an alternative to the rulemakings previously described, the NRC considered not taking the action (i.e., the “no-action” alternative). Not revising the ECCS cladding acceptance criteria could result in instances, following a LOCA, in which cladding ductility is not guaranteed to be maintained. Under the no action alternative, licensees will continue to submit exemption requests for NRC approval of fuel cladding other than zircaloy or ZIRLO™.

The NRC does not find this alternative acceptable to preserving public health and safety. The revised requirements are necessary because recent research has indicated that the existing PCT and oxidation restrictions do not take into consideration newly discovered cladding embrittlement mechanisms, and that the existing restrictions may not always be adequate to ensure post quench ductility of fuel cladding. The revised requirements ensure post quench ductility and core coolability following a postulated LOCA.

The rule allows an entity to use a risk-informed approach to address the effects of debris for long-term cooling. An alternative to addressing debris using this risk-informed approach is to continue to address the effects of debris using deterministic methods and approved models, as described in SECY-12-0093, “Closure Options for Generic Safety Issue – 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance,” dated July 9, 2012 (ADAMS Accession No. ML121310648). However, the NRC has added the alternative approach to provide entities the additional flexibility to address the effects of debris on long-term cooling using risk-informed methodologies, which may be implemented in a more timely and cost-efficient manner.

Alternative Use of Resources:

This action does not involve the use of any resources not previously considered by the NRC in its past environmental statements for issuance of operating licenses for the facilities that will be affected by this action.

Agencies and Persons Consulted:

The NRC staff developed the final rule and this environmental assessment. In accordance with its stated policy, the NRC provided a copy of the final rule and the environmental assessment to designated State Liaison Officers and requested their comments.

No other agencies were consulted.

## XIII. Paperwork Reduction Act.

***This is a placeholder.***

**Public Protection Notification.**

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless the document requesting or requiring the collection displays a currently valid OMB control number.

## XIV. Congressional Review Act.

This final rule is rule as defined in the Congressional Review Act (5 U.S.C. §§ 801-808). However, the Office of Management and Budget has not found it to be a major rule as defined in the Congressional Review Act.

## XV. Criminal Penalties

For the purposes of Section 223 of the Atomic Energy Act of 1954, as amended (AEA), the NRC is issuing this final rule to amend §§ 50.8, 50.34, 50.46a, 50.46c, appendix A to 10 CFR part 50, appendix K to 10 CFR part 50, and §§ 52.47, 52.79, 52.137, and 52.157 under one or more of Sections 161b, 161i, or 161o of the AEA. Willful violations of the rule would be subject to criminal enforcement. Criminal penalties as they apply to regulations in 10 CFR part 50, are discussed in § 50.111.

## XVI. Agreement State Compatibility.

Under the Policy Statement on Adequacy and Compatibility of Agreement States Programs, approved by the Commission on June 20, 1997, and published in the *Federal Register* (62 FR 46517; September 3, 1997), this rule is classified as compatibility category "NRC." Compatibility is not required for Category "NRC" regulations. The NRC program elements in this category are those that relate directly to areas of regulation reserved to the NRC by the AEA or the provisions of Title 10 of the CFR, and although an Agreement State may not adopt program elements reserved to the NRC, it may wish to inform its licensees of certain requirements via a mechanism that is consistent with the particular State's administrative procedure laws, but does not confer regulatory authority on the State.

## XVII. Voluntary Consensus Standards.

The National Technology Transfer and Advancement Act of 1995, Pub. L. 104-113, requires that Federal agencies use technical standards that are developed or adopted by voluntary consensus standards bodies unless using such a standard is inconsistent with applicable law or is otherwise impractical. The NRC is not aware of any voluntary consensus standard that could be used as an alternative to the proposed Government-unique standard in the final rule, in order to determine the acceptability of emergency core cooling systems and fuel assemblies for nuclear power reactors. The NRC will consider using a voluntary consensus standard if an appropriate standard is identified.

## XVIII. Availability of Guidance.

The NRC is issuing new guidance, RG 1.222, “Measuring Breakaway Oxidation Behavior,” RG 1.223, “Determining Post Quench Ductility,” RG 1.224, “Establishing Analytical Limits for Zirconium-Alloy Cladding Material” and RG 1.2xx, "Risk-Informed Approach for Addressing the Effects of Debris on Post Accident Long-Term Core Cooling," for the implementation of the requirements in this rulemaking. You may access information and comment submissions related to the guidance by searching on http://www.regulations.gov under Docket ID <NRC-20YY-XXXX>.

RG 1.222 describes an acceptable experimental technique to measure and periodically confirm the breakaway oxidation behavior of a zirconium-alloy cladding material. RG 1.222 also provides guidance on establishing a frequency for confirmatory testing that is sufficient to provide reasonable assurance that the fuel manufacturing process will provide performance consistent with the analytical limits specified in accordance with paragraph (g)(1)(iii) of § 50.46c. RG 1.223 described an acceptable experimental technique for measuring the ductile-to-brittle transition for a zirconium-based cladding alloy in accordance with paragraph (g)(1)(ii) of § 50.46c. RG 1.224 describes an acceptable approach to establish limits for post-quench ductility and breakaway oxidation for zirconium-alloy cladding material required by paragraphs (g)(1)(ii) and (g)(1)(iii) of § 50.46c. RG 1.224 also provides guidance on how to consider oxygen diffusion from inside surfaces in ECCS evaluation to meet the requirements of paragraph (g)(2)(i) of § 50.46c. Finally, RG 1.224 provides acceptable hydrogen pick-up models for use in combination with the hydrogen-based embrittlement limits required by paragraph (g)(1)(ii) of § 50.46c. RG 1.2xx describes acceptable methods and approaches for addressing paragraph (e), “Alternate risk-informed approach for addressing the effects of debris on long-term core cooling" and applicable portions of paragraph (m)(4), "Reporting, corrective actions, and updates" of § 50.46c.

*Changes in Regulatory Guidance Relative to the Draft Regulatory Guidance*

Three draft regulatory guides were issued for public comment concurrent with the proposed rule: DG-1261, "Conducting Periodic Testing for Breakaway Oxidation Behavior" (ADAMS Accession No. ML12284A324); DG-1262, "Testing for Post Quench Ductility" (ADAMS Accession No. ML12284A325); and DG-1263, "Establishing Analytical Limits for Zirconium-Based Alloy Cladding" (ADAMS Accession No. ML12284A323). One draft regulatory guide, DG-1322, "Risk-Informed Approach for Addressing the Effects of Debris on Post Accident Long-Term Core Cooling," was separately issued for public comment on April 20, 2015 (ADAMS Accession No. ML15023A025). These documents are available as described in Section IX, "Availability of Documents," of this document, or online at http://www.nrc.gov/reading-rm/doc-collections/.

The proposed rule included a requirement (*see* § 50.46c(g)(1)(iii)) to measure the onset of breakaway oxidation for a zirconium cladding alloy based on an acceptable experimental technique. The proposed rule also included a requirement that the breakaway oxidation onset measurement be evaluated relative to emergency core cooling system performance, and confirmed annually through periodic testing. The DG-1261 described an experimental technique acceptable to the NRC staff to measure the onset of breakaway oxidation in order to support a specified and acceptable limit on the total accumulated time that a cladding may remain at high temperature, as well as a method acceptable to the NRC to implement the periodic testing and reporting requirements in the proposed rule.

The NRC received many comments on the proposed rule language and the associated guidance provided in DG-1261. The industry commenters expressed views that requiring licensees to report breakaway oxidation results was unnecessary and suggested that the fuel cladding vendors could address the concerns regarding breakaway oxidation with their quality assurance programs. A member of the public made similar comments. A few fuel cladding vendors proposed that periodic test program plans could be developed by the fuel cladding vendors and approved by the NRC. Industry commenters also expressed that the sample frequency should be reduced and be more flexible. Fuel cladding vendors provided comments on a number of aspects of the testing protocol, such as the extent of repeat testing, temperature calibration, sample preparation, and requested more flexibility. On the other hand, a member of the public commented that the test protocol were not similar enough to the expected conditions in a LOCA. The NRC agreed that the periodic testing and reporting requirements could be revised in a way that adds flexibility, decreases cost and burden of breakaway oxidation testing and still achieves the safety objective. The NRC agreed that the objective of the rule can be achieved with rule language that requires a fuel vendor to submit information on its breakaway oxidation testing program for NRC review and approval and that the requirement for licensees to report breakaway oxidation results could be removed. The NRC is issuing RG 1.222 with a number of changes, relative to DG-1261, as a result of comment received on the draft guide and proposed rule. RG 1.222 describes an acceptable experimental technique to measure and periodically confirm the breakaway oxidation behavior of a zirconium-alloy cladding material. Unlike the proposed rule, the final rule does not include a required frequency of periodic testing or a requirement to report results of periodic testing for breakaway oxidation to the NRC. The final rule still requires periodic testing for breakaway oxidation behavior, but outlines a performance-based criterion for an acceptable periodic testing program and frequency. RG 1.222 provides guidance on establishing a frequency for confirmatory testing that is sufficient to provide reasonable assurance that the fuel manufacturing process will provide performance consistent with the analytical limits specified in accordance with paragraph (g)(1)(iii) of § 50.46c.

The proposed rule also included a requirement to establish analytical limits on peak cladding temperature and time at elevated temperature corresponding to the measured ductile-to-brittle transition for the zirconium-alloy cladding material. The DG-1262 described an experimental technique that is acceptable to the NRC for measuring the ductile-to-brittle transition for a zirconium-based cladding alloy.

The NRC received many comments on the proposed rule language and the associated guidance provided in DG-1262. Fuel cladding vendors provided comments on a number of aspects of the testing protocol, such as the extent of repeat testing, temperature calibration, sample preparation, and requested more flexibility. The NRC agreed that the testing protocol could be made more flexible and still achieve the safety objective. The NRC is issuing RG 1.223 with a number of changes, relative to DG-1262, as a result of comment received on the draft guide and proposed rule.

The proposed rule required that analytical limits on post-quench ductility and breakaway oxidation behavior for the zirconium-alloy cladding material be established based on data from NRC-approved experimental techniques. The NRC issued DG-1263 to provide a method of using experimental data to establish regulatory limits.

The NRC received many comments on the proposed rule language and the associated guidance provided in DG-1263. Industry commenters requested more flexibility to define an acceptable data set to evaluate the ductile-to-brittle transition, namely the flexibility to "bin" together samples within a narrow range of hydrogen content. Industry commenters also requested that the guidance include a set of conditions that if met, could eliminate the need for licensing a new cladding alloy to include data collected from irradiated material testing. The industry commenters provided recommended conditions for this provision. The industry commenters also expressed concern about the lack of approved hydrogen pick-up models that would be needed to implement the hydrogen-dependent analytical limit provided in DG-1263.

The NRC agreed that guidance could be developed to outline conditions where data in a narrow range of hydrogen content could be "binned" in order to evaluate the ductile-to-brittle transition. The NRC also agreed that guidance could be developed, based on information from NRC's LOCA research program that showed equivalency between irradiated and un-irradiated, pre-hydrided material, to outline conditions where new cladding alloys could be licensed without testing of irradiated material. Finally, the NRC agreed that the lack of approved hydrogen pick-up models could present an undue burden on implementation of the final rule and therefore outlined acceptable hydrogen-pick up models for all currently approved cladding alloys. The NRC is issuing RG 1.224 with a number of changes, relative to DG-1263, as a result of comment received on the draft guide and proposed rule.

The NRC received many comments on DG-1322, some of which had also been received on the risk-informed aspects of the proposed rule. One set of comments pointed out that the draft guidance did not address several sub-paragraphs under (e) and (m) contained in § 50.46c.

Other comments indicated that the detailed and simplified approaches in DG-1322 were not clearly differentiated, and that the discussion on uncertainty was highly repetitive. These comments led to a restructuring of RG 1.229, such that general information applicable to any risk-informed analysis, such as the uncertainty discussion, is in Section C, and specific technical approaches to the systematic risk assessment are contained in appendices. Other comments indicated that additional clarity was needed regarding the scope of the risk assessment, the treatment of uncertainty, the use of the PRA required by § 50.46c, and the update of the risk assessment. RG 1.229 incorporated a number of changes to address these and many other topics raised by the public comments.

## XIX. Availability of Documents.

The NRC is making the documents identified in the following table available to interested persons through one or more of the methods provided in the **ADDRESSES** section of this document:

| Document | PDR | ADAMS | Web |
|---|---|---|---|
| SECY-98-300 "Options for Risk-Informed Revisions to 10 CFR part 50 – Domestic Licensing of Production and Utilization Facilities," dated December 23, 1998 | X | ML992870048 | |
| Petition for Rulemaking submitted by David J. Modeen on behalf of the Nuclear Energy Institute requesting amendment of 10 CFR 50.44 and 50.46 | X | ML003723791 | |
| *Federal Register* Notice (65 FR 34599), "Petition for Rulemaking filed by David J. Modeen, Nuclear Energy Institute; Consideration of Petition in the Rulemaking Process" | X | ML081780439 | X |

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|---|---|---|---|
| SRM-SECY-02-0057, "Update to SECY-01-0133, 'Fourth Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.46 (ECCS Acceptance Criteria),'" dated March 31, 2003 | X | ML030910476 | X |
| Petition for Rulemaking submitted by Mark Edward Leyse re addressing corrosion of fuel cladding surfaces and a change in the calculations for a loss-of-coolant accident | X | ML070871368 | X |
| *Federal Register* Notice (72 FR 28902), "Mark Edward Leyse; Receipt of Petition for Rulemaking" | X | ML071290466 | X |
| *Federal Register* Notice (73 FR 71564), "Mark Edward Leyse; Consideration of Petition in Rulemaking Process" | X | ML082240164 | X |
| NUREG/CR-6967, "Cladding Embrittlement During Postulated Loss-of-Coolant Accidents" | X | ML082130389 | X |
| Research Information Letter (RIL)-0801, "Technical Basis for Revision of Embrittlement Criteria in 10 CFR 50.46" | X | ML081350225 | X |
| Summary of September 24, 2008, Public Workshop on Technical Basis | X | ML083010496 | |
| GL-1985-022, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," dated December 3, 1985 | X | ML031150731 | |
| RG 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems, Revision 0," dated June 1974 | X | ML111680318 | |
| Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal Pump Strainer While Operating in Suppression Pool Cooling Mode," dated October 7, 1995 | X | ML082490807 | |
| Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors," dated May 6, 1996 | X | ML082401219 | |
| Completion of Staff Reviews of NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," and NRC Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode," dated October 18, 2001 | X | ML012970229 | |

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| Bulletin 2003-01, “Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors,” dated June 9, 2003 | X | ML031600259 | |
| GL 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors,” dated September 13, 2004 | X | ML042360586 | |
| SECY-10-0113, “Closure Options for Generic Safety Issue – 191, Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance,” dated August 26, 2010 | X | ML101820296 | |
| SRM-SECY-10-0113, dated December 23, 2010 | X | ML103570354 | |
| SECY-12-0093, “Closure Options for Generic Safety Issue – 191, Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance,” dated July 9, 2012 | X | ML121320270 | |
| SRM-SECY-12-0093, dated December 14, 2012 | X | ML12349A378 | |
| RG 1.174, Revision 2, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes in the Licensing basis,” dated May 2011 | X | ML100910006 | |
| RG 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” dated March 2009 | X | ML090410014 | |
| Plant Safety Assessment of RIL 0801 | X | ML090340073 | |
| *Federal Register* Notice (73 FR 44778), “Notice of Availability and Solicitation of Public Comments on Documents Under Consideration to Establish the Technical Basis for New Performance-Based Emergency Core Cooling System Requirements” | | | X |
| Supplemental research material – additional PQD tests | X | ML090690711 | |
| Supplemental research material – additional breakaway testing | X | ML090700193 | |
| Draft proposed procedure for Conducting Oxidation and Post-Quench Ductility Tests with Zirconium-Based Alloys | X | ML090900841 | X |
| Draft proposed procedure for Conducting Breakaway Oxidation Tests with Zirconium-based cladding alloys | X | ML090840258 | X |
| Update on Breakaway Oxidation of Westinghouse ZIRLO™ Cladding | X | ML091330334 | X |
| Impact of Specimen Preparation of Breakaway Oxidation of Westinghouse ZIRLO™ Cladding | X | ML091350581 | X |

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| Advance Notice of Proposed Rulemaking, published on August 13, 2009 (74 FR 40765) | X | ML091250132 | X |
| Summary of April 28-29, 2010, Public Meeting on ANPR | X | ML101300490 | |
| SRM-SECY-12-0034, "Proposed Rulemaking – 10 CFR 50.46c: Emergency Core Cooling System Performance During Loss of Coolant Accidents (RIN 3150-AH42)" | X | ML13007A478 | X |
| TR WCAP 16793-NP, Revision 2, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous, and Chemical Debris in the Recirculating Fluid," Appendix A | X | ML11292A021 | |
| PWROG ECCS Analysis Report | X | ML11139A309 | |
| BWROG ECCS Analysis Report | X | ML111950139 | |
| ECCS Audit Report | X | ML12041A078 | |
| Supplement to RIL-0801, "Technical Basis for Revision of Embrittlement Criteria in 10 CFR 50.46" | X | ML113050484 | |
| NUREG-2119, "Mechanical Behavior of Ballooned and Ruptured Cladding" | X | ML12048A475 | X |
| § 50.46c and PRM-50-71 Comment Response Document | X | ML12283A213 | |
| Regulatory Analysis | X | ML12283A188 | |
| Proposed Rule Information Collection Analysis | X | ML112520328 | |
| Draft Regulatory Guide 1261, "Conducting Periodic Testing for Breakaway Oxidation Behavior" | X | ML12284A324 | |
| Draft Regulatory Guide 1262, "Testing for Post Quench Ductility" | X | ML12284A325 | |
| Draft Regulatory Guide 1263, "Establishing Analytical Limits for Zirconium-Based Alloy Cladding" | X | ML12284A323 | |
| Request to Withdraw 50.46a from Commission Consideration | X | ML121500380 | |
| Staff Requirements – SECY-10-0161 – Final Rule: Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements (10 CFR 50.46a) (RIN 3150-AH29) | X | ML12117A121 | |
| June 24-26, 2014 Category 3 Public Meeting Summary – Overview of 50.46c Proposed Rule and Draft Regulatory Guides | | ML14177A048 | |
| July 23, 2014, Category 3 Public Meeting Summary – 50.46c Proposed Rule Topics Specific to BWRs | | ML14204A265 | |
| March 17-19, 2015, Category 3 Public Meeting Summary – Implementation and Regulatory Analysis | | ML15099A571 | |

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|---|---|---|---|
| April 23, 2015, Category 3 Public Meeting Summary – Implementation of 50.46c | | ML15138A434 | |
| May 7, 2015, Category 3 Public Meeting Summary – Implementation of 50.46c | | ML15156A891 | |
| June 4, 2015, Category 3 Public Meeting Summary – Implementation of 50.46c | | ML15169A024 | |
| April 29-30, 2015, Category 3 Public Meeting Summary – Draft Regulatory Guidance | | ML15132A743 | |
| June 9, 2015, Category 3 Public Meeting Summary – 50.46c Long-Term Cooling Provision | | ML15135A144 | |

## List of Subjects

**10 CFR Part 50**

Antitrust, Classified information, Criminal penalties, Fire protection, Intergovernmental relations, Nuclear power plants and reactors, Radiation protection, Reactor siting criteria, Reporting and recordkeeping requirements.

**10 CFR Part 52**

Administrative practice and procedure, Antitrust, Backfitting, Combined license, Early site permit, Emergency planning, Fees, Inspection, Limited work authorization, Nuclear power plants and reactors, Probabilistic risk assessment, Prototype, Reactor siting criteria, Redress of site, Reporting and recordkeeping requirements, Standard design, Standard design certification.

For the reasons set out in the preamble and under the authority of the Atomic Energy

Act of 1954, as amended; the Energy Reorganization Act of 1974; and 5 U.S.C. 552 and 553, the NRC is adopting the following amendments to 10 CFR parts 50 and 52.

**PART 50 -- DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES**

1. The authority citation for part 50 continues to read as follows:

**AUTHORITY:** Atomic Energy Act of 1954, secs. 11, 101, 102, 103, 104, 105, 108, 122, 147, 149, 161, 181, 182, 183, 184, 185, 186, 187, 189, 223, 234 (42 U.S.C. 2014, 2131, 2132, 2133, 2134, 2135, 2138, 2152, 2167, 2169, 2201, 2231, 2232, 2233, 2234, 2235, 2236, 2237, 2239, 2273, 2282); Energy Reorganization Act of 1974, secs. 201, 202, 206, 211 (42 U.S.C. 5841, 5842, 5846, 5851); Nuclear Waste Policy Act of 1982, sec. 306 (42 U.S.C. 10226); 44 U.S.C. 3504 note; Sec. 109, Pub. L. 96-295, 94 Stat. 783.

2. In § 50.8, paragraph (b) is revised to read as follows:

**§ 50.8 Information collection requirements: OMB approval.**

* * * * *

(b) The approved information collection requirements contained in this part appear in §§ 50.30, 50.33, 50.34, 50.34a, 50.35, 50.36, 50.36a, 50.36b, 50.44, 50.46, 50.46c, 50.47, 50.48, 50.49, 50.54, 50.55, 50.55a, 50.59, 50.60, 50.61, 50.61a, 50.62, 50.63, 50.64, 50.65, 50.66, 50.68, 50.69, 50.70, 50.71, 50.72, 50.74, 50.75, 50.80, 50.82, 50.90, 50.91, 50.120, 50.150, and appendices A, B, E, G, H, I, J, K, M, N, O, Q, R, and S to this part.

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3. In § 50.34, paragraphs (a)(4) and (b)(4) are revised to read as follows:

**§ 50.34 Contents of applications; technical information.**

(a)     *     *     *

(4) A preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. Analysis and evaluation of ECCS cooling performance, fuel system performance, and the need for high point vents following postulated loss-of-coolant accidents must be performed in accordance with the requirements of §§ 50.46, 50.46b, and 50.46c, as applicable, for facilities for which construction permits may be issued after December 28, 1974.

*     *     *     *     *

(b)     *     *     *

(4) A final analysis and evaluation of the design and performance of structures, systems, and components with the objective stated in paragraph (a)(4) of this section and taking into account any pertinent information developed since the submittal of the preliminary safety

analysis report. Analysis and evaluation of ECCS cooling performance, fuel system performance, and the need for high point vents following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of §§ 50.46 and 50.46c, as applicable, for facilities for which a license to operate may be issued after December 28, 1974.

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**§ 50.46a [Added and Reserved]**

4. Section 50.46a is redesignated as § 50.46b, and a new § 50.46a is added and reserved.

5. A new § 50.46c is added to read as follows:

**§ 50.46c Emergency core cooling system performance during loss-of-coolant accidents (LOCA).**

(a) *Applicability*. The requirements of this section apply to the design of a light water nuclear power reactor and to the following entities who design, construct or operate a light water nuclear power reactor; each applicant for or holder of a construction permit under this part, each applicant for or holder of an operating license under this part, including a holder of a renewed operating license under 10 CFR part 54 (until the licensee has submitted the certification required under § 50.82(a)(1) to the NRC), each applicant for or holder of a combined license under part 52 of this chapter, including an applicant for an holder of a renewed combined license (until the licensee has submitted the certification required under § 50.82(a)(1) or § 52.11(a)(1) of this chapter to the NRC, as applicable), each applicant for a standard design certification, including an applicant for renewal of a standard design certification (including the applicant for that design certification after the NRC has adopted a final design certification rule), each applicant for a standard design approval under part 52 of this chapter, and each applicant for or holder of a manufacturing license, including an an applicant for or holder of a renewed manufacturing license under part 52 of this chapter.

(b) *Definitions*. As used in this section:

*Breakaway oxidation,* for zirconium-alloy cladding material, means a change in the crystallographic structure of the cladding external oxide layer, resulting in an increase in the oxidation rate and rapid embrittlement due to hydrogen absorption.

*Cladding* means the material structure containing the fissile material and providing a barrier to prevent fission product transport or release to the coolant.

*Crud* means any foreign substance deposited on the surface of fuel cladding prior to initiation of a LOCA.

*ECCS evaluation model* means the calculational framework for evaluating the behavior of the light water reactor reactor system (including fuel) during a postulated LOCA. It includes one or more computer programs and all other information necessary for application of the calculational framework to a specific LOCA, such as mathematical models used, assumptions included in the programs, procedure for treating the program input and output information, specification of those portions of analysis not included in computer programs, values of parameters, and all other information necessary to specify the calculational procedure.

*Loss-of-coolant accident (LOCA)* means a hypothetical accident that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system.

(c) *Relationship to other NRC regulations.* The requirements of this section are in addition to any other requirements applicable to an emergency core cooling system (ECCS) set forth in this part, except as noted in this paragraph. The analytical limits established in accordance with this section, with cooling performance calculated in accordance with an NRC approved ECCS evaluation model, are in implementation of the general requirements with respect to ECCS cooling performance design set forth in this part, including in particular Criterion 35 of appendix A to this part. If the effects of debris on long-term cooling are evaluated using a risk-informed method as described in paragraph (e) of this section, then this method and results can be relied upon to demonstrate compliance with other requirements of this part (including Criteria 38 and 41 of appendix A to this part), as allowed by this section and Criteria 35, 38 and 41, but only to the extent requested in an application and approved by the NRC.

(d) *Emergency core cooling system performance.*

(1) *ECCS performance criteria.* Each LWR must be provided with an ECCS designed to satisfy the following performance requirements in the event of, and following, a postulated LOCA. The demonstration of ECCS performance must comply with paragraph (d)(2) of this

section:

(i) Core temperature during and following the LOCA event does not exceed the analytical limits for the fuel design used for ensuring acceptable performance as defined in this section.

(ii) The ECCS provides sufficient coolant so that decay heat will be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

(2) *ECCS performance demonstration*. ECCS performance must be demonstrated using an ECCS evaluation model meeting the requirements of paragraph (d)(2)(i) or (d)(2)(ii) of this section, and satisfy the analytical requirements in paragraphs (d)(2)(iii), (d)(2)(iv), and (d)(2)(v) of this section. Paragraph (e) of this section may be used for consideration of debris in long term cooling as described in paragraph (d)(2)(iii) of this section.

(i) *Realistic ECCS evaluation model.* A realistic ECCS evaluation model must describe the behavior of the reactor system during a loss-of-coolant accident in a realistic manner. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for so that, when the calculated ECCS cooling performance is compared to the applicable specified and NRC-approved analytical limits, there is a high level of probability that the limits would not be exceeded.

(ii) *Appendix K model*. Alternatively, an appendix K ECCS evaluation model may be developed in conformance with the required and acceptable features of appendix K to this part.

(iii) *Core geometry and coolant flow*. The ECCS evaluation model must address calculated changes in core geometry and must consider those factors, including debris, that may alter localized coolant flow in the core or inhibit delivery of coolant to the core. However, a licensee may evaluate effects of debris on long-term cooling using a risk-informed approach as specified in paragraph (e) of this section, in which case the ECCS evaluation model specified in paragraph (d)(2)(i) or d(2)(ii) of this section need not include the effects of debris on long-term cooling.

(iv) *LOCA analytical requirements*. ECCS performance must be demonstrated for a range of postulated loss-of-coolant accidents of different sizes, locations, and other properties, sufficient to provide assurance that the most severe postulated loss-of-coolant accidents have been identified. ECCS performance must be demonstrated for the accident, and the post-accident recovery and recirculation period.

(v) *Modeling requirements for fuel designs: uranium oxide or mixed uranium-plutonium oxide pellets within zirconium-alloy cladding*. If the reactor is fueled with uranium oxide or mixed uranium-plutonium oxide pellets within cylindrical zirconium-alloy cladding, then the ECCS evaluation model must address the fuel system modeling requirements in paragraph (g)(2) of this section.

(3) *Required documentation.* Upon implementation of this section in accordance with paragraph (p) of this section, the documentation requirements of this paragraph apply and supersede the requirements in appendix K to this part, section II, “Required Documentation.”

(i)(A) A description of the ECCS evaluation model must be submitted to the NRC. The description must be sufficiently complete to permit technical review of the analytical approach, including the equations used, their approximations in difference form, the assumptions made, and the values of all parameters or the procedure for their selection, as for example, in accordance with a specified physical law or empirical correlation.

(B) Detailed source code of each computer program, in the same form as used in the ECCS evaluation model, must be provided to the NRC upon request.

(ii) For each computer program, solution convergence must be demonstrated by studies of system modeling, and/or noding and calculational time steps.

(iii) Appropriate sensitivity studies must be performed for each ECCS evaluation model, to evaluate the effect on the calculated results of variations in noding, phenomena assumed in the calculation to predominate, including pump operation or locking, and values of parameters over their applicable ranges. For items to which results are shown to be sensitive, the choices made must be justified.

(iv) To the extent practicable, predictions of the ECCS evaluation model, or portions thereof, must be compared with applicable experimental information.

(v) The technical adequacy of the calculational methods used in the ECCS evaluation models must be documented. For models covered by paragraph (d)(2)(i) of this section, this documentation must demonstrate that the performance criteria of paragraph (d)(1) of this section would not be exceeded. For models covered by paragraph (d)(2)(ii) of this section, this documentation must address compliance with required features of section I of appendix K to this part, and must demonstrate that the performance criteria of paragraph (d)(1) of this section would not be exceeded.

(e) *Alternate risk-informed approach for addressing the effects of debris on long-term core cooling.*

(1) *Attributes of an acceptable risk-informed approach.* An entity may request that the NRC approve a risk-informed approach for addressing the effects of debris on long-term core cooling to demonstrate compliance with the requirements in paragraph (d)(2)(iii) of this section. If the alternate risk-informed approach is used, then the ECCS evaluation model specified in paragraph (d)(2)(i) or (d)(2)(ii) of this section need not include the effects of debris on long-term cooling. If an entity desires to change the methods employed in the systematic processes in paragraph (e)(1)(iii) of this section, as approved by the NRC, then the entity shall obtain NRC review and approval before the change is implemented. The risk-informed approach must:

(i) Demonstrate that any increase in core damage frequency and large early release frequency resulting from implementing the alternative risk-informed approach will be small;

(ii) Maintain sufficient defense-in-depth and safety margins; and

(iii) Utilize systematic processes that evaluate the risk of debris for internal and external events initiated during full power, low power, and shutdown operation. These process include the following:

(A) A PRA that, at a minimum, models severe accident scenarios resulting from internal events occurring at full power operation and reasonably reflects the current plant configuration and operating practices, and applicable plant and industry operational experience, is of sufficient scope, level of detail, and technical adequacy to support the alternative process, and is subjected to a peer review process that assesses the PRA against a standard or set of acceptance criteria that is approved for use by the NRC.

(B) A PRA or other risk assessment method, such as margins-type approaches, bounding calculations, or other systematic evaluation techniques to evaluate hazards and operating modes not covered in paragraph (e)(1)(iii)(A) of this section.

(C) Engineering calculations, tests, and other supporting information used in the risk assessment.

(iv) Ensure that the scope, level of detail and technical adequacy of the systematic risk assessment is commensurate with the reliance on risk information and that it was performed under a quality assurance program.

(v) Utilize a monitoring program that ensures the acceptance criteria in paragraphs (e)(1)(i) and (ii) of this section will continue to be met. The monitoring program must assess the effects of design or plant modifications, procedure changes, as-found conditions, identified changes or errors in the analysis, industry operating experience, and any other information that could result in increased risk, or decreased defense-in-depth or safety margins, under the alternative risk-informed approach. However, this requirement does not apply to design certification rules.

(2) *Contents of application.* An entity seeking to use the risk-informed approach under this paragraph, must submit an application with the following information:

(i) A description of the alternative risk-informed approach;

(ii) A description of the quality assurance program and the measures taken to assure that the scope, level of detail, and technical adequacy of the systematic processes that evaluate the plant for internal and external events initiated during full power, low power, and shutdown operation (including the PRA, margins-type approaches, or other systematic evaluation techniques used to evaluate severe accidents) are commensurate with the reliance on risk information;,

(iii) A description of, and basis for acceptability of, the evaluations conducted to demonstrate compliance with paragraphs (e)(1)(i) and (e)(1)(ii) of this section; and

(iv) A description of the monitoring program.

(3) *NRC approval.* If the NRC determines that the application demonstrates that the requirements of paragraph (e)(1) of this section are met then it may approve the use of the risk-informed approach for addressing debris effects on long-term cooling when issuing the license, regulatory approval or amendments thereto. The NRC's approval must specify the circumstances under which the licensee or applicant, as applicable, shall notify the NRC of changes or errors in the risk evaluation approach utilized to address the effects of debris on long-term cooling.

(f) [Reserved]

(g) *Fuel system designs: uranium oxide or mixed uranium-plutonium oxide pellets within cylindrical zirconium-alloy cladding.*

(1) *Fuel performance criteria.* Fuel consisting of uranium oxide or mixed uranium-plutonium oxide pellets within cylindrical zirconium-alloy cladding must be designed and manufactured to meet the following requirements:

(i) *Peak cladding temperature*. Except as provided in paragraph (g)(1)(ii) of this section, the calculated maximum fuel element cladding temperature shall not exceed 2200 °F.

(ii) *Post-quench ductility*. Analytical limits on peak cladding temperature and integral time at temperature shall be established that correspond to the measured ductile-to-brittle transition for the zirconium-alloy cladding material based on an NRC-approved experimental technique. The calculated maximum fuel element temperature and time at elevated temperature shall not exceed the established analytical limits. The analytical limits must be approved by the NRC. If the peak cladding temperature, in conjunction with the integral time at temperature analytical limit, established to preserve cladding ductility is lower than the 2200 ºF limit specified in paragraph (g)(1)(i) of this section, then the lower temperature shall be the applicable analytical limit on peak cladding temperature.

(iii) *Breakaway oxidation.* An analytical time limit that has been shown to preclude breakaway oxidation using an NRC-approved experimental technique must be determined and specified for each zirconium-alloy cladding material. The analytical limits must be approved by the NRC. The total time that the cladding is predicted to remain above the temperature that the zirconium-alloy has been shown to be susceptible to breakaway oxidation must be less than the analytical limit. The breakaway oxidation behavior must be periodically confirmed using an NRC-approved experimental technique capable of determining the effect of composition changes or manufacturing changes on the breakaway oxidation behavior. The frequency of confirmatory testing must provide reasonable assurance that fuel is being manufactured consistent with the specified analytical limit.

(iv) *Maximum hydrogen generation*. The calculated total amount of hydrogen generated from any chemical reaction of the fuel cladding with water or steam must not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(v) *Long-term cooling*. After any calculated successful initial operation of the ECCS, the calculated core temperature must be maintained to prevent further cladding failure, and the ECCS shall provide sufficient coolant to remove decay heat, for the extended period of time required by the long-lived radioactivity remaining in the core.

(2) *Fuel system modeling requirements.* The ECCS evaluation model required by paragraph (d)(2) of this section must model the fuel system in accordance with the following requirement:

(i) If an oxygen source is present on the inside surfaces of the cladding at the onset of the LOCA, then the effects of oxygen diffusion from the cladding inside surfaces must be considered in the ECCS evaluation model.

(ii) The thermal effects of crud and oxide layers that accumulate on the fuel cladding during plant operation must be evaluated.

(h) [Reserved]

(i) [Reserved]

(j) [Reserved]

(k) *Use of NRC-approved fuel in reactor*.

(1) *Fuel load*. A licensee may not load fuel into a reactor unless the licensee determines that the fuel either meets the requirements of paragraph (d) of this section or, for uranium oxide and mixed uranium-plutonium oxide pellets within cylindrical zirconium-alloy cladding, the fuel specific analytical limits and requirements in paragraph (g) of this section, or complies with technical specifications governing lead test assemblies in its license.

(2) *Operation*. If a licensee determines that fuel in the reactor no longer complies with the requirements of paragraph (d) or, for uranium oxide and mixed uranium–plutonium oxide pellets within cylindrical zirconium-alloy cladding, the fuel-specific analytical limits and requirements in paragraph (g) of this section, then the licensee must take immediate action to come into compliance with paragraph (d) or (g) of this section as applicable.

(l) *Authority to impose restrictions on operation.* The Director of the Office of Nuclear Reactor Regulation or the Director of the Office of New Reactors may impose restrictions on reactor operation if it is found that the evaluations of ECCS cooling performance submitted are not consistent with the requirements of this section.

(m) *Reporting, corrective actions, and updates*. Each entity subject to the requirements of this section shall comply with paragraphs (m)(1) and (m)(2) of this section. Each entity demonstrating acceptable long--term core cooling under the provisions of paragraph (e) of this section shall also comply with the requirements of paragraph (m)(6) through (8) of this section.

(1) *ECCS evaluation model: reporting*.

(i) If an entity identifies any change to, or error in, an ECCS evaluation model or the application of such a model that does not result in any predicted response that exceeds any of the acceptance criteria specified in this section and is itself not significant as defined in paragraph (n) of this section, then a report describing each such change or error and a demonstration that the error or change is not significant must be submitted to the NRC no later than 12 months after the change or discovery of the error.

(ii) If an entity identifies any change to, or error in, an ECCS evaluation model, or the application of such a model, that does not result in any predicted response that exceeds any of the acceptance criteria but is significant as defined in paragraph (n) of this section, then a report describing each such change or error, and a schedule for providing a reanalysis and implementation of corrective actions must be submitted within 60 days of the change or discovery of the error.

(iii) If a licensee of a facility licensed to operate identifies any change to, or error in, an ECCS evaluation model or the application of such a model, that results in any of the acceptance criteria specified in this section to be exceeded at the facility, then the licensee shall report the change or error under §§ 50.55(e), 50.72, and 50.73, as applicable, and submit a report describing each such change or error, and a schedule for providing a reanalysis and implementation of corrective actions within 30 days of the change or discovery of the error.

(2) *ECCS evaluation model: corrective action*.

(i) If a licensee of a facility licensed to operate identifies any change to, or error in, an ECCS evaluation model or the application of such a model, that results in any of the acceptance criteria specified in this section to be exceeded at the facility, then the licensee (in the case of a combined license under part 52 of this chapter, after the Commission has made the finding under § 52.103(g) of this chapter) shall take immediate action to bring the facility into compliance with the acceptance criteria. In addition, the corrective action as described in the report required by paragraph (m)(1) of this section must be implemented.

(ii) If a design certification applicant is required by paragraphs (m)(1)(ii) of this section to submit a reanalysis, or identifies a change to, or error in an ECCS evaluation model, or in the application of such a model, that results in any predicted response that exceeds any of the acceptance criteria specified in this section, then the applicant must submit a reanalysis, accompanied by either a revision to its design certification application under review, or an application to amend the design certification application, as applicable, reflecting the reanalysis.

(3) through (5) [Reserved]

(6) *Risk-informed consideration of debris: reporting*. If an entity implementing the risk-

informed approach to address debris effects determines that either the acceptance criteria of paragraph (e)(1)(i) of this section have been exceeded or the requirements of paragraph (e)(1)(ii) of this section are no longer met, then the following reporting actions must be taken.

(i) A design certification applicant before issuance of a final design certification rule and a design certification renewal applicant before issuance of a final renewed design certification rule shall evaluate and report, in a timely fashion, information concerning compliance with the acceptance criteria in paragraph (e)(1) of this section in accordance with the requirements of part 21 of this chapter.

(ii) A design certification applicant during the period of validity under § 52.55(a) and (b) of this chapter, once the design certification is referenced in a combined license application or combined license, shall submit, in a timely fashion, a report regarding information concerning compliance with the acceptance criteria of paragraph (e)(1) of this section in accordance with the requirements of part 21 of this chapter. Until the design certification rule is referenced, no reporting is required.

(iii) A combined license applicant, after performing the evaluation required by paragraph (e) of this section and including the information in its application, determines that any acceptance criterion of paragraph (e)(1) of this section is not met, then the applicant shall submit, in a timely fashion, a report regarding information concerning compliance with the acceptance criteria of paragraph (e)(1) of this section.

(iv) A combined license holder, before the Commission finding under § 52.103(g) of this chapter, shall submit, in a timely fashion, a report whenever the failure to comply with the acceptance criteria of paragraph (e)(1) of this section also requires a licensee notification under § 52.99 of this chapter.

(v) Licensees and combined license holders after the Commission finding under § 52.103(g) of this chapter shall submit, in a timely fashion, a report concerning compliance with the acceptance criteria of paragraph (e)(1) of this section. The report must be submitted in accordance with § 50.72 or 50.73.

(7) *Risk-informed consideration of debris: corrective action.* If an entity implementing the risk-informed approach to address debris effects determines that either the acceptance criteria of paragraph (e)(1)(i) of this section have been exceeded or the requirements of paragraph (e)(1)(ii) of this section are no longer met, then the following corrective actions must be taken;

(i) A design certification applicant before issuance of a design certification rule and a design certification renewal applicant before issuance of a final renewed design certification rule shall submit, in a timely fashion, an amendment to its pending application. The amendment must describe any changes to the design and/or changes in the analyses, evaluations, modeling (including the PRA and its supporting analyses) and ITAAC needed to demonstrate that the design meets the acceptance criteria in paragraph (e)(1) of this section.

(ii) A design certification applicant during the period of validity under § 52.55(a) and (b) of this chapter, once the design certification rule is referenced in a combined license application or combined license, shall request, in a timely fashion, an amendment of the design certification.

The amendment must describe any changes to the design and/or changes in the analyses, evaluations, modeling (including the PRA and its supporting analyses) and ITAAC needed to demonstrate that the design meets the acceptance criteria in paragraph (e)(1) of this section. Until the design certification rule is referenced, no corrective action is required.

(iii) A combined license applicant and a combined license holder under part 52 of this chapter shall submit, in a timely fashion, an amendment to its application or an application for amendment of its combined license, as applicable. The amendment or application, as applicable, must include necessary changes to its updated final safety analysis report, any necessary changes to ITAAC (with the bases for the changes) and, if applicable, a request for exemption from a referenced design certification rule (but need not address the criteria for obtaining an exemption). The amendment or application must demonstrate that the acceptance criteria of paragraph (e)(1) of this section are met, and must describe any design, procedural, or operational changes or changes to the analyses, evaluations, modeling (including the PRA and its supporting analyses) and ITAAC needed to demonstrate that the design meets the acceptance criteria in paragraph (e)(1) of this section. A combined license holder, after the Commission finding under § 52.103(g) of this chapter, shall take timely action to ensure that the acceptance criteria of paragraphs (e)(1)(i) and (e)(1)(ii) of this section are met.

(iv) The applicant for an operating license under this part 50 shall submit, in a timely fashion, an amendment to its pending application. The amendment must describe any changes to the design and/or changes in the analyses, evaluations, modeling (including the PRA and its supporting analyses) needed to demonstrate that the design meets the acceptance criteria in paragraph (e)(1) of this section. A holder of an operating license under this part 50 shall take timely action to ensure that the acceptance criteria of paragraphs (e)(1)(i) and (e)(1)(ii) of this section are met.

(v) The NRC need not address either the issue finality criteria in §§ 52.63, 52.83, and 52.93 of this chapter or the backfitting criteria in § 50.109 when acting on an entity's submittal required by this paragraph (m)(7) and shall, as part of any approval, issue any necessary exemption upon a finding that the exemption is authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest.

(8*) Risk-informed consideration of debris: updates.*

(i) Each licensee shall update its risk informed evaluations under paragraph (e)(1) of this section no later than 48 months after initial NRC approval or the latest update. However, this requirement does not apply to holders of combined licenses before initial loading of fuel under § 52.103(g) of this chapter, or to design certification rules. Each licensee that desires to change the methods or approaches employed in the NRC approved risk-informed evaluation of debris shall submit an amendment to its operating license under 10 CFR 50.90 through 50.92. The amendment should describe any changes the licensee wishes to make to the analyses, evaluations, and modeling (including the PRA and its supporting analyses).

(ii) Each holder of a combined license shall, no later than the initial loading of fuel under § 52.103(g) of this chapter, update the analyses, evaluations, and modeling performed under paragraph (e) of this section. The updating must correct identified errors, and incorporate licensee-adopted changes to the plant design, the licensee's proposed operational practices, and any applicable industry operational experience known to the licensee. As appropriate, the licensee shall update the PRA and its supporting analyses, and re-perform the evaluations of risk, defense-in-depth, and safety margins to confirm that the acceptance criteria identified in paragraph (e)(1) of this section continue to be met. If the licensee determines that any acceptance criterion of paragraph (e)(1) of this section is not met, then the licensee shall submit, in a timely fashion, an application for amendment of its combined license (and departure from a referenced design certification rule, if applicable), including necessary changes to its updated final safety analysis report and any necessary changes to the ITAAC. The amendment application must demonstrate that the acceptance criteria of paragraph (e)(1) of this section are met, and must describe any changes to the analyses, evaluations and modeling needed to support that conclusion. The application must explain either the bases for any change to ITAAC or why no changes to ITAAC are needed. The application must, if applicable, include a request for exemption from a referenced design certification rule, but need not address the criteria for obtaining an exemption. The licensee shall also submit any report required by § 52.99 of this chapter. The NRC need not address the issue finality criteria in §§ 52.63, 52.83, and 52.98 of this chapter when acting on this amendment, and shall – as part of any approved amendment - issue any necessary exemption upon a finding that the exemption is authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest.

(n) *Significant change or error in the ECCS evaluation model.*

(1) *Uranium and mixed urainium-plutionium oxide fuel.* For uranium oxide and mixed uranium-plutonium oxide pellets within cylindrical zirconium-alloy cladding, a significant change or error is one that results in a calculated–

(i) Peak fuel cladding temperature different by more than 50 °F from the temperature calculated for the limiting transient using the last acceptable evaluation model, or is a cumulation of changes and errors such that the sum of the absolute magnitudes of the respective temperature changes is greater than 50 °F; or

(ii) Integral time at temperature different by more than 0.4 percent ECR from the oxidation calculated for the limiting transient using the last acceptable evaluation model, or is a cumulation of changes and errors such that the sum of the absolute magnitudes of the respective oxidation changes is greater than 0.4 percent ECR.

(2) *Other Fuel Types. [Reserved]*

(o) [Reserved]

(p) *Implementation.*

(1) Construction permit applications docketed after [INSERT DATE THAT IS 30 DAYS AFTER THE DATE OF PUBLICATION IN THE FEDERAL REGISTER], and construction permits issued under this part which are based on such applications and are issued after **[INSERT DATE THAT IS 30 DAYS AFTER THE DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**, must comply with the requirements of this section.

(2)(i) Each holder of an operating license issued under this part as of **[INSERT DATE THAT IS 30 DAYS AFTER THE DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**, and each holder of an operating license issued under this part which is based upon a construction permit in effect as of **[INSERT DATE THAT IS 30 DAYS AFTER THE DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]** (including deferred and reinstated construction permits), must submit an implementation plan and schedule for achieving compliance with the provisions of this regulation with the exception of the consideration of debris effects under paragraph (d)(2)(iii) of this section. The implementation plan must identify the evaluation model(s), fuel design(s) and cladding alloy(s), and analytical limits to be used in the ECCS performance demonstration, along with the relative level of effort needed to complete the performance demonstration. The schedule must identify, for each element of the ECCS performance demonstration required to be submitted to the NRC for review (e.g., evaluation model, hydrogen uptake model, cladding alloy), the earliest possible date for submission and the expected date of submission. The implementation plan and schedule must be submitted within 6 months of **[INSERT DATE THAT IS 30 DAYS AFTER THE DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**, and updated by the licensee every 12 months until the license amendment request has been submitted and docketed by the NRC for review.

(ii) The licensee's request for NRC approval under paragraph (d)(2) of this section must be in the form of license amendment application under § 50.90. The application must be submitted by no later than 60 months after **[INSERT DATE THAT IS 30 DAYS AFTER THE DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**.

(iii) Licensees must be in compliance with the requirements of this section no later than 84 months after **[INSERT DATE THAT IS 30 DAYS AFTER THE DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**. Until such compliance is achieved, the requirements of § 50.46 continue to apply for purposes of ECCS design and fuel design.

(3) Operating license applications docketed after [INSERT DATE THAT IS 30 DAYS AFTER THE DATE OF PUBLICATION IN THE ***FEDERAL REGISTER***], and operating licenses issued under this part which are based upon construction permits issued in accordance with paragraph (p)(1) of this section and are issued after **[INSERT DATE THAT IS 30 DAYS AFTER THE DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**, must comply with the requirements of this section.

(4) Standard design certification applications, standard design approval applications, and manufacturing license applications under part 52 of this chapter (including applications for amendment), any of which are docketed after **[INSERT DATE THAT IS 30 DAYS AFTER THE DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**; renewal applications for standard design certifcations and manufacturing licenses issued after [INSERT DATE THAT IS 30 DAYS AFTER THE DATE OF PUBLICATION IN THE FEDERAL REGISTER]; applications for new branches of standard design certifications docketed after **[INSERT DATE THAT IS 30 DAYS AFTER THE DATE OF PUBLICATION IN THE *FEDERAL REGISTER*];** and standard design certifications, standard design approvals, manufacturing licenses, and their renewal (as applicable) any of which are based upon applications subject to this paragraph (p)(4), must comply with the requirements of this section.

(5) Standard design certifications under part 52 of this chapter issued before **[INSERT DATE THAT IS 30 DAYS AFTER THE DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**, must comply with the requirements of this section by the time of their first renewal (if a renewal is sought).

(6) Standard design certifications, standard design approvals, and manufacturing licenses under part 52 of this chapter issued after **[INSERT DATE THAT IS 30 DAYS AFTER THE DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**, whose applications were pending as of **[INSERT DATE THAT IS 30 DAYS AFTER THE DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**, and new branches of standard design certifications issued after **[INSERT DATE THAT IS 30 DAYS AFTER THE DATE OF PUBLICATION IN THE *FEDERAL REGISTER*],** whose applications were pending as of **[INSERT DATE THAT IS 30 DAYS AFTER THE DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**, must comply with the requirements of this section by the time of renewal.

(7) Combined license applications under part 52 of this chapter whose applications are docketed after **[INSERT DATE THAT IS 30 DAYS AFTER THE DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**, and combined licenses which are based on such applications and are issued after [INSERT DATE THAT IS 30 DAYS AFTER THE DATE OF PUBLICATION IN THE FEDERAL REGISTER] must comply with the requirements of this section.

(8) Combined licenses under part 52 of this chapter issued before **[INSERT DATE THAT IS 30 DAYS AFTER THE DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**, and combined licenses issued after **[INSERT DATE THAT IS 30 DAYS AFTER THE DATE OF**

**PUBLICATION IN THE *FEDERAL REGISTER*],** whose applications were docketed before **[INSERT DATE THAT IS 30 DAYS AFTER THE DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**, must comply with the requirements of this section by initial fuel loading or 84 months from **[INSERT DATE THAT IS 30 DAYS AFTER THE DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**, whichever is later.

6. In appendix A to part 50, under the heading, "Criteria," criteria 35, 38, and 41 are revised to read as follows:

**Appendix A to Part 50 – General Design Criteria for Nuclear Power Plants**

* * * * *

*Criterion 35 -- Emergency core cooling*. A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that 1) fuel and clad damage that could interfere with continued effective core cooling is prevented and 2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

The effects of debris on system safety function with respect to long-term cooling may be evaluated in accordance with all requirements applicable to the risk-informed approach in § 50.46c.

* * * * *

Criterion 38 – Containment heat removal system. A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

The effects of debris on safety system function with respect to the maintenance of containment pressure and temperature may be evaluated in accordance with all requirements applicable to the risk-informed approach in § 50.46c.

* * * * *

Criterion 41 – Containment atmosphere cleanup. Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

The effects of debris on system safety function following occurrence of the postulated accidents may be evaluated in accordance with all requirements applicable to the risk-informed approach in § 50.46c.

*    *    *    *    *

7. In appendix K to part 50, a new paragraph II.6 is added to read as follows:

**Appendix K to Part 50 – ECCS Evaluation Models**

*    *    *    *    *

II. *    *    *

6. Upon each entity's implementation of § 50.46c in accordance with § 50.46c(p), the documentation requirements in § 50.46c(d)(3) apply and supersede the requirements of section

II of this appendix for that entity and associated regulatory approval.

**PART 52 – LICENSES, CERTIFICATIONS AND APPROVALS FOR NUCLEAR POWER PLANTS**

8. The authority citation for part 52 continues to read as follows:

**AUTHORITY:** Atomic Energy Act of 1954, secs. 103, 104, 147, 149, 161, 181, 182, 183, 185, 186, 189, 223, 234 (42 U.S.C. 2133, 2134, 2167, 2169, 2201, 2231, 2232, 2233, 2235, 2236, 2239, 2273, 2282); Energy Reorganization Act of 1974, secs. 201, 202, 206, 211 (42 U.S.C. 5841, 5842, 5846, 5851); 44 U.S.C. 3504 note.

9. In § 52.47, paragraph (a)(4) is revised to read as follows:

**§ 52.47 Contents of applications; technical information.**

*      *      *      *      *

(a) *      *      *

(4) An analysis and evaluation of the design and performance of structures, systems, and components with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. Analysis and evaluation of emergency core cooling system (ECCS) cooling performance, fuel system design, and the need for high-point vents following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of

§§ 50.46, 50.46b and 50.46c of this chapter, as applicable;

*     *     *     *     *

10. In § 52.79, paragraph (a)(5) is revised to read as follows:

**§ 52.79 Contents of applications; technical information in final safety analysis report.**

(a) *     *     *

(5) An analysis and evaluation of the design and performance of structures, systems, and components with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. Analysis and evaluation of ECCS cooling performance, fuel system design and the need for high-point vents following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of §§ 50.46, 50.46b and 50.46c of this chapter, as applicable;

*     *     *     *     *

11. In § 52.137, paragraph (a)(4) is revised to read as follows:

**§ 52.137 Contents of applications; technical information.**

*     *     *     *     *

(a) * * *

(4) An analysis and evaluation of the design and performance of SSCs with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of SSCs provided for the prevention of accidents and the mitigation of the consequences of accidents. Analysis and evaluation of ECCS cooling performance, fuel system design and the need for high-point vents following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of §§ 50.46, 50.46b, and 50.46c of this chapter, as applicable;

* * * * *

12. In § 52.157, paragraph (f)(1) is revised to read as follows:

**§ 52.157 Contents of applications; technical information in the final safety analysis report.**

* * * * *

(f) * * *

(1) An analysis and evaluation of the design and performance of structures, systems, and components with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal

operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. Analysis and evaluation of ECCS cooling performance, fuel system design and the need for high-point vents following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of §§ 50.46, 50.46b, and 50.46c of this chapter, as applicable;

* * * * *

Dated at Rockville, Maryland, this 6th day of March, 2014.

For the Nuclear Regulatory Commission.

Annette L. Vietti-Cook,
Secretary of the Commission