[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]

Performance-Based Emergency Core Cooling Systems (ECCS) Cladding Acceptance Criteria

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC or the Commission) is proposing to amend its regulations to revise the acceptance criteria for the emergency core cooling system (ECCS) for light-water nuclear power reactors. The proposed ECCS acceptance criteria are performance-based, and reflect recent research findings which identified new embrittlement mechanisms for fuel rods with zirconium alloy cladding under loss-of-coolant accident (LOCA) conditions. Further, the proposed rule addresses two petitions for rulemaking 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. The NRC is also seeking public comment on three draft regulatory guides that would support the implementation of the proposed rule.

DATES: Submit comments on the rule and draft guidance by **[INSERT DATE 75 DAYS AFTER PUBLICATION IN THE *FEDERAL REGISTER*]**. To facilitate NRC review, please distinguish between comments submitted on the proposed rule and comments submitted on the draft guidance. Submit comments on the information collection aspects of this rule by **[INSERT DATE 30 DAYS AFTER PUBLICATION IN THE *FEDERAL REGISTER*]**. Comments received after these dates will be considered if it is practical to do so, but assurance of consideration cannot be given to comments received after these dates.

ADDRESSES: The methods for accessing information and comment submissions, and submitting comments on the proposed rule are different from the methods for accessing information and comment submissions, and submitting comments on the draft regulatory guides.

Proposed rule.

You may access information and comment submissions related to this proposed rule by searching on http://www.regulations.gov under Docket ID NRC-2008-0332. You may submit comments on the proposed rule by 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: 301-492-3668; e-mail: Carol.Gallagher@nrc.gov.
- **E-mail comments to**: Rulemaking.Comments@nrc.gov. If you do not receive an automatic e-mail reply confirming receipt, then contact us at 301-415-1677.
- **Fax comments to:** Secretary, U.S. Nuclear Regulatory Commission at 301-415-1101.
- **Mail comments to:** Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, ATTN: Rulemakings and Adjudications Staff.

- **Hand deliver comments to:** 11555 Rockville Pike, Rockville, Maryland 20852, between 7:30 a.m. and 4:15 p.m. (Eastern Time) Federal workdays; telephone: 301-415-1677.

Draft Regulatory Guides.

You may access information and comment submissions related to the draft regulatory guides by searching on http://www.regulations.gov under Docket IDs NRC-2012-0041, NRC-2012-0042, and NRC-2012-0043 respectively. You may submit comments on the draft regulatory guides by the following methods:

- **Federal rulemaking Web site:** Go to http://www.regulations.gov and search for Docket IDs NRC-2012-0041, NRC-2012-0042, and NRC-2012-0043, respectively. Address questions about NRC dockets to Carol Gallagher; telephone: 301-492-3668; e-mail: Carol.Gallagher@nrc.gov.

- **Mail comments to:** Cindy Bladey, Chief, Rules, Announcements, and Directives Branch (RADB), Office of Administration, Mail Stop: TWB-05-B01M, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

- **Fax comments to:** RADB at 301-492-3446.

Information Collections.

You may submit comments on the information collections by the methods described in the **SUPPLEMENTARY INFORMATION** section of this document, under the heading, "Paperwork Reduction Act Statement."

For additional direction on accessing information and submitting comments, see "Accessing Information and Submitting Comments" in the **SUPPLEMENTARY INFORMATION** section of this document.

FOR FURTHER INFORMATION CONTACT: Tara Inverso, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, telephone: 301-415-1024, e-mail: Tara.Inverso@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:

I. Accessing Information and Submitting Comments

A. Accessing Information

Please refer to Docket ID NRC-2008-0332, Docket ID NRC-2012-0041, Docket ID NRC-2012-0042, or Docket ID NRC-2012-0043 when contacting the NRC about the availability of information for this proposed rule or draft regulatory guides, respectively. You may access information related to this proposed rulemaking or draft regulatory guides by the following methods:

- **Federal Rulemaking Web Site:** Go to http://www.regulations.gov and search for Docket ID NRC-2008-0332 for the proposed rule, and Docket ID NRC-2012-0041, Docket ID NRC-2012-0042, or Docket ID NRC-2012-0043 for the draft regulatory guides.

- **NRC's Agencywide Documents Access and Management System (ADAMS):** You may access publicly-available documents online in the NRC Library 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. The ADAMS accession number for each document referenced in this notice (if that document is available in ADAMS) is provided the first time that a document is referenced. In addition, for the convenience of the reader, the ADAMS accession numbers are provided in a table in the section of this notice entitled, *Availability of Documents.*

- **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.

B. Submitting Comments

Please include the appropriate NRC Docket ID in the subject line of your comment submission, in order to ensure that the NRC is able to make your comment submission available to the public in that docket.

The NRC cautions you not to include identifying or contact information in comment submissions that you do not want to be publicly disclosed. The NRC posts all comment submissions at http://www.regulations.gov as well as entering the comment submissions into

ADAMS, and the NRC does not edit comment submissions to remove identifying or contact information.

If you are requesting or aggregating comments from other persons for submission to the NRC, then you should inform those persons not to include identifying or contact information in their comment submissions that they do not want to be publicly disclosed. Your request should state that the NRC will not edit comment submissions to remove such information before making the comment submissions available to the public or entering the comment submissions into ADAMS.

II. Background

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 (ADAMS Accession No. ML992870048), the 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, it 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. 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) 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^{TM}). The NEI stated that reactor fuel vendors had subsequently developed new cladding materials other than zircaloy and ZIRLO^{TM} and that, in order for licensees to use these new materials under the regulations, licensees needed to request NRC approval of exemptions from §§ 50.44 and 50.46.

On May 31, 2000, the NRC published a notice of receipt (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, "§ 50.46c and PRM-50-71 Comment Response Document" (ADAMS Accession No. ML112520303).

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* (FR) 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 proposed rule.[1]

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),'" 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.

Research Results

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), to develop the body of technical information needed to

[1] PRM-50-71 also requested changes to § 50.44. Those changes were addressed in a rulemaking which 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.

support the new regulations.

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. The research results revealed that alloy composition has a minor effect on embrittlement, but that the cladding corrosion that occurs as fuel burnup increases has a substantial effect on embrittlement. One of the major findings of the NRC's research program was that hydrogen, which is absorbed in the cladding as a result of zirconium oxidation (e.g., corrosion) under normal operation, has a significant influence on embrittlement during a postulated LOCA. Increased hydrogen content increases both the solubility of oxygen in zirconium and the rate at which it is diffused within the metal, thus increasing the amount of oxygen in the metal during high temperature oxidation in LOCA conditions. Further, the NRC's research program found that oxygen from the oxide fuel pellets enters the cladding from the inner surface if a bonding layer exists between the fuel pellet and the cladding, in addition to the oxygen that enters from the oxide layer on the outside of the cladding. Moreover, under some small-break LOCA conditions (such as extended time-at-temperature around 1,000 degrees Celsius (°C) (1832 degrees Fahrenheit (°F))), a phenomenon termed breakaway oxidation can take place, allowing large amounts of hydrogen to diffuse into the cladding, exacerbating the embrittlement process. Breakaway oxidation is defined as the fuel cladding oxidation phenomenon in which weight gain rate deviates from normal kinetics. This change occurs with a rapid increase of hydrogen pickup during prolonged exposure to a high temperature steam environment, which promotes lack of ductility.

The research results also confirmed a previous finding that if cladding rupture occurs during a LOCA, large amounts of hydrogen from the steam-cladding reaction can enter the cladding inside surface near the rupture location. 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 (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. When the NRC publicly released NUREG/CR-6967, the NRC published in the FR 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 firms, fuel vendors, and the general public. A summary of the workshop, including a list of attendees and presentations, is available at ADAMS 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 in the 2008 public workshop), because: i) the public workshop was held, in part, to discuss public comments on the technical basis information, and ii) further opportunity to comment is available during the proposed rule's formal public comment period.

Based upon a preliminary safety assessment in response to the research findings in RIL-0801, the NRC 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 completed a more detailed safety assessment which confirmed current plant safety for every operating reactor. See Section III, "Operating Plant Safety," of this document for further information.

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.

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	ML 052230093*
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*

		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

*ADAMS file is a transcript of the ACRS meeting.

PRM-50-84

On March 15, 2007, Mark Leyse submitted a PRM to the NRC (ADAMS Accession No. ML070871368). In the petition, which was docketed as PRM-50-84, the petitioner requests 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[2] and/or oxide layers on fuel rod cladding surfaces. The petitioner requests 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

2 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.

zirconium-clad fuel in order to ensure compliance with § 50.46(b) ECCS acceptance criteria;

2) Amend Appendix K to Part 50 to explicitly require that steady-state temperature distribution and stored energy in the reactor fuel at the onset of a postulated 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 need to apply to any NRC-approved, best-estimate ECCS evaluation models used in lieu of Appendix K to 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 FR (72 FR 28902) and requested public comment. The public comment period ended on August 6, 2007. 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 NRC's response to public comments received on the petition, was published in the FR on November 25, 2008 (73 FR 71564). Although there is no direct relationship between the subject of crud and the anticipated new ECCS acceptance criteria requirements, the petition deals with 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 considering the petitioner's proposed changes in this rulemaking.

III. Operating Plant Safety

In response to the research findings in RIL-0801, the NRC performed a preliminary safety assessment of currently operating reactors (ADAMS Accession No. ML090340073). This assessment found that, due to realistic fuel rod power history, measured cladding performance under LOCA conditions, and current analytical conservatisms, sufficient safety margin 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 can 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 Pressurized Water Reactor (PWR) Owners Group (ADAMS Accession No. ML11139A309) and Boiling Water Reactor (BWR) Owners Group (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 evaluation models and defining cladding alloy-specific analytical limits, the Owners Group (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, for every operating reactor, current safe operation. As documented in the audit report and safety assessment (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. See Section V.E, “Implementation,” of this document for the staff recommended implementation plan developed based on this information.

IV. Advance Notice of Proposed Rulemaking: Public Comment

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 which 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 solicited responses to 12 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 at ADAMS Accession No. ML101300490.

As a result of comments received on the ANPR, a number of changes were made to the proposed rule package, as described below. The NRC's detailed discussion of the public comments submitted on the ANPR, including a detailed list of commenters, is contained in a separate document, "§ 50.46c and PRM-50-71 Comment Response Document" (ADAMS

Accession No. ML112520303). The most significant changes that were made to the preliminary draft rule in the ANPR as the result of public comments are:

- The specific experimental technique for measuring cladding ductility (i.e., $\geq$ 1.00 percent permanent strain prior to failure during ring-compression loading at a temperature of 135 °C and a displacement rate of 0.033 millimeters per second (mm/sec)) was removed from the rule and provided as one approved method within draft regulatory guide (DG)-1262, "Testing for Postquench Ductility," (ADAMS Accession No. ML110840283).
- The specific experimental technique for measuring time until breakaway oxidation (i.e., hydrogen uptake reaches 200 weight part per million (wppm) anywhere on a cladding segment subjected to high temperature steam oxidation ranging from 1200 °F to 1875 °F (649 °C to 1024 °C)) was removed from the rule and provided as one approved method within DG-1261, "Conducting Periodic Testing for Breakaway Oxidation Behavior," (ADAMS Accession No. ML110840089).
- The proposed risk-informed change to the reporting requirements (objective three of the ANPR) was abandoned. The majority of public comments received on the proposed reporting criteria suggested that the concept was complex, and might promote unnecessary burden or misinterpretation.
- The applicability of the zirconium-based alloy fuel specific performance requirements was expanded to include uranium-plutonium mixed oxide fuel.
- The applicability of the PQD analytical limits in DG-1263, "Establishing Analytical Limits for Zirconium-Based Alloy Cladding," (ADAMS Accession No. ML110871607) was expanded to encompass cladding hydrogen concentration up to 800 wppm.
- Many changes and improvements were made in the development of DG-1261, DG-1262, and DG-1263.

- A staged implementation plan was developed.

V. Proposed Requirements for ECCS Performance During LOCAs

The proposed rule would establish a general, performance-based rule governing ECCS performance for light-water nuclear power reactors (LWR), regardless of fuel design or cladding material. This represents a significant change from the current ECCS regulations, which apply to "uranium oxide pellets within cylindrical zircaloy or ZIRLO™ cladding." Because ECCS system requirements must be expressed independent of fuel type, and because ECCS system performance ultimately must be based upon maintaining the fuel used in a safe (analyzed) condition, the new rule separates the ECCS system requirements from the need for the applicant/licensee to establish the fuel system design performance criteria constituting a safe condition.

In proposed § 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 current regulations, the ECCS performance is demonstrated by NRC-approved evaluation models in proposed § 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 which account for recent research findings. New performance objectives and analytical limits may be necessary for other fuel designs to take into consideration all degradation mechanisms and any unique features of the particular fuel system for which the ECCS is trying to cool.

The proposed rule follows the general regulatory approach of the existing regulations by establishing non-prescriptive, performance-based regulatory language for demonstrating acceptable ECCS system performance and determining the fuel's performance characteristics.

However, because the embrittlement criteria in the current regulations for fuel with zirconium-based cladding continue to be acceptable (although incomplete, as will be discussed) the proposed rule retains the current regulations' 2200 ^{0}F limit for fuel with zirconium-based cladding as well as limitations on oxidation and hydrogen generation.

The organization and CFR designations of the NRC's requirements governing ECCS (currently in § 50.46) and reactor cooling venting systems (currently in § 50.46a) are expected to change, as a result of: 1) ongoing rulemaking activities; 2) the proposed implementation schedule for those activities; and 3) the need to maintain the current requirements in place for those licensees that have not transitioned to the new requirements (following the implementation schedule which would be provided in the final rule). A detailed description of the transition of CFR designations is provided in Section VI, "Section by Section Analysis."

A. Applicability of Performance-Based Rule; Consideration of PRM-50-71

The NRC proposes to expand the applicability of the rule from "uranium oxide pellets within cylindrical zircaloy or ZIRLO^{TM} cladding" to any LWR, regardless of fuel design or cladding material. The proposed rule would be applicable to applicants for and holders of construction permits, operating licenses, combined licenses, and standard design approvals and by applicants for certified designs and for manufacturing licenses. The only exception to the rule's applicability would be for any licensee which has submitted certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel, in accordance with § 50.82(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 zirconium-based alloy cladding and, more recently, to licensees using approved fuel designs with Optimized ZIRLO^{TM} zirconium-based alloy cladding.

The proposed 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 through a rulemaking. However, the licensee would still need to submit a license amendment. During this approval process, either: 1) specified and NRC-approved analytical limits have been established, along with an NRC-approved 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 performance objectives and associated analytical limits which take into consideration all degradation mechanisms and any unique features of the particular fuel system have been established, along with an NRC-approved evaluation model, by which to judge the ECCS performance for new fuel designs.

The NRC recognizes that a small number of fuel rods may fail 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 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 specified and 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 proposed extension to all LWR fuel types addresses an NEI petition for rulemaking (PRM-50-71) dated March 14, 2000, as amended to on April 12, 2000, 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}." If the NRC adopts the proposed rule in final form, then PRM-50-71 would be granted and resolved.

B. Performance-Based Aspects of the Proposed Rule

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 principle ECCS performance requirements are simple in nature (i.e., remove residual heat and maintain core temperatures at acceptable levels), the system capabilities and capacities must be designed based on specified performance objectives taking into consideration all degradation mechanisms and any unique features of the particular fuel system for which the ECCS is trying 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 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 which impact the ECCS performance demonstration (e.g., fuel rod balloon and burst, cladding inside-diameter oxygen ingress).

To achieve the rulemaking objective of developing a more performance-based rule, significant changes in format and structure are being proposed relative to § 50.46. In place of the current prescriptive § 50.46(b) analytical limits, the proposed rule would define the following principle 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.
- 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.

In addition, the proposed rule would dictate specific analytical requirements for demonstrating compliance to the ECCS performance requirements. For instance, to demonstrate compliance with these system performance requirements, fuel-specific performance objectives and associated analytical limits which take into consideration all degradation mechanisms and any unique features of the particular fuel system would be established, along with an NRC-approved evaluation model, by which to judge the ECCS performance.

The proposed rule includes specific performance requirements for fuel designs consisting of uranium oxide or mixed uranium-plutonium oxide fuel pellets within cylindrical zirconium-alloy cladding by which to judge ECCS performance. 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 degradation mechanisms and unique features have been identified, specific performance objectives defined, and fuel design specific performance requirements have been established and included in the proposed rule. For this fuel system design, the performance objective is to maintain the coolable fuel rod bundle array. In other words, the objective is to maintain fuel pellets within its cladding and fuel rods within the fuel bundle lattice. Existing 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), and 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 following § 50.46(b) requirements would remain unchanged in the proposed § 50.46c:

- *Peak cladding temperature*. The calculated maximum fuel element cladding temperature shall not exceed 2200 °F. The peak cladding temperature requirements currently in § 50.46(b)(1) would be moved to § 50.46c(g)(1)(i).
- *Maximum hydrogen generation*. 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) would be moved to § 50.46c(g)(1)(iv).

In the current 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 which demonstrated that the current combination of peak cladding temperature (2200 °F (1204 °C)) and local cladding oxidation (17 percent equivalent cladding reacted (ECR)) criteria may not always ensure PQD. The impact of these research findings on cladding ductility is addressed in the following section.

1. Hydrogen-Enhanced Beta-Layer Embrittlement:

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, which provides its overall ductility. The presence of hydrogen within the cladding enhances this embrittlement process.

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 current provision limits maximum local oxidation to preserve cladding ductility. Maximum local oxidation is used as a surrogate to limit the integral time-at-temperature (ITT) 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 the recent LOCA research program, the Cathcart-Pawel (CP) weight gain correlation was used to integrate time-at-temperature and define the point at which ductility was lost (nil ductility). Section 1.3 of NUREG/CR-6967 defines the following equations used to integrate time-at-temperature:

$ECR_{\text{One-sided oxidation}} = 43.9\ [(Wg/h)/(1-h/Do)]$, (Eqn. 5 of NUREG/CR-6967)

$ECR_{\text{Two-sided oxidation}} = 87.8\ (Wg/h)$, (Eqn. 6 of NUREG/CR-6967)

where ECR is in percent, Wg is in g/cm^2, h is cladding thickness in cm, and Do is cladding outside diameter in cm. The CP weight gain correlation (Wg) is defined as follows.

$Wg = 0.602\ \exp(-1.005\text{x}10^{-4}/T)t^{1/2}$ (Eqn. 4 of NUREG/CR-6967)

where Wg is given in g/cm^2, T is temperature in Kelvin, and t is time in seconds.

Measurements of weight gain were performed on many of the steam-oxidized cladding samples tested in the LOCA research program. For example, Table 22 of NUREG/CR-6967 provides both measured ECR and calculated CP-ECR for the zircaloy-2 cladding samples tested. Instead of correlating measured plastic strain or measured offset displacement with measured ECR or measurements of the post-quench cladding microstructure (e.g., beta layer thickness), the research findings correlate the ductile-to-brittle transition to calculated CP-ECR (using the equations previously stated). In this instance, calculated ECR is used to integrate time-at-temperature and does require knowledge of measured ECR. However, an accurate or conservative weight gain model based on measured oxidation, which may be alloy-specific or vary significantly from CP predictions, needs to be used for predicting rate of energy release and hydrogen generation from the metal/water reaction in the LOCA heat balance calculation.

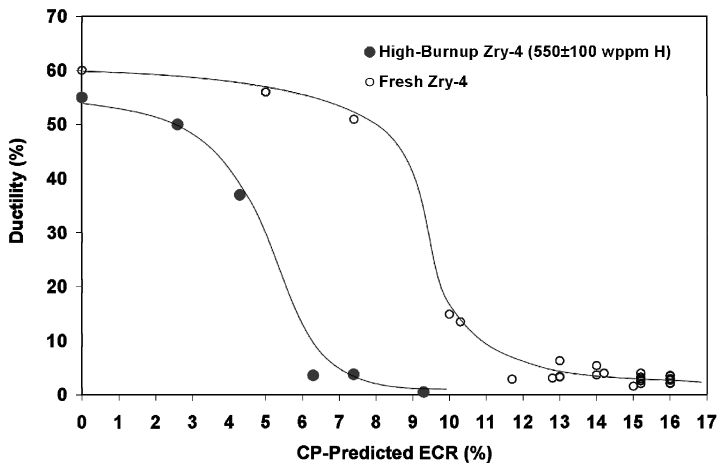
In an attempt to more accurately characterize the degrading phenomenon, the proposed rule would replace the term maximum local oxidation with ITT, which more directly relate to the parameter of interest (i.e., embrittlement due to oxygen diffusion). This should clarify the need to have: 1) an accurate or conservative weight gain correlation based on measured oxidation for estimating the rate of energy release and hydrogen generation from the metal/water reaction, and 2) a consistent analytical technique to integrate time-at-temperature in both the empirical database (i.e., allowable CP-ECR) and evaluation model (i.e., predicted CP-ECR).

During normal operation, the cladding metal absorbs some hydrogen from the corrosion process. When that cladding is exposed to high-temperature LOCA conditions, the elevated hydrogen levels increase the solubility of oxygen in the beta phase and the rate of diffusion of oxygen into the beta phase. Thus, even for LOCA temperatures below 1204 °C (2200 °F), embrittlement can occur for time periods corresponding to less than 17-percent oxidation in corroded cladding with significant hydrogen pickup.

Figure 1 illustrates the effect of hydrogen on ring-compression test ductility measurements. Test specimens included high-burnup (a 71- to 74-micrometer corrosion-layer thickness) and as-fabricated (fresh) PWR Zircaloy-4 cladding segments. Cladding samples were oxidized on two sides at approximately 1200 °C (~ 2200 °F) and cooled at approximately 11 ^{0}C per second to 800 °C (1472 ^{o}F). As-fabricated samples were quenched at 800 °C, whereas the high-burnup samples were slow-cooled from 800 °C to room temperature.

FIGURE 1: Measured Offset Strains

(Source: NUREG/CR-6967)



To address this phenomenon (as well as to achieve a more performance-based rule), the NRC proposes to replace the existing prescriptive analytical limits with a performance-based requirement which would require licensees to establish specified and NRC-approved analytical limits on peak cladding temperature (PCT) and ITT. These limits should correspond to the measured ductile-to-brittle transition for the zirconium-based alloy cladding based upon an NRC-approved experimental technique. If the peak cladding temperature which preserves cladding ductility is lower than the 2200 °F limit, the licensee should use the lower temperature.

The NRC is issuing draft regulatory guide DG-1263 (ADAMS Accession No. ML110871607) for comment. The draft regulatory guide provides licensees with "specified and NRC-approved analytical limits on PCT and integral time at temperature (ITT)," based upon the

NRC's LOCA research program's measured ductile-to-brittle transition for zirconium-based alloy cladding. In addition, the NRC is issuing DG-1262 (ADAMS Accession No. ML110840283) for comment, which provides licensees with "an NRC-approved experimental technique" for conducting PQD measurements and developing analytical limits. These draft regulatory guides specify an approach acceptable to NRC. Even if the draft regulatory guides are adopted in final form, licensees may propose alternative approaches to those described in those regulatory guides.

It is important to recognize that a consistent integration technique should be used to quantify time at elevated temperature in both the experiments and evaluation model. For example, the NRC-approved analytical limits on ITT in DG-1263 were based on the NRC LOCA research program results which, in turn, integrated time at elevated temperature using the CP weight gain correlation. For consistency with DG-1263, future LOCA analyses must integrate time at elevated temperature using the same CP weight gain correlation when comparing against these analytical limits. For this case, Appendix K evaluation models would continue to use the Baker-Just (BJ) weight gain correlation for estimating the rate of energy release and hydrogen generation from the metal/water reaction.

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 flow. As a result, the existing 2200 °F limit (specified in § 50.46c(g)(1)(i) of the proposed rule) remains an absolute upper limit for zirconium-based alloys on peak cladding temperature. However, as reflected in this proposed requirement, a lower peak cladding temperature may be required to preserve ductility.

2. Oxygen Ingress from Cladding Inside Diameter:

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.

To address this phenomenon, the NRC proposes to add an analytical requirement to the evaluation model which would require licensees to, if an oxygen source is present on the inside surfaces of the cladding at the onset of LOCA, consider the effects of oxygen diffusion from the cladding inside surfaces in the evaluation model.

The NRC recognizes that the availability of a cladding ID 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, the burden of 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 evaluation model, resides with the applicant. It is anticipated that identifying the magnitude and onset of oxygen ID diffusion would be part of the NRC's review and approval of LOCA evaluation models or vendor fuel designs. A conservative analytical limit is provided in draft regulatory guide DG-1263 (ADAMS Accession No. ML110871607).

3. 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 oxide phase transformation when exposed to long durations of high temperature steam oxidation, alloying composition and manufacturing process (e.g., surface roughness) influence the timing of this phenomenon.

Any fuel rod which 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 general design criterion (GDC) 35 requirements for coolable core geometry. To address this phenomenon, the NRC proposes to add a performance-based requirement that the licensee measure the onset of breakaway oxidation for each reload batch on manufactured cladding material and to report any changes in the onset of breakaway oxidation at least annually. This requirement, along with a periodic test requirement, would confirm that slight composition changes or manufacturing changes have not inadvertently altered the cladding's susceptibility to oxidation. The NRC is issuing DG-1261 (ADAMS Accession No. ML110840089), which will provide licensees with "an NRC approved experimental technique" for conducting breakaway oxidation measurements and developing analytical limits. Even if the draft regulatory guide is finalized, licensees may also provide an alternative approach to that proposed in the draft regulatory guide.

4. Applicability 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 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., 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.

Consistent with the technical basis of the proposed rule, draft Regulatory Guide DG-1262 describes an NRC-approved experimental technique for defining the ductile-to-brittle transition. This experimental procedure involves measuring ductility using ring compression testing performed on small, unflawed segments of fuel rod cladding previously exposed to steam oxidation at a defined peak cladding temperature and the integrated time at temperature profile (expressed as CP-ECR). While this experimental approach captures embrittlement of the zirconium metal due to oxygen diffusion and the effects of pre-existing hydrogen on the rate of embrittlement, it does not capture all of the degradation mechanisms experienced in the region of the fuel rod surrounding a cladding rupture. In addition to embrittlement due to oxygen ingress (which is doubled in the burst region due to steam entering cladding rupture), the burst region experiences cladding wall thinning, cladding rupture, and increased hydrogen uptake (hydrogen absorbed from zirconium oxidation on the cladding ID). All of these degradation mechanisms impact the performance of the fuel rod under LOCA conditions. As such, the ductile-to-brittle transition based on ring compression tests of unflawed cladding segments may not fully represent the region of the fuel rod surrounding the cladding rupture.

The rupture region contains non-uniform distributions of: 1) oxygen concentration within the base metal and zirconium oxide thickness, 2) soluble hydrogen and zirconium hydrides, 3) cladding wall thickness (due to ballooning), and 4) cladding flaws (due to ballooning and rupture). The overall goal of preserving cladding ductility may not apply to the rupture area that contains non-uniform distributions of flaws, cladding thickness, hydrogen distribution, and oxygen levels.

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 report “Mechanical Behavior of Ballooned and Ruptured Cladding” (ADAMS Accession No. ML12048A475). The integral LOCA testing confirms that continued exposure to high temperature steam environment weakens the already flawed region of the fuel rod surrounding the cladding rupture. Hence, limitations on peak cladding temperature and integral time at temperature are necessary to preserve an acceptable amount of mechanical strength and fracture toughness. In addition, this 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 which preserve sufficient fracture toughness to ensure burst region survival), or 2) apply the ductility-based analytical limits to the entire fuel rod.

In the absence of a credible analysis of loads, cladding stresses, and cladding strains for a degraded LOCA core, there are no absolute metrics to determine how much ductility or strength would be needed to "guarantee" that fuel-rod cladding would maintain its geometry during and following LOCA quench. It is also not clear what impact severance of some fuel rods into two pieces 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. If wall thinning and double-sided oxidation are accounted for, then it was determined that applying the hydrogen-based embrittlement limit developed in previous work at ANL to limit oxidation in the balloon region of the irradiated fuel rods tested at Studsvik was sufficient to preserve reasonable behavior of the ballooned and ruptured region.

The integral LOCA research concluded that application of the hydrogen-dependant ductility-based analytical limits on PCT and ITT (when applied within the burst region) preserve the mechanical behavior of high-burnup rods tested to that measured for as-fabricated cladding oxidized to 17 percent CP-ECR. Assuming highly conservative upper bounds on thermal expansion loading during quench, the residual mechanical behavior preserved by this limit was determined to be adequate to demonstrate that coolable geometry is maintained. As such, 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 decision is reflected in DG-1263 and supported by the technical basis documented in the staff report, "The Mechanical Behavior of Ballooned and Ruptured Cladding," (ADAMS Accession No. ML12048A475).

5. Long-Term Cooling:

The current regulation in § 50.46(b)(5) 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.

The proposed rule would define a performance-based requirement to ensure acceptable fuel performance during long-term cooling. Specifically, the proposed rule would require a specified and NRC-approved analytical limit on peak cladding temperature be established which corresponds to the measured ductile-to-brittle transition for the zirconium-based alloy cladding material based upon an NRC-approved experimental technique. It would also require that the calculated maximum fuel element temperature should not exceed the established analytical limit.

C. Reporting Requirements

The ANPR identified the third objective of the rulemaking as the revision of the LOCA reporting requirements. Specifically, the ANPR indicated that the NRC considered revising the reporting criteria by redefining what constitutes a significant change or error in such a manner as to make the reporting requirements dependent upon the margin between the acceptance criteria limits and the calculated values of the respective parameters (i.e., PCT or CP-ECR). After reviewing the public comments received, the NRC recognizes that the proposed reporting requirements specified in the ANPR were complex, and might, as a result, promote unnecessary burden or misinterpretation. As such, the reporting requirements of this proposed rule would not incorporate a dependence on margin between the acceptance criteria and calculated parameters.

The proposed rule would add a reporting requirement and definition of significant change or error based on predicted changes in maximum local oxidation (i.e., ECR), reformat the reporting section to clarify existing requirements, and add a reporting requirement based on periodic breakaway oxidation measurements. Any changes or errors which prolong the temperature transient may further challenge the integral time-at-temperature analytical limit; however, they may not significantly change the predicted PCT. As such, this change or error would not be captured in the reporting requirements. To improve the reporting and evaluation of changes or errors of this type, the NRC would expand the definition of significant change or error to include maximum local oxidation. The threshold for significant, 0.4 percent ECR would be 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 proposed rule would require the use of maximum local oxidation (i.e., % ECR) to evaluate the impact of a change or error on the predicted integral time-at-temperature.

Reporting requirements with respect to any "change to or error discovered in an NRC-approved evaluation model or in the application of such a model" have been a source of confusion. Two common misconceptions are: 1) baseline values when estimating a significant change or error (i.e., greater than 50 °F), and 2) 30-day reporting including "a proposed schedule for providing a reanalysis." When estimating a significant change or error, the proposed rule provides threshold values for both peak cladding temperature and local oxidation. The baseline predictions used to assess a significant change or error should be the peak cladding temperature and maximum local oxidation values documented in a plant's updated final safety analysis report (UFSAR). These values should represent the latest LOCA analyses which were submitted and reviewed by the NRC staff as part of a license amendment request (e.g., power uprate, fuel transition) 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 peak cladding temperature (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 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 significant. At this point, the licensee should 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.

When a change to or error in an evaluation 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 proposed 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, error or operation which does not result in any predicted response which exceeds any acceptance criteria and is itself not significant*.

The licensee must:

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

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

The licensee must:

   a. Submit a 30-day report documenting the change(s), error(s), or operation, estimated magnitude of changes in predicted results, and the schedule for providing a new AOR. The NRC will review the new AOR.
   b. Revise UFSAR to include new AOR.
   c. Use the UFSAR PCT/ECR predictions as a baseline for the future evaluations.

3. *Change, error or operation which results in any predicted response to exceed acceptance criteria*.

The licensee must:

   a. Take immediate actions to bring plant into compliance with acceptance criteria.
   b. Report the change, error or operation under §§ 50.55(e), 50.72, and 50.73, as applicable.
   c. Submit a 30-day report documenting the change(s), error(s), or operation, estimated magnitude of changes in predicted results, and the schedule for providing a new AOR. The NRC will review the new AOR.

d. Revise UFSAR to include new AOR.

e. Use the UFSAR PCT/ECR predictions as a baseline for the future evaluations.

The proposed reporting paragraph (m) reflects reformatting of the current reporting provisions in order to separately identify these three scenarios and clarify their respective requirements.

The proposed rule would also add the requirement to report results of breakaway oxidation measurements to the NRC. The licensees would be required to measure breakaway oxidation prior to each reload batch, and report the measurements within the calendar year following the testing. The breakaway oxidation phenomenon is explained in detail in sub-section B.3, “Breakway Oxidation” of this section, “Proposed Requirements for ECCS Performance During LOCAs.” This reporting requirement would be specific to zirconium-alloy cladding and may not be applicable to other cladding materials.

D. Consideration of PRM-50-84: Thermal Effects of Crud and Oxide Layers

Determination of PRM

This proposed rule would address issues raised in a PRM which was submitted on March 15, 2007, and docketed as PRM-50-84. The petition requests that the NRC conduct rulemaking in 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 Part 50 to explicitly require that the steady-state temperature distribution and stored energy in the reactor fuel at the onset of the postulated 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 need to apply to any NRC-approved, best-estimate ECCS evaluation models used in lieu of Appendix K to 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 (72 FR 29802), the NRC published a notice of receipt for this petition in the FR and requested public comment on the petition. The public comment period ended on August 6, 2007. After evaluating the public comments, the NRC decided that each of the petitioner's issues should be considered in the rulemaking process. On this basis, the NRC closed the docket on the petition for rulemaking. The NRC's determination, and evaluation of public comments received, was published in the FR on November 25, 2008 (73 FR 71564).

Technical Issues in PRM-50-84

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 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 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 evaluation model, licensees would 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, engineering judgment is used to determine that new analyses would be performed to determine the effect the new crud conditions have on the final calculated results. 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. Should proposed rule be adopted in final form, the NRC believes this regulatory approach to address crud and oxide accumulation during plant operation would satisfactorily address 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 models properly account for the thermal effects of crud and/or oxide layers that have accumulated during operations at power, the proposed rule would add a requirement to evaluate the thermal effects of crud and oxide layers that may have accumulated on the fuel cladding during plant operation. If the NRC adopts the proposed rule in final form, then the second request of PRM-50-84 would be resolved.

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 has decided not to propose the specific rule language recommended by petitioner, the proposed new zirconium-specific requirements, if adopted in final form, would 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 proposed rule addresses each of the three issues raised in PRM-50-84. If the NRC adopts the proposed rule in final form, then PRM-50-84 would be granted in part and resolved.

E. Implementation

The proposed rule would specify the date for compliance with the rule for existing operating license holders as well as holders of new reactor construction permits, combined licenses, and applicants for standard design certifications. For existing operating nuclear power reactors, this includes a staged schedule for implementation. The NRC has developed this staged implementation to improve the efficiency and effectiveness of this migration toward the new ECCS requirements for the existing operating fleet. As part of this plan, licensees have been divided among three implementation tracks based upon existing margin to the revised requirements and anticipated level of effort to demonstrate compliance. The purpose of the staged implementation approach is to bring licensees into compliance as quickly as possible, while accounting for: 1) differences between realistic and Appendix K LOCA models; and 2) the level of effort and scope of analyses required for compliance. Table 1 provides an overview of the implementation schedule for the existing fleet. Note that the compliance schedule requirement represents the date that the licensee submits either the letter report or license amendment request (as opposed to the date of NRC approval). The proposed track assignments for every operating reactor is provided in Table 1 of proposed § 50.46c(o)(10).

Table 1: Implementation Plan

Implementation Track	Basis	Anticipated Level of Effort	Number of Plants		Compliance Demonstration
			BWR	PWR	
1	All plants which satisfy new requirements without new analyses or model revisions.	Low	27	38	No later than 24 months from effective date of rule
2	PWR plants using realistic LBLOCA models requiring new analyses. BWR/2 plants.	Medium	2	14	No later than 48 months from effective date of rule
3	PWR plants using Appendix K LB and SB models requiring new analyses. BWR/3 plants.	Medium - High	6	17	No later than 60 months from effective date of rule

To support the implementation of the proposed requirements on individual plant dockets, fuel vendors would be encouraged to submit for NRC review alloy-specific hydrogen uptake models and any LOCA model updates (e.g., incorporation of CP weight gain correlation) no later than 12 months from the effective date of the final rule. Upon approval, these models and methods could be used to demonstrate the ECCS performance against the new analytical limits.

For Track 1 plants which would not require new ECCS evaluations, licensees should complete any necessary engineering calculations, update their plant UFSAR, and provide a letter report to the NRC documenting compliance with § 50.46c. The NRC recognizes that to demonstrate compliance, these plants would need to utilize newly-approved hydrogen uptake models and integrate time at temperature using the CP weight gain correlation (for Appendix K models).

For any plant which would require a new ECCS evaluation, including adopting a previously approved realistic evaluation model, revisions to existing evaluation model, new LOCA break spectrum analysis, multiple rod survey (e.g., BU – rod power tradeoff), technical specification or core operating limit report (COLR) changes, licensees would need to submit the new LOCA AOR and, where applicable, a license amendment request updating COLR list of approved methods.

The NRC has developed a phased implementation approach for applicants and holders of standard design approvals, design certifications, combined operating licenses and manufacturing licenses granted under Part 52.

The proposed implementation plan for reactors approved under Part 52 would allow the applicant for a design certification, standard design approval, or manufacturing license either submitted to or docketed by the NRC prior to the effective date of the rule, to come into compliance with the rule at the time of any application for renewal.

An applicant for a design certification, standard design approval, or manufacturing license submitted or docketed after the effective date of the rule must comply with the provisions of the rule.

The holder of a combined license granted prior to the effective date of the rule would be permitted to operate the plant for one fuel cycle before demonstrating compliance with the rule. Doing so would permit adequate time to submit demonstration of compliance with the rule prior to achieving fuel burnup for which the cladding limitations are imposed by the rule. In this case the holder of the combined operating license would be required to remain in compliance with the ECCS performance acceptance criteria in place at the time the combined operating license was granted.

Applicants for combined licenses docketed after the effective date of the rule must comply with the provisions of the rule.

The proposed 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 proposed rule also reflects the NRC's expectation that, for new reactors licensed to operate prior to the effective date of the rule, operation at least the initial fuel cycle using fuel which has not been analyzed under the proposed 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 current 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.

VI. Section-by-Section Analysis

The organization and CFR designations of the NRC's requirements governing emergency core cooling (currently in § 50.46) and reactor cooling venting systems (currently in § 50.46a) are expected to change. These changes would result from:

1) The current schedule for Commission serial adoption of two rulemakings: i) the finalization of the proposed rule on risk-informed changes to ECCS systems, currently referred to as the § 50.46a rulemaking, followed by; ii) the finalization of this proposed

rule on performance-based changes to ECCS requirements and cladding acceptance criteria, currently referred to as the § 50.46c rulemaking;

2) The proposed schedule for implementation of these rules; and

3) The need to maintain current requirements in place for those reactors which have not transitioned to the new requirements under the implementation schedule to be specified in the final rule.

The following table ~~presents a graphic display of~~shows how the organization and CFR designation of these rules will evolve, if~~as~~ the ~~Commission~~ NRC sequentially adopts the two final rules ~~*seriatim*~~, and licensees complete implementation of the alternate cladding requirements. The NRC notes that, in an SRM 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 NRC does not plan to publish a notice in the *Federal Register* withdrawing the § 50.46a proposed rule. The NRC staff plans to resubmit the draft final rule for Commission consideration in conjunction with the Near-Term Task Force Recommendation 1 activities. Therefore, the § 50.46a rulemaking still may be finalized before the § 50.46c rulemaking, as assumed in the following table.

Existing NRC Requirements and Proposed New Regulations (Bolded rules are currently in effect)	**Rulemaking and Implementation Activities**		
	Adoption of Final Risk-Informed ECCS Requirements (§ 50.46a)	**Initial Codification of Final Performance-Based Fuel Cladding Requirements**	**End of phased implementation period for performance-based cladding requirements**
§ 50.46 ECCS Acceptance Criteria	§ 50.46 ECCS Acceptance Criteria (*unchanged*)	§ 50.46 ECCS Acceptance Criteria (*unchanged*)	§ 50.46 ECCS Acceptance Criteria (*see discussion for § 50.46c under this*

			column)
Risk-Informed ECCS Requirements *(currently designated in final rulemaking package as § 50.46a)*	§ 50.46a Risk-Informed ECCS Requirements	§ 50.46a Risk-Informed ECCS Requirements	§ 50.46a Risk-Informed ECCS Requirements
§ 50.46a Reactor Coolant Venting Systems	Redesignated as § 50.46b	NA (*Redesignation as § 50.46b completed*)	NA (*Redesignation as § 50.46b completed*)
Performance-based ECCS and Cladding Requirements *currently designated in draft proposed rulemaking package as § 50.46c*	NA	§ 50.46c Alternate Fuel Cladding Requirements	NA *(administrative rulemaking would: (i) remove superseded fuel cladding requirements in § 50.46; and (ii) redesignate § 50.46c as § 50.46.)*

A. Section 50.46c - Heading

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

B. Section 50.46c(a) - Applicability

Paragraph (a) would define the applicability of the proposed rule which remains limited to LWR, but would be expanded beyond fuel designs consisting of uranium oxide pellets within cylindrical zircaloy or ZIRLO™ cladding. The proposed rule would also be applicable to applicants for and holders of construction permits, operating licenses, combined licenses, and standard design approvals, and also to applicants for certified designs and for manufacturing licenses.

C. Section 50.46c(b) - Definitions

Paragraph (b) would provide definitions for terms used in this section. The definitions of *Loss-of-coolant accident* and *Evaluation model* would remain unchanged from those currently located in § 50.46(c)(1) and (c)(2), respectively.

The definition of *Breakaway oxidation* would be added.

D. Section 50.46c(c) – Relationship to other NRC regulations

Paragraph (c) would provide the relationship of § 50.46c to other NRC regulations. This relationship is the same as that of the current regulation found in § 50.46(d).

E. Section 50.46c(d) – Emergency core cooling system design

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

Paragraph (d)(2) would require that ECCS performance be demonstrated using an NRC-approved 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 model would remain unchanged from the current regulation found in § 50.46(a)(1)(i) and (ii), respectively. Similarly, the model requirement that calculated changes in core geometry must be addressed would remain unchanged from the current regulation found in § 50.46(b)(4). Paragraph (d)(2)(iii) would explicitly require that the evaluation model address calculated changes in core geometry, and consider factors which may alter localized coolant flow or inhibit delivery of coolant to the core. Demonstration of ECCS performance in the post-accident recovery period, or long-term cooling, is expected to consider inhibition of core flow that can result from such factors as, but not limited to, pump damage, piping damage, and deposition of debris and/or chemicals associated with the long-term cooling mode of recirculation coolant collection from the reactor building sump. Consideration of debris and/or chemical deposition is already required by the current rule, and the proposed rule does not alter the current efforts to address such factors under programs such as Generic Safety Issue (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.

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

Paragraph (d)(2)(v) would require that the ECCS model ~~must~~ 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) would provide the evaluation model documentation requirements currently provided in Appendix K, ~~Part~~ Section II, “Required Documentation.”

Paragraphs (e) and (f) would be added to reserve rulemaking space for future amendments to § 50.46c.

F. Section 50.46c(g) – Fuel system designs: uranium oxide or mixed uranium-plutonium oxide pellets within cylindrical zirconium-alloy cladding

This section would be 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) would establish 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 current requirement at § 50.46(b)(1).

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

Paragraph (g)(1)(iii) would be 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) would establish an analytical limit on maximum hydrogen generation to avoid an explosive concentration of hydrogen gas. This requirement would be the same as that of the current regulation in~~'s~~ § 50.46(b)(3).

Paragraph (g)(1)(v) would be added to establish a performance-based requirement to ensure acceptable fuel performance during long-term cooling. This performance requirement is consistent with the current requirement to "maintain the calculated core temperature at an acceptably low value" located in § 50.46(b)(5).

Paragraph (g)(2) would establish fuel design specific modeling requirements necessary in addition to the generic ECCS evaluation model requirements in paragraph (d)(2). Paragraph (g)(2)(i) would require consideration of oxygen diffusion from the cladding inside surface. This would be a new evaluation model requirement.

Paragraph (g)(2)(ii) would be 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) would be added to reserve rulemaking space for future amendments to § 50.46c, including any changes that stem from using newly designed fuel and cladding materials.

G. Section 50.46c(k) – Use of NRC-approved fuel in reactor.

Paragraph (k) would prohibit licensees from loading fuel into a reactor, or operate 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.

H. Section 50.46c(l) – Authority to impose restrictions on operation.

Paragraph (l) would provide 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 would be expanded, relative to the authority currently granted in § 50.46(a)(2), to address licenses issued under 10 CFR Part 52.

I. Section 50.46c(m) - Reporting.

Paragraph (m)(1) would establish required action and reporting requirements if a licensee identifies any change to or error in an evaluation model or the application of such a model, or any operation inconsistent with the evaluation model. For clarity, this paragraph was divided into three categories of changes or errors, each with its own proposed actions and reporting. These requirements are unchanged from the current § 50.46(a)(3), with the exception of conforming to analytical limits established in the proposed rule.

Paragraph (m)(1)(i) would establish required action and reporting requirements if a licensee identifies any change to or error in an evaluation model or the application of such a model, or any operation inconsistent with the evaluation model which does not result in any predicted response which exceeds any acceptance criteria and is itself not significant.

Paragraph (m)(1)(ii) would establish required action and reporting requirements if a licensee identifies any change to or error in an evaluation model or the application of such a model, or any operation inconsistent with the evaluation model which does not result in any predicted response which exceeds any acceptance criteria but is significant (as defined in paragraph (m)(2)).

Paragraph (m)(1)(iii) would establish required action and reporting requirements for~~if~~ a licensee who identifies any change to or error in an evaluation model.

Paragraph (m)(1)(iv) would require an amendment to the design certification application reflecting any reanalysis required by paragraph (m)(1)(ii) or (m)(1)(iii) be submitted by the applicant in concert with the reanalysis.

Paragraph (m)(2) would be added to provide the definition of a significant change or error. The definition would be expanded, relative to the 50 °F change in calculated peak cladding temperature in § 50.46(a)(3)(i), to include a 0.4 percent ECR change in calculated cladding oxidation.

Paragraph (m)(3) would require the onset of breakaway oxidation to be measured for each reload batch, and would require any changes in the time to the onset of breakaway oxidation to be assessed against the integral time and to be reported annually. This would be a new reporting requirement.

Paragraph (n) would be added to reserve rulemaking space for future amendments to § 50.46c.

J. Section 50.46(o) - Implementation.

This section would establish the implementation requirements and schedule for the existing fleet and for new reactors. Paragraph (o)(1) would require construction permits under Part 50 issued after the effective date of the rule to comply with the requirements of § 50.46c.

Paragraph (o)(2) would require operating licenses under Part 50 based upon construction permits (including deferred and reinstated construction permits) to comply with the requirements of § 50.46c by no later than the time frame established for operating reactors in the implementation table. Until that point, the construction permits identified by this paragraph must comply with § 50.46.

Paragraph (o)(3) would require operating licenses under Part 50 issued after the effective date of the rule to comply with the requirements of § 50.46c.

Paragraph (o)(4) would require operating licenses under Part 50 (as of the effective date of the rule) to comply with the requirements of § 50.46c by no later than the applicable dat~~ea~~ set forth in the implementation table for operating reactors.

Paragraph (o)(5) would require standard design certifications, standard design approvals, and manufacturing licenses under 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. 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 (o)(6) would require standard design certifications under 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 (o)(5), such applicants will have had ample time necessary to comply with the provisions of the rule.

Paragraph (o)(7) would require standard design certifications, standard design approvals, and manufacturing licenses, along with new branches of certifications under 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. Those entities that are in the approval process at the time the rule becomes effective will be required to comply at time of renewal. This will provide ample time to develop and receive approval for the methodologies necessary to comply with the rule. Paragraph (o)(8) would require combined licenses under Part 52 that are docketed after the effective date of the rule to comply with the provisions of the rule.

Paragraph (o)(9) would require applications for combined licenses under Part 52 that are docketed or issued after the effective date of the rule to comply with § 50.46c no later than completion of the first fueling outage after the initial fuel load. Those entities which are granted combined licenses prior to the effective date of the rule must comply with the rule no later than the first refueling outage after initial fuel load. This affords those entities ample time to develop and submit the necessary methodologies.

K. 10 CFR Part 50, Appendix K ECCS evaluation models.

~~Appendix KII.5 would be amended to conform to the proposed amendments discussed.~~In Appendix K, a new paragraph II.6 would be added to clarify that, for those entities that have implemented § 50.46c, the requirements for documentation are located within § 50.46c(d)(3).

L. Redesignation of Venting Requirements in Section 50.46a

This proposed rule would redesignate the current § 50.46a, "Acceptance criteria for reactor coolant system venting systems," as proposed § 50.46b. A new section 50.46a would be added and reserved for future use as the rulemaking to provide a risk-informed alternative to the LOCA technical requirements.

M~~L~~. Conforming changes throughout 10 CFR Parts 50 and 52

Several administrative changes would be made throughout 10 CFR Parts 50 and 52 in order to conform with the proposed rule and proposed redesignation of the venting requirements in current § 50.46a. Section 50.8 would be amended to add the proposed 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) refer to § 50.46, the proposed rule would add "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) refer to § 50.46a, the proposed rule would instead refer to § 50.46b.

VII. Specific Request for Comments on the Proposed Rule

In addition to the general opportunity to submit comments on the proposed rule, the NRC also requests comments on the following questions:

NRC Question 1. To address the breakaway oxidation phenomenon, the NRC proposes to add a performance-based requirement in § 50.46c(g)(1)(iii) that the licensee measure the onset of breakaway oxidation periodically on manufactured cladding material and to report any changes in the onset of breakaway oxidation at least annually. This requirement, along with a periodic test requirement (defined as each reload batch in the proposed rule language), would confirm that slight composition changes or manufacturing changes have not inadvertently altered the cladding's susceptibility to breakaway oxidation. The NRC is considering adopting, as a final rule, a requirement that each licensee measure breakaway oxidation behavior for each re-load batch. The NRC requests specific comment on the type of data reported and the proposed frequency of required testing. The objective of periodic testing is to prevent effected fuel from being loaded into a reactor. At the same time, the objective is to do so without adding ineffective and unnecessary burden. Therefore, other sampling approaches may be more effective. For example, should the licensee be required to report data relevant solely to their reload fuel batch or should the licensee be able to report representative data based on periodic testing (e.g., test every 10,000 rods, tubing lot, or ingot) of the same zirconium-based alloy cladding compiled during the period from the last report.

NRC Question 2. The NRC is proposing, in § 50.46c(o), a staged implementation plan for the proposed rule. As part of this plan, licensees have been divided among three implementation tracks based upon existing margin to the revised requirements and anticipated level of effort to demonstrate compliance. The NRC requests specific comment on the staged implementation plan, track assignments, or alternative means to implement the requirements of the proposed rule.

NRC Question 3. The NRC is proposing, in § 50.46c(o)(5) through (9), an implementation approach which takes into account various combinations of design certification, standard design approvals, manufacturing licenses and combined operating license and their status in relation to the effective date of the rule. The proposed implementation plan for new reactors would allow the applicant for a design certification, standard design approval, and manufacturing license under review at the time of the effective date of the rule to come into compliance with the rule at time of renewal. The holder of a combined operating license issued prior to the effective date of the rule would be permitted to operate the plant for one fuel cycle before coming into compliance with the rule. Thus, the NRC is proposing to recognize that new reactors may operate for the initial fuel cycle with fuel for which the burnup effects being accounted for in the rule would not be a consideration. Applications for design certifications, standard design approvals, manufacturing licenses and combined licenses submitted after the effective date of the rule would be expected to be in compliance with the rule at the time of approval.

The NRC is requesting input regarding this implementation proposal and if there is a simpler approach that could be taken.

NRC Question 4. Paragraph (g)(1)(v) of the proposed rule would require that a specified and NRC-approved limit on long-term peak cladding temperature be established which preserves a measure of cladding ductility throughout the period of long term demonstration (e.g., 30 days). The current regulation § 50.46(b)(5) stipulates that long-term temperature be maintained "at an acceptably low value." The proposed rule would define the performance-based metric to judge an acceptably low temperature. Ductility is a favorable material property and its preservation provides a degree of assurance that the fuel rods will maintain their coolable bundle array. The NRC is requesting input regarding this performance objective to determine if this is the most suitable performance-based metric to demonstrate long-term cladding performance.

Alternatively, the proposed rule could establish an analytical limit of long-term fuel rod cladding temperature related to observed corrosion behavior. For example, the Pressurized Water Reactor Owners Group (PWROG) has applied as a long-term core cooling acceptance criterion that the cladding temperature be maintained below 800 °F. Doing so will ensure that additional corrosion and hydrogen pickup over a 30-day period will not significantly affect cladding properties. Topical Report (TR) WCAP-16793-NP, Revision 1, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid," Appendix A (ADAMS Accession No. ML091190484). The NRC seeks comment on this acceptance criterion and if there is justification for a different temperature limit.

NRC Question 5. The NRC is considering the cumulative effects of regulation (CER) as it relates to this rulemaking. The CER consists of the challenges licensees face in addressing the implementation of new regulatory positions, programs, and requirements (e.g., rulemaking, guidance, generic letters, backfits, inspections). The CER stems from the total burden imposed on licensees by the NRC from simultaneous or consecutive regulatory actions that can adversely affect the licensee's capability to implement those requirements while continuing to operate or construct its facility in a safe and secure manner.

During the development of this proposed rulemaking, the NRC engaged external stakeholders through multiple public meetings, an ANPR, and public comments. Additionally, the proposed rule would establish a staged implementation plan which reduces overall implementation burden on licensees.

With regard to CER, the NRC requests specific comment on the proposed rule's implementation schedule in light of any existing CER challenges; specifically:

a. Does the proposed rule's effective date, compliance date, or submittal dates provide sufficient time to implement the new proposed requirements including changes to programs, procedures, and the facility, in light of any ongoing CER challenges?

b. If there are ongoing CER challenges, what do you suggest as a means to address this situation (e.g., if more time is required for implementation of the new requirements, what time period is sufficient)?

c. Are there unintended consequences (e.g., does the proposed rule create conditions that would be contrary to the proposed rule's purpose and objectives)? If so, what are the unintended consequences? Please comment on the NRC's cost and benefit estimates in the proposed rule regulatory analysis. Specifically, please comment on the vendor hydrogen uptake and LOCA model costs, costs of PQD and breakaway testing, and licensee analysis costs.

VIII. Request for Comment: Draft Regulatory Guidance

The NRC is seeking public comment on three regulatory guides: DG-1261, "Conducting Periodic Testing for Breakaway Oxidation Behavior" (ADAMS Accession No. ML110840089), DG-1262, "Testing for Post Quench Ductility" (ADAMS Accession No. ML110840283), and DG-1263, "Establishing Analytical Limits for Zirconium-Based Alloy Cladding" (ADAMS Accession No. ML110871607). You can access these documents as described in Section IX, "Availability of Documents," or online at http://www.nrc.gov/reading-rm/doc-collections/.

The proposed rule would add the 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 calls for the evaluation of the measurement relative to emergency core cooling system performance (*see* § 50.46c(g)(1)(iii)), and periodic testing and reporting of the values measured (*see* § 50.46c(m)(3)). DG-1261 describes 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 proposed rule would also require licensees 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 (*see* § 50.46c(g)(1)(i) and (ii)). DG-1262 describes an experimental technique that is acceptable to the NRC for measuring the ductile-to-brittle transition for a zirconium-based cladding alloy. DG-1263 provides a method of using experimental data to establish regulatory limits.

You may submit comments on the draft regulatory guides by the following methods:

- **Federal rulemaking Web site:** Go to http://www.regulations.gov and search for Docket IDs NRC-2012-0041, NRC-2012-0042, NRC-2012-0043, respectively. Address questions about NRC dockets to Carol Gallagher; telephone: 301-492-3668; e-mail: Carol.Gallagher@nrc.gov.

- **Mail comments to:** Cindy Bladey, Chief, Rules, Announcements, and Directives Branch (RADB), Office of Administration, Mail Stop: TWB-05-B01M, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

- **Fax comments to:** RADB at 301-492-3446.

IX. 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 requested amendment of 10 CFR Part 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
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 Embrittelemnet 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	
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 Breakaway Oxidation of Westinghouse ZIRLO Cladding	X	ML091350581	X
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	
TR WCAP 16793-NP, Revision 1, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous, and Chemical Debris in the Recirculating Fluid," Appendix A	X	ML091190484	
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	
§ 50.46c and PRM-50-71 Comment Response Document	X	ML112520303	
Regulatory Analysis	X	ML112520277	
Proposed Rule Information Collection Analysis	X	ML112520328	
Draft Regulatory Guide 1261, "Conducting Periodic Testing for Breakaway Oxidation Behavior"	X	ML110840089	
Draft Regulatory Guide 1262, "Testing for Post Quench Ductility"	X	ML110840283	
Draft Regulatory Guide 1263, "Establishing Analytical Limits for Zirconium-Based Alloy Cladding"	X	ML110871607	
Request to Withdraw 50.46a f~~F~~rom 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	

X. Criminal Penalties

For the purposes of Section 223 of the Atomic Energy Act of 1954, as amended (AEA), the NRC is issuing the proposed rule to amend § 50.46 under one or more sections of 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 Part 50 are discussed in § 50.111.

XI. 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 FR (62 FR 46517; September 3, 1997), this rule is classified as compatibility ANRC. Compatibility is not required for Category ANRC 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 States administrative procedure laws, but does not confer regulatory authority on the State.

XII. Plain Language

The Plain Writing Act of 2010 (Pub. L. 111-274) requires Federal agencies to write documents in a clear, concise, well-organized manner that also follows other best practices appropriate to the subject or field and the intended audience. Although regulations are exempt under the act, the NRC is applying the same principles to its rulemaking documents. Therefore, the NRC has written this document, including the proposed new and amended rule language, to be consistent with the Plain Writing Act. In addition, where existing rule language must be changed, the NRC has rewritten that language to improve its organization and readability. The NRC requests comment on the proposed rule specifically with respect to the clarity and effectiveness of the language used. Comments should be sent to the NRC as explained in the ADDRESSES section of this document.

XIII. 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 proposed 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.

XIV. 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 Part 51, that this rule, if adopted, would not be 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 proposed amendments would require licensees, in some cases, to submit an additional license amendment. The NRC 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:

Identification of the Action:

The proposed action is the amendment of the NRC regulation, § 50.46, which concerns the NRC's requirements for ECCSs for LWRs. The proposed amendment would establish performance-based requirements and also account 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 post-quench ductility (PQD). Further, the proposed amendment would expand the applicability of § 50.46 to all fuel design and fuel cladding materials. In addition, this proposed rule would address the issues raised in two PRMs (docketed as PRM-50-71 and PRM-50-84).

The Need for Action:

The proposed 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 the statements of consideration for this proposed rule, indicated that the current combination of peak cladding temperature (2200 °F (1204 °C)) and local cladding oxidation criteria do not always ensure post-quench ductility (PQD). The research also identified previously unknown embrittlement mechanisms. The proposed action would replace the limits on peak cladding temperature and local oxidation with specific cladding performance requirements and acceptance criteria which ensure that an adequate level of cladding ductility is maintained throughout the postulated LOCA.

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

Lastly, the proposed rule would require licensees to take into account the deposition of crud on the fuel cladding during plant operation. This change addresses PRM 50-84.

Environmental Impacts of the Proposed Action:

This environmental assessment focuses on those aspects of the proposed rulemaking in 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 proposed rule requirements for the following reasons:

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

2) The proposed amendments to the Part 50 ECCS requirements are unrelated to the systems, structures and components which mitigate the consequences of a LOCA. These proposed amendments, if approved, would revise and expand the performance requirements for which the ECCS response is judged. With these enhancements, the reactor core would remain coolable. Therefore, the consequences of a postulated LOCA are not changed by the proposed rule.

3) The proposed amendments to the Part 50 ECCS requirements would 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.

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

For the reasons discussed, the action will 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 would be no increase in occupational or public radiation exposure.

With regard to potential nonradiological impacts, the proposed rule would have no significant impact on the environment. The proposed rule to revise and expand the ECCS performance requirements would be applied by an NRC nuclear reactor power plant licensee to the restricted area of its facility only, and in many cases would 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 proposed amendments, if approved, would 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-Steven's Act. Similarly, it is extremely unlikely that there will be any impacts to socioeconomic, or to historic properties and cultural resources.

The proposed amendments would not affect the facility, structures, systems and components (SSCs) or operator actions. Therefore, there would be no significant nonradiological environmental impacts associated with the proposed action.

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

<u>Alternatives to the Proposed Action</u>:

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 current peak cladding temperature (PCT) and oxidation restrictions do not take into consideration newly discovered cladding embrittlement mechanisms, and that the current 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.

Alternative Use of Resources:

This action would 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 would be affected by this action.

Agencies and Persons Consulted:

The NRC staff developed the proposed rule and this environmental assessment. In accordance with its stated policy, the NRC provided a copy of the proposed rule to designated liaison officials for each state. No other agencies were consulted.

There appears to be no significant impact to human health or the environment from implementation of the proposed action. However, the general public should note that the NRC is seeking public participation. Comments on any aspect of the environmental assessment may be submitted to the NRC via e-mail to Rulemaking.Comments@nrc.gov or via mail to Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, ATTN: Rulemakings and Adjudications Staff.

The NRC has sent a copy of the environmental assessment and this proposed rule to every State Liaison Officer and requested their comments on the environmental assessment.

XV. Paperwork Reduction Act Statement

This proposed rule contains new or amended information collection requirements that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq). This proposed rule has been submitted to the Office of Management and Budget (OMB) for review and approval of the information collection requirements.

Type of submission, new or revision: Revision

The title of the information collection: 10 CFR 50.46c, Emergency Core Cooling System Performance During Loss-of-Coolant Accidents.

The form number if applicable: Not applicable.

How often the collection is required: LOCA model updates, License Amendment Requests, and compliance letters will be submitted one time during implementation; significant errors will be reported on occasion (within 30 days); other errors or changes in analysis will be reported annually.

Who will be required or asked to report: Fuel design vendors, all operating reactors, all applicants for or holders of construction permits, each applicant for an operating license, each applicant for or holder of a combined license, each applicant for a standard design certification, each applicant for a standard design approval, and each applicant for a manufacturing license.

An estimate of the number of annual responses: 301

The estimated number of annual respondents: 71 during the first three years of implementation;

a total of 115 will be impacted by the rule.

An estimate of the total number of hours needed annually to complete the requirement or request: 47,858 hours (48,058 hours reporting and -200 hours recordkeeping)

Abstract:

The NRC is proposing to amend its regulations to revise the acceptance criteria for the emergency core cooling system (ECCS) for light-water nuclear power reactors as currently required by 10 CFR Part 50. The rule would establish a five-year staged implementation approach to improve the efficiency and effectiveness of the migration to the new ECC requirements. As the first step, vendors will develop, and submit to the NRC for review via topical reports, hydrogen pickup models and LOCA model updates. The vendors would also obtain post-quench ductility (PQD) analytical methods by either selecting analytical limits provided in a regulatory guide, using an NRC-approved experimental approach, or using an experimental approach developed by the vendor. Those PQD limits developed via an experimental method would be submitted to the NRC via a topical report. The vendors would also perform long-term cooling tests to determine long term cooling limit for each of the nine cladding alloys. In addition, vendors would perform initial breakaway testing. The licensees would report the initial breakaway results to the NRC via their license amendment request. Those licensees that meet the new requirements without new analyses or model revisions would complete any necessary engineering calculations, update their plant UFSAR, and provide a letter report to the NRC documenting compliance. Those licensees which would require new analyses or model revisions to demonstrate compliance would be required to submit a new LOCA analysis of record. The rule would also require licensees to conduct periodic breakaway testing, and include those results in the yearly ECCS report. Lastly, the rule would add a requirement to report errors in ECR to the NRC. This would be submitted within the same yearly ECCS report.

The NRC is seeking public comment on the potential impact of the information collections contained in this proposed rule (or proposed policy statement) and on the following issues:

1. Is the proposed information collection necessary for the proper performance of the functions of the NRC, including whether the information will have practical utility?
2. Is the estimate of burden accurate?
3. Is there a way to enhance the quality, utility, and clarity of the information to be collected?
4. How can the burden of the information collection be minimized, including the use of automated collection techniques?

The public may examine and have copied, for a fee, publicly available documents, including the draft supporting statement, at the NRC's Public Document Room, One White Flint North, 11555 Rockville Pike, Room O-1 F21, Rockville, Maryland, 20852. OMB clearance requests are available at the NRC worldwide Web site: *http://www.nrc.gov/public-involve/doc-comment/omb/index.html*. The document will be available on the NRC home page site for 60 days after the signature date of this notice.

Send comments on any aspect of these proposed information collections, including suggestions for reducing the burden and on the previously stated issues, by **[INSERT DATE 30 DAYS AFTER PUBLICATION IN THE FEDERAL REGISTER]** to the Information Services Branch (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by Internet electronic mail to INFOCOLLECTS.Resource@nrc.gov and to the Desk Officer, Chad Whiteman, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0011), Office of Management and Budget, Washington, DC 20503. Comments received after this date will be considered if it is practical to do so, but assurance of consideration cannot be given to comments received after this date. You may also e-mail comments to Chad Whiteman at CWhiteman@omb.eop.gov or comment by telephone at (202) 395-4718.

Public Protection Notification

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

XVI. Regulatory Analysis: Availability

The NRC has prepared a draft regulatory analysis on this proposed regulation (ADAMS Accession No. ML112520277). The analysis examines the costs and benefits of the alternatives considered by the Commission. The NRC requests public comments on the draft regulatory analysis. Availability of the regulatory analysis is indicated in Section IX of this document. Comments on the draft analysis may be submitted to the NRC by any method provided in the **ADDRESSES** section of this document.

XVII. Regulatory Flexibility Certification

Under the Regulatory Flexibility Act (5 U.S.C. 605(b)), the Commission certifies that this rule would not, if promulgated, have a significant economic impact on a substantial number of small entities. This proposed rule affects light water nuclear power reactors. None of the companies that own and operate these facilities falls within the scope of the definition of “small entities” set forth in the Regulatory Flexibility Act or the size standards established by the NRC (§ 2.810).

XVIII. Backfitting and Issue Finality

Proposed 10 CFR 50.46c Rule

The proposed rule, 10 CFR 50.46c, would apply to current nuclear power plant licensees (including holders of renewed licenses). The proposed rule would apply to all current and future applicants for combined licenses under Part 52 regardless of fuel design or cladding material, including all current and future applicants for combined licenses under Part 52 that reference one of the existing standard design certification rules in Part 52, Appendices A through D. The proposed rule would apply to all current and future applicants for LWR standard design certification rules under Part 52. Finally, the proposed rule would apply to all future applicants for manufacturing licenses under Part 52 (there are no current applicants or holders of manufacturing licenses). Each of these classes of licenses and regulatory approvals is discussed in the following sections.

Operating Licenses

With respect to current nuclear power plant licensees, the NRC assumes that imposition of the proposed rule would constitute backfitting as defined in § 50.109(a)(1). However, the NRC believes that the proposed rule must be imposed upon current nuclear power plant licensees in order to ensure adequate protection to the public health and safety by restoring that level of protection (i.e., reasonable assurance of adequate protection) which the NRC thought would be achieved (throughout the entire term of licensed operation) by the current rule. Therefore, the NRC has determined that the proposed rule is necessary to ensure that the facility provides adequate protection to the health and safety of the public, and that a backfit analysis as described in §§ 50.109(a)(3) and (b) need not be prepared under the exception in

§ 50.109(a)(4)(ii).

Imposing the redefinition of fuel cladding acceptance criteria on current nuclear power plant licensees is justified under the provisions of § 50.109(a)(4)(ii) as the requirements of the proposed rule are necessary to ensure adequate protection to the public health and safety by restoring that level of protection (i.e., reasonable assurance of adequate protection) which the NRC thought would be achieved (throughout the entire term of licensed operation) by the current rule.

Information developed through the NRC's high burnup fuel research program has identified that the current criterion for preventing fuel cladding embrittlement may not be adequate to ensure the health and safety of the public. As discussed in Sections II and V of this Statement of Considerations, zirconium-based alloy fuel cladding materials may be subject to embrittlement at a lower combination of temperature and level of oxygen absorption (17 percent) than currently allowed under § 50.46(b)(1) due to absorption of hydrogen during normal operation. The proposed rule would correct those limits initially established to prevent embrittlement of zirconium-based alloy cladding material based on the new research information. In addition, the research work has identified new phenomena, such as breakaway oxidation and oxygen diffusion from the cladding inside surfaces, which are believed to further adversely affect the fuel cladding embrittlement process. Thus, post quench ductility (which is necessary to ensure coolable core geometry)[3] is not guaranteed following a postulated LOCA. The proposed rule would establish new requirements for zirconium-based alloys to prevent breakaway oxidation and account for oxygen diffusion from the oxide fuel pellet during the

[3] The Commission concluded, as part of the 1973 Emergency Core Cooling System rulemaking, that retention of ductility in the zircaloy cladding material was determined to be the best guarantee of its remaining intact during the hypothetical loss-of-coolant accident, thereby maintaining a coolable core geometry. *See Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors,* CLI-73-39, *at page 1098* (December 28, 1973).

operating life of the fuel. In sum, the NRC believes that imposing the requirements of the proposed rule is necessary to prevent embrittlement of fuel cladding and to restore the rule to the level of reasonable assurance of adequate protection to public health and safety.

Combined License Applicants

Imposing the requirements of the proposed rule on current and future applicants for combined licenses under Subpart C of Part 52 would not constitute backfitting. Neither the Backfit Rule nor the finality provisions for combined licenses in §§ 52.83 or 52.98 protect either a current or prospective applicant for a combined license from changes in the NRC rules and regulations. The NRC has long adopted the position that the Backfit Rule does not protect current or prospective applicants from changes in NRC requirements or guidance because the policies underlying the Backfit Rule are largely inapplicable in the context of a current or future application. This position also applies to each of the issue finality provisions in Part 52.

Standard Design Certifications

Imposing the requirements of the proposed rule on current and future applicants for standard design certification rules would not constitute backfitting. Neither the Backfit Rule nor the finality provisions for final design certification rules in § 52.63 protect either a current or prospective applicant for a standard design certification rule from changes in the NRC rules and regulations.

Manufacturing Licenses

Imposing the requirements of the proposed rule on future applicants for manufacturing licenses would not constitute backfitting. The NRC has not issued any manufacturing licenses under Part 52, and neither the Backfit Rule nor the finality provisions for manufacturing licenses in § 52.171 protect a prospective manufacturing applicant from changes in the NRC rules and regulations.

Draft Regulatory Guides

The NRC is issuing, for public comment, three draft regulatory guides which would support implementation of § 50.46c. These draft regulatory guides are DG-1261, "Conducting Periodic Testing for Breakaway Oxidation Behavior," (ADAMS Accession No. ML110840089), DG-1262, "Testing for Post Quench Ductility," (ADAMS Accession No. ML110840283), and DG-1263, "Establishing Analytical Limits for Zirconium-Based Alloy Cladding" (ADAMS Accession No. ML110871607). The draft regulatory guides provide guidance on compliance with those proposed new requirements for ECCS not contained in the current ECCS rule, 10 CFR 50.46.

The first issuance of new guidance on a new rule provision[4] does not constitute

[4] The NRC notes that while the proposed 10 CFR 50.46c includes both "amended" requirements and "new" requirements, the three regulatory guides only provide "new" guidance on "new" § 50.46c requirements. By "new" requirements, the NRC means that these requirements have no analogue in the current ECCS rule. For example, the proposed § 50.46c(g)(1)((iii) criterion on breakaway oxidation is a "new" requirement because there is no provision in current § 50.46 requiring consideration of that phenomenon. By contrast, "amended," means that the proposed rule contains several requirements which have analogues to requirements in the existing rule but are being addressed differently. An example of an "amended" requirement would be proposed § 50.46c(d)(1), because that provision: (i) addresses, *in language which differs from the current rule's language*, matters which are addressed in the current rule, including § 50.46(a)(1)(i); and (ii) contains substantively different (proposed) requirements when compared to the current rule, but the proposed requirements are directed at technical matters already addressed in the current ECCS rule. For example, the proposed § 50.46c(g)(1)((iii) criterion on breakaway oxidation is a "new" requirement because there is no provision in current § 50.46 requiring consideration of that phenomenon. By contrast, "amended," means that the proposed rule contains several requirements which have analogues to requirements in the existing rule but are being addressed differently. An example of an "amended" requirement would be proposed § 50.46c(d)(1), because that provision: (i) addresses, *in language which differs from the current rule's language*, matters which are addressed in the current rule, including § 50.46(a)(1)(i); and (ii) contains substantively different (proposed) requirements when compared to the current rule, but the proposed requirements are directed at technical matters already addressed in the current rule.

backfitting, inasmuch as: (i) the guidance on the new rule provision must be consistent with the regulatory requirements in the new rule provision; and (ii) the backfittiing basis for the new rule provision should also be applicable to the issuance of guidance on that new rule provision. Therefore, the first issuance of new guidance addressing new provisions of § 50.46c does not constitute issuance of "changed" or "new" guidance within the meaning of the definition of "backfitting" in 10 CFR 50.109(a)(1), or constitute an action inconsistent with any of the issue finality provisions in 10 CFR Part 52. Accordingly, no further consideration of backfitting is needed to support issuance of the three new regulatory guides in final form.

List of Subjects ~~in 10 CFR Part 50~~

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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. 553, the NRC is proposing to adopt 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: Secs. 102, 103, 104, 105, 161, 182, 183, 186, 189, 68 Stat. 936, 937, 938, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 444, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, as amended, 202, 206, 88 Stat. 1242, as amended, 1244, 1246 (42 U.S.C. 5841, 5842, 5846); sec. 1704, 112 Stat. 2750 (44 U.S.C. 3504 note); Energy Policy Act of 2005, Pub. L. No. 109–58, 119 Stat. 594 (2005). Section 50.7 also issued under Pub. L. 95–601, sec. 10, 92 Stat. 2951 as amended by Pub. L. 102-486, sec 2902, 106 Stat. 3123 (42 U.S.C. 5841). Section 50.10 also issued under secs. 101, 185, 68 Stat. 955, as amended (42 U.S.C. 2131, 2235); sec. 102, Pub. L. 91–190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.13, 50.54(dd), and 50.103 also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138).

Sections 50.23, 50.35, 50.55, and 50.56 also issued under sec. 185, 68 Stat. 955 (42 U.S.C. 2235). Sections 50.33a, 50.55a and Appendix Q also issued under sec. 102, Pub. L. 91–190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.34 and 50.54 also issued under sec. 204, 88 Stat. 1245 (42 U.S.C. 5844). Sections 50.58, 50.91, and 50.92 also issued under Pub. L. 97–415, 96 Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80–50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Appendix F also issued under sec. 187, 68 Stat. 955 (42 U.S.C. 2237).

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, ~~50.46a,~~ 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, and appendices A, B, E, G, H, I, J, K, M, N, O, Q, R, and S to this part.

* * * * *

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 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~~a,~~ and 50.46c of this part, 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 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.

*   *   *   *   *

§ 50.46a [Added and Reserved]

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

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5~~4~~. A new § 50.46c is added to read as follows:

§ 50.46c Emergency core cooling system performance during loss-of-coolant accidents.

(a) *Applicability*. The requirements of this section apply to the design of a light water nuclear power reactor (LWR), and to the following entities who design, construct or operate an LWR: each applicant for or holder of a construction permit under this part, each applicant for or holder of an operating license under this part (until the licensee has submitted the certification required under 10 CFR 50.82(a)(1) to the NRC), each applicant for or holder of a combined license under 10 CFR part 52, each applicant for 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 10 CFR part 52, and each applicant for a manufacturing license under 10 CFR part 52.

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

Breakaway oxidation, for zirconium-alloy cladding material, means the fuel cladding oxidation phenomenon in which weight gain rate deviates from normal kinetics. This change occurs with a rapid increase of hydrogen pickup during prolonged exposure to a high temperature steam environment, which promotes loss of cladding ductility.

Evaluation model means the calculational framework for evaluating the behavior of the reactor system (including fuel) during a postulated loss-of-coolant accident (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. The analytical limits established in accordance with this section, with cooling performance calculated in accordance with an NRC approved 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 of this part.

(d) *Emergency core cooling system design.*

(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 loss-of-coolant accident (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 evaluation model meeting the requirements of either paragraph (d)(2)(i) or (d)(2)(ii),

paragraph (d)(2)(iii), and paragraph (d)(2)(iv), and satisfy the analytical requirements in paragraph (d)(2)(v) of this section. The evaluation model must be reviewed and approved by the NRC.

(i) *Realistic ECCS model*. A realistic model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident. 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 ECCS evaluation model may be developed in conformance with the required and acceptable features of appendix K ECCS Evaluation Models.

(iii) *Core geometry and coolant flow*. The ECCS evaluation model must address calculated changes in core geometry and must consider those factors that may alter localized coolant flow or inhibit delivery of coolant to the core.

(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 (o) of this section, the documentation requirements of this paragraph apply and supersede the requirements in 10 CFR part 50, Appendix K, section II, “Required Documentation.”

(i)(A) A description of each evaluation model must be furnished. 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) A complete listing of each computer program, in the same form as used in the evaluation model, must be furnished to the Nuclear Regulatory Commission upon request.

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

(iii) Appropriate sensitivity studies must be performed for each evaluation model, to evaluate the effect on the calculated results of variations in no~~d~~ding, 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 evaluation model, or portions thereof, must be compared with applicable experimental information.

(v) Elements of evaluation models reviewed will include technical adequacy of the calculational methods, including: for models covered by paragraph (d)(2)(ii) of this section, compliance with required features of section I of Appendix K; and, for models covered by paragraph (d)(2)(i) of this section, assurance of a high level of probability that the performance criteria of paragraph (d)(1) of this section would not be exceeded.

(vi) For operating licenses issued under this part as of **[EFFECTIVE DATE OF RULE],** required documentation of Table 1 must be submitted to demonstrate compliance by the date specified in Table 1.

(e) [Reserved]

(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 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) *Cladding embrittlement*. Analytical limits on peak cladding temperature and integral time at temperature shall be established which 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 used in place of the 2200 °F limit.

(iii) *Breakaway oxidation.* The total accumulated time that the cladding is predicted to remain above a temperature at which the zirconium-alloy has been shown to be susceptible to breakaway oxidation shall not be greater than a limit which corresponds to the measured onset of breakaway oxidation for the zirconium-alloy cladding material based on an NRC-approved experimental technique. The limit must be approved by the NRC.

(iv) *Maximum hydrogen generation*. The calculated total amount of hydrogen generated from any chemical reaction of the fuel 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.

(v) *Long-term cooling*. An analytical limit on long-term peak cladding temperature shall be established which corresponds 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 shall not exceed the established analytical limit. The analytical limit must be approved the by NRC.

(2) *Fuel system modeling requirements.* The 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 evaluation model.

(ii) The thermal effects of crud and oxide layers that accumulate on the fuel cladding during plant operation must be evaluated. For the purposes of this paragraph, crud means any foreign substance deposited on the surface of fuel cladding prior to initiation of a LOCA.

(h) [Reserved]

(i) [Reserved]

(j) [Reserved]

(k) *Use of NRC-approved fuel in reactor*. A licensee may not load fuel into a reactor, or operate the reactor, unless the licensee either determines that the fuel meets the requirements of paragraph (d) of this section, or complies with technical specifications governing lead test assemblies in its license.

(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*.

(1) Each entity subject to the requirements of this section, which identifies any change to or error in an evaluation model or the application of such a model, or any operation inconsistent with the evaluation model or resulting noncompliance with the acceptance criteria in this section, shall comply with the requirements of this paragraph.

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

(ii) If a licensee identifies a change, error or operation which does not result in any predicted response which exceeds any of the acceptance criteria but is significant, then a report describing each such change, error or operation, and a schedule for submitting a reanalysis and implementation of corrective actions must be submitted within 30 days of the change, discovery of the error, or operation.

(iii) If a licensee identifies a change, error or operation which results in any predicted response which exceeds any of the acceptance criteria specified in this section to be exceeded at a facility with an operating license (or, in the case of a combined license under 10 CFR part 52, after the Commission has made the finding under 10 CFR 52.103(g)), then the licensee shall take immediate action to bring the facility into compliance with the acceptance criteria. In addition, the entity shall report the change, error or operation under §§ 50.55(e), 50.72, and 50.73, as applicable, and submit a report describing each such change, error or operation and a schedule for submitting a reanalysis and implementation of corrective actions within 30 days of the change, discovery of the error, or operation.

(iv) If a design certification applicant is required by paragraphs (m)(1)(ii) and (iii) of this section to submit a reanalysis, then that reanalysis must be accompanied by an application to amend the design certification application to reflect the reanalysis.

(2) For the purposes of this section, a significant change or error is one which 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 NRC-approved 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 NRC-approved 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.

(3) Each holder of an operating license or combined license shall measure breakaway oxidation for each reload batch. The holder must report the results to the NRC annually (i.e., anytime within each calendar year), in accordance with § 50.4 or § 52.3 of this chapter, and evaluate the results to determine if there is a failure to conform or a defect that must be reported in accordance with the requirements of 10 CFR part 21.

(n) [Reserved]

(o) *Implementation.*

(1) Construction permits issued under this part after **[EFFECTIVE DATE OF RULE]**

must comply with the requirements of this section at their issuance.

(2) Operating licenses issued under this part which are based upon construction permits in effect as of **[EFFECTIVE DATE OF RULE]** (including deferred and reinstated construction permits) must comply with the requirements of this section by no later than the applicable date set forth in Table 1. Until such compliance is achieved, the requirements of § 50.46 continue to apply.

(3) Operating licenses issued under this part after **[EFFECTIVE DATE OF RULE]** must comply with the requirements of this section.

(4) Operating licenses issued under this part as of **[EFFECTIVE DATE OF RULE]** must comply with the requirements of this section by no later than the applicable date set forth in Table 1. Until such compliance is achieved, the requirements of § 50.46 continue to apply.

(5) Standard design certifications, standard design approvals, and manufacturing licenses under part 52 of this chapter, whose applications (including applications for amendment) are docketed after **[EFFECTIVE DATE OF RULE]**, and new branches of these certifications whose applications are docketed after **[EFFECTIVE DATE OF RULE]** must comply with this section at their issuance.

(6) Standard design certifications under part 52 issued before **[EFFECTIVE DATE OF RULE],** must comply with this section by no later than their renewal.

(7) Standard design certifications, standard design approvals, and manufacturing licenses under part 52 of this chapter issued after **[EFFECTIVE DATE OF RULE]** whose applications were pending as of **[EFFECTIVE DATE OF RULE]** and new branches of

certifications issued after **[EFFECTIVE DATE OF RULE]** whose applications were pending as of **[EFFECTIVE DATE OF RULE]** must comply with this section by no later than their renewal.

(8) Combined licenses under part 52 of this chapter docketed after **[EFFECTIVE DATE OF RULE]** must comply with this section at their issuance.

(9) Combined licenses under part 52 of this chapter docketed or issued before **[EFFECTIVE DATE OF RULE]** must comply with this section no later than completion of the first refueling outage after initial fuel load. Until such compliance is achieved, the requirements in § 50.46 continue to apply.

Table 1: Implementation dates for Nuclear Power Plants with operating licenses and construction permits as of **[EFFECTIVE DATE OF RULE].**

Track	Reactor Type	Plant Name	Compliance Demonstration
1	PWR	Arkansas Nuclear One - Unit 1 Braidwood Station – Unit 1 Byron Station – Unit 1 Calvert Cliffs Nuclear Power Plant – Unit 1 Calvert Cliffs Nuclear Power Plant – Unit 2 Comanche Peak Nuclear Power Plant – Unit 1	No later than 24 months from effective date of rule

		Comanche Peak Nuclear Power Plant – Unit 2	
		Crystal River Nuclear Generating Plant – Unit 3	
		Davis-Besse Nuclear Power Station – Unit 1	
		Diablo Canyon Power Plant – Unit 2	
		Fort Calhoun Station – Unit 1	
		H.B. Robinson Steam Electric Plant - Unit 2	
		Indian Point Nuclear Generating – Unit 2	
		J.M. Farley Nuclear Plant – Unit 1	
		J.M. Farley Nuclear Plant – Unit 2	
		Millstone Power Station – Unit 2	
		Millstone Power Station – Unit 3	
		North Anna Power Station – Unit 1	
		North Anna Power Station – Unit 2	
		Oconee Nuclear Station – Unit 1	
		Oconee Nuclear Station – Unit 2	
		Oconee Nuclear Station – Unit 3	

		Palisades Nuclear Plant Point Beach Nuclear Plant – Unit 1 Point Beach Nuclear Plant – Unit 2 Prairie Island Nuclear Generating Plant – Unit 1 Prairie Island Nuclear Generating Plant – Unit 2 R.E. Ginna Nuclear Power Plant Saint Lucie Plant – Unit 1 Seabrook Station – Unit 1 Sequoyah Nuclear Plant – Unit 1 Sequoyah Nuclear Plant – Unit 2 Three Mile Island – Unit 1 Turkey Point Nuclear Generating – Unit 3 Turkey Point Nuclear Generating – Unit 4 Vogtle Electric Generating Plant – Unit 1 Vogtle Electric Generating Plant – Unit 2 Wolf Creek Generating Station – Unit 1	

	BWR	Browns Ferry Nuclear Plant – Unit 1 Browns Ferry Nuclear Plant – Unit 2 Browns Ferry Nuclear Plant – Unit 3 Brunswick Steam Electric Plant – Unit 1 Brunswick Steam Electric Plant – Unit 2 Clinton Power Station – Unit 1 Columbia Generating Station Cooper Nuclear Station Duane Arnold Energy Center E.I. Hatch Nuclear Plant – Unit 1 E.I. Hatch Nuclear Plant – Unit 2 Fermi – Unit 2 Hope Creek Generating Station – Unit 1 Grand Gulf Nuclear Station – Unit 1 J.A. Fitzpatrick Nuclear Power Plant LaSalle County Station – Unit 1	

		LaSalle County Station – Unit 2 Limerick Generating Station – Unit 1 Limerick Generating Station – Unit 2 Nine Mile Point Nuclear Station – Unit 2 Peach Bottom Atomic Power Station – Unit 2 Peach Bottom Atomic Power Station – Unit 3 Perry Nuclear Power Plant – Unit 1 River Bend Station – Unit 1 Susquehanna Steam Electric Station – Unit 1 Susquehanna Steam Electric Station – Unit 2 Vermont Yankee Nuclear Power Station	
2	PWR	Beaver Valley Power Station – Unit 1 Beaver Valley Power Station – Unit 2 Braidwood Station – Unit 2 Byron Station – Unit 2 Catawba Nuclear Station – Unit 1	No later than 48 months from effective date of rule

		Catawba Nuclear Station – Unit 2 D.C. Cook Nuclear Plant – Unit 1 D.C. Cook Nuclear Plant – Unit 2 Diablo Canyon Power Plant – Unit 1 Indian Point Nuclear Generating – Unit 3 Kewaunee Power Station McGuire Nuclear Station – Unit 1 McGuire Nuclear Station – Unit 2 Watts Bar Nuclear Plant – Unit 1	
	BWR	Nine Mile Point Nuclear Station – Unit 1 Oyster Creek Nuclear Generating Station	
3	PWR	Arkansas Nuclear One - Unit 2 Callaway Plant – Unit 1 Palo Verde Nuclear Generating Station – Unit 1 Palo Verde Nuclear Generating Station – Unit 2 Palo Verde Nuclear Generating Station – Unit 2	No later than 60 months from effective date of rule

		Saint Lucie Plant – Unit 2 Salem Nuclear Generating Station – Unit 1 Salem Nuclear Generating Station – Unit 2 San Onfre Nuclear Generating Station – Unit 2 San Onfre Nuclear Generating Station – Unit 3 Shearon Harris Nuclear Power Plant – Unit 1 South Texas Project – Unit 1 South Texas Project – Unit 2 Surry Power Plant – Unit 1 Surry Power Plant – Unit 2 V.C. Summer Nuclear Station – Unit 1 Waterford Steam Electric Station – Unit 3	
	BWR	Dresden Nuclear Power Station – Unit 2 Dresdan Nuclear Power Station – Unit 3 Monticello Nuclear Generating Plant – Unit 1 Pilgrim Nuclear Power Station	

		Quad Cities Nuclear Power Station, Unit 1 Quad Cities Nuclear Power Station, Unit 2	

*   *   *   *   *

6~~5~~. In Appendix K to p~~P~~art 50, ~~paragraph (II)(5) is revised to read as follows~~a new paragraph II.6 is added to read as follows:

Appendix K to Part 50 – ECCS Evaluation Models

*   *   *   *   *

II.   *   *   *

~~5~~6. ~~General Standards for Acceptability--Elements of evaluation models reviewed will include technical adequacy of the calculational methods, including: For models covered by §50.46c(d)(4)(i) compliance with required features of section I of this Appendix K; and, for models covered by § 50.46c(d)(4), assurance of a high level of probability that the performance criteria of § 50.46c(b) would not be exceeded.~~Upon implementation of 10 CFR 50.46c in accordance with § 50.46c(o), the documentation requirements in § 50.46c(d)(3) apply and supersede the requirements of section II of this appendix.

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

7~~6~~. The authority citation for part 52 continues to read as follows:

AUTHORITY: Secs. 103, 104, 161, 182, 183, 185, 186, 189, 68 Stat. 936, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 444, as amended (42 U.S.C. 2133, 2201, 2232, 2233, 2235, 2236, 2239, 2282); secs. 201, 202, 206, 88 Stat. 1242, 1244, 1246, as amended (42 U.S.C. 5841, 5842, 5846); sec. 1704, 112 Stat. 2750 (44 U.S.C. 3504 note); Energy Policy Act of 2005, Pub. L. No. 109-58, 119 Stat. 594 (2005), secs. 147 and 149 of the Atomic Energy Act.

8~~7~~. 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 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~~a~~ and 50.46c of this chapter, as applicable;~~.~~

* * * * *

9~~8~~. 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 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~~a~~ and 50.46c of this chapter, as applicable;~~.~~

* * * * *

10~~9~~. 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 SSC 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 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~~a~~, and 50.46c of this chapter, as applicable;~~.~~

* * * * *

11~~0~~. 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 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~~a~~, and 50.46c of this chapter, as applicable;~~.~~

*   *   *   *   *

Dated at Rockville, Maryland, this______________ day of_______, 2012.

For the Nuclear Regulatory Commission.

Annette L. Vietti-Cook,
Secretary of the Commission.

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~~a~~, and 50.46c of this chapter, as applicable;~~.~~

* * * * *

Dated at Rockville, Maryland, this_____________ day of______, 2012.

For the Nuclear Regulatory Commission.

Annette L. Vietti-Cook,
Secretary of the Commission.

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OFFICE	NRR/DPR/PRMB: PM	RES/DE/RGDB*	NRR/DSS*	NRR/DPR/PRMB: BC	NRR/DPR:D
NAME	TInverso	AHicks	PClifford	SHelton	TMcGinty (RNelson for)
DATE	09/28/2011	12/19/2011	09/28/2011	10/07/2011	11/08/2011
OFFICE	NRR/DSS:D	OIS/IRSD: TL*	OE:D*	NRO*	RES*
NAME	WRuland	TDonnell	RZimmerman (JWray for)	MJohnson (CAder for)	BSheron (BHolian for)
DATE	11/22/2011	12/07/2011	11/17/2011	11/21/2011	11/18/2011
OFFICE	ADM/DAS/RADB*	CFO*	OGC*	NRR	EDO
NAME	CBladey (LTerry for)	JDyer (GPeterson for)	BJones (GMizuno for)	ELeeds (BBoger for)	RBorchardt
DATE	11/18/2011	11/15/2011	01/11/2012	01/26/2012	3/1/2012

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