DRAFT FINAL § 50.46a RULE LANGUAGE

Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements

(ADAMS Accession no. ML102440294)

The NRC staff provided this draft final rule language to the Advisory Committee for Reactor Safeguards in preparation for the subcommittee meeting on September 22, 2010.

NOTE: The availability of this draft rule language is intended to inform stakeholders of the current status of the NRC's activities to provide a risk-informed alternative to the current ECCS requirements. This draft final rule language may be incomplete or in error in one or more respects and may be subject to further revisions during the rulemaking process. The NRC is not soliciting formal public comments on this draft rule language.

Any questions on this rule language may be addressed to the NRC rulemaking project manager, Richard Dudley (301-415-1116; richard.dudley@nrc.gov).

List of Subjects

10 CFR Part 50

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

10 CFR Part 52

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

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended; the Energy Reorganization Act of 1974; and 5 U.S.C. 552 and 553; the NRC is adopting the following amendments to 10 CFR Parts 50 and 52.

PART 50 -- DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

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

Authority: 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. 194 (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 § 50.34, paragraphs (a)(4) and (b)(4) are revised to read as follows:

§ 50.34 Contents of application; 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.

(i) Analysis and evaluation of ECCS cooling performance and the need for high point vents following postulated loss-of-coolant accidents must be performed under the requirements of either § 50.46 or § 50.46a, and § 50.46b for facilities whose operating licenses were issued after December 28, 1974, but before [EFFECTIVE DATE OF RULE], and for facilities for which construction permits may be issued after [EFFECTIVE DATE OF RULE] and are demonstrated under § 50.46a(c)(2) to have designs that are similar to the designs of reactors licensed before [EFFECTIVE DATE OF RULE].

(ii) Analysis and evaluation of ECCS cooling performance and the need for high point vents following postulated loss-of-coolant accidents must be performed under the requirements of § 50.46 and § 50.46b for facilities for which construction permits may be issued after [EFFECTIVE DATE OF RULE] and are not demonstrated under § 50.46a(c)(2) to have designs that are similar to the designs of reactors licensed before [EFFECTIVE DATE OF RULE].

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(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.

(i) Analysis and evaluation of ECCS cooling performance following postulated LOCAs must be performed under the requirements of either § 50.46 or § 50.46a, and § 50.46b for facilities whose operating licenses were issued after December 28, 1974, but before [EFFECTIVE DATE OF RULE], and for facilities whose operating licenses are issued after [EFFECTIVE DATE OF RULE] and are demonstrated under § 50.46a(c)(2) to have designs that are similar to the designs of reactors licensed before [EFFECTIVE DATE OF RULE].

(ii) Analysis and evaluation of ECCS cooling performance following postulated LOCAs must be performed under the requirements of §§ 50.46 and 50.46b for facilities whose operating licenses are issued after [EFFECTIVE DATE OF RULE] and are not demonstrated under § 50.46a(c)(2) to have designs that are similar to the designs of reactors licensed before [EFFECTIVE DATE OF RULE].

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3. In § 50.46, paragraph (a) is amended by adding an introductory paragraph and revising paragraph (a)(1)(i) to read as follows:

§ 50.46 Acceptance criteria for emergency core cooling systems for light-water nuclear power plants.

(a) Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircalloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS). The ECCS system must be designed under the requirements of this section or § 50.46a for facilities whose operating licenses were issued before [EFFECTIVE DATE OF RULE]; for facilities whose operating licenses, combined licenses under part 52 of this chapter, or manufacturing licenses under part 52 of this chapter are issued after [EFFECTIVE DATE OF RULE] and are demonstrated under § 50.46a(c)(2) to have designs that are similar to the designs of reactors licensed before [EFFECTIVE DATE OF RULE]; and for design approvals and design certifications under part 52 of this chapter issued after [EFFECTIVE DATE OF RULE] that are demonstrated under § 50.46a(c)(2) to have designs that are similar to the designs of reactors licensed before [EFFECTIVE DATE OF RULE]. The ECCS system must be designed under the requirements of this section for facilities whose operating licenses, combined licenses under part 52 of this chapter, or manufacturing licenses under part 52 of this chapter are issued after [EFFECTIVE DATE OF RULE] and are not demonstrated under § 50.46a(c)(2) to have designs that are similar to the designs of reactors licensed before [EFFECTIVE DATE OF RULE]; and for design approvals and design certifications under part 52 of this chapter that are not demonstrated under § 50.46a(c)(2) to have designs that are similar to the designs of reactors licensed before [EFFECTIVE DATE OF RULE].

(1)(i) The ECCS system must be designed so that its calculated cooling performance following postulated LOCAs conforms to the criteria set forth in paragraph (b) of this section. ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated LOCAs of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated LOCAs are calculated. Except as provided in paragraph (a)(1)(ii) of this section, the evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a LOCA. 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 criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded. Appendix K, Part II Required Documentation, sets forth the documentation requirements for each evaluation model. This section does not apply to a nuclear power reactor facility for which the certifications required under § 50.82(a)(1) have been submitted.

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4. Section 50.46a is redesignated as § 50.46b, and a new § 50.46a is added to read as follows:

§ 50.46a Alternative acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.

(a) *Definitions*. For the purposes of this section:

(1) Changes enabled by this section means changes to the facility, technical specifications, and procedures that satisfy the alternative ECCS analysis requirements under this section but do not satisfy the ECCS requirements under 10 CFR 50.46.

(2) Evaluation model means the calculational framework for evaluating the behavior of the reactor system during a postulated design-basis 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.

(3) Loss-of-coolant accidents (LOCAs) means the hypothetical accidents 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. LOCAs involving breaks at or below the transition break size (TBS) are design-

basis accidents. LOCAs involving breaks larger than the TBS are beyond design-basis accidents.

(4) Operating configuration means those plant characteristics, such as power level, equipment unavailability (including unavailability caused by corrective and preventive maintenance), and equipment capability that affect plant response to a LOCA.

(5) Transition break size (TBS) for reactors licensed before [EFFECTIVE DATE OF RULE] is a break area equal to the cross-sectional flow area of the inside diameter of the largest piping attached to the reactor coolant system for a pressurized water reactor, or the inside diameter of the larger of the feedwater line inside containment or the residual heat removal line inside containment for a boiling water reactor. For reactors licensed after [EFFECTIVE DATE OF RULE], and for design certifications, design approvals, and manufacturing licenses approved or issued after [EFFECTIVE DATE OF RULE], the TBS will be determined on a plant-specific basis.

(b) *Applicability and scope.*

(1) The requirements of this section may be applied to each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircalloy or ZIRLO cladding whose operating license was issued prior to [EFFECTIVE DATE OF RULE]; to each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircalloy or ZIRLO cladding whose operating license, combined license under part 52 of this chapter or manufacturing license under part 52 of this chapter is issued after [EFFECTIVE DATE OF RULE] and whose design is demonstrated under § 50.46a(c)(2) to be similar to the designs of reactors licensed before [EFFECTIVE DATE OF RULE]; and to each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircalloy or ZIRLO cladding whose design approval or design certification under part 52 of this chapter is demonstrated under § 50.46a(c)(2) to be similar to the designs of reactors

licensed before [EFFECTIVE DATE OF RULE]. The requirements of this section do not apply to a reactor for which the certification required under § 50.82(a)(1) has been submitted.

(2) The requirements of this section are in addition to any other requirements applicable to ECCS set forth in this part, with the exception of § 50.46. The criteria set forth in paragraphs (e)(3) and (e)(4) of this section, with cooling performance calculated in accordance with an acceptable evaluation model or analysis method under paragraphs (e)(1) and (e)(2) of this section, are in implementation of the general requirements with respect to ECCS cooling performance design set forth in this part, including in particular Criterion 35 of Appendix A to this part.

(c) *Application*. (1) A construction permit holder or licensee of a facility, or other entity seeking to implement this section shall submit an application for a license amendment under § 50.90 that contains the following information:

(i) A written evaluation demonstrating applicability of the results in NUREG-1829, “Estimating Loss-of-Coolant Accident (LOCA) Frequencies through the Elicitation Process”; March 2008 and NUREG-1903, “Seismic Considerations for the Transition Break Size”; February 2008, to the licensee’s facility. As part of this evaluation, the application must contain a plant specific analysis demonstrating that the risk of seismically-induced LOCAs larger than the TBS is comparable to or less than the seismically-induced LOCA risk associated with the NUREG-1903 results.

(ii) Identification of the approved analysis method(s) for demonstrating compliance with the ECCS criteria in paragraph (e) of this section.

(iii) A description of the risk-informed evaluation used to:

(A) Demonstrate that the proposed changes to the facility meet the requirements in paragraph (f) of this section; and

(B) Develop any proposed alternative to the 14-day per year period specified in paragraph (d)(5), if applicable.

(iv) A construction permit holder or licensee of a facility, or other entity who wishes to make changes enabled by this section without prior NRC review and approval must submit for NRC approval a process to be used for evaluating the acceptability of these changes; including:

(A) A description of the approach, methods, and decisionmaking process to be used for evaluating compliance with the acceptance criteria in paragraphs (f)(1), (f)(2), and (f)(3) of this section; and

(B) A description of the PRA model and non-PRA risk assessment methods to be used for demonstrating compliance with paragraphs (f)(4) and (f)(5) of this section.

(v) A description of non safety equipment that is credited for demonstrating compliance with the ECCS acceptance criteria in paragraph (e) of this section.

(vi) A written evaluation demonstrating how the leak detection program in place at the facility satisfies the criteria in paragraph (d)(2) of this section.

(2) An applicant for a construction permit, operating license, design approval, design certification, manufacturing license, or combined license or holder of a design approval seeking to implement the requirements of this section shall, in addition to the information required by paragraphs (c)(1)(i)-(iv) of this section, submit an analysis demonstrating why the proposed reactor design is similar to the designs of reactors licensed before [EFFECTIVE DATE OF RULE] such that the provisions of this section may properly apply. The analysis must also include a recommendation for an appropriate TBS and a justification that the recommended TBS is consistent with the technical basis for this section.

(3) Acceptance criteria. The NRC may approve an application to use this section if:

(i) The evaluation submitted under paragraph (c)(1)(i) of this section demonstrates that the NUREG-1829 results are applicable to the facility, and the risk of seismically-induced LOCAs larger than the TBS is comparable to or less than the seismically-induced LOCA risk associated with the NUREG-1903 results;

(ii) The method(s) for demonstrating compliance with the ECCS acceptance criteria in paragraphs (e)(3) and (e)(4) of this section meet the requirements in paragraphs (e)(1) and (e)(2) of this section;

(iii) The risk-informed evaluation process(es) the licensee used to:

(A) Make changes under this section is adequate for determining whether the acceptance criteria in paragraph (f) of this section have been met; and

(B) Demonstrate that any short time other than 14 days per year that may be proposed for use under paragraph (d)(5) of this section, is consistent with the mitigative capability available, the configuration specific risk, the philosophy of defense-in-depth, and adequate safety margins.

(iv) If applicable, the risk-informed process the licensee proposes to use for making changes under paragraph (f)(1) of this section, is adequate for determining whether the acceptance criteria in paragraph (f) of this section, have been met.

(v) Non safety equipment that is credited for demonstrating compliance with the ECCS acceptance criteria in paragraph (e) of this section is identified in plant Technical Specifications.

(vi) For all applicants other than those holding operating licenses issued before [EFFECTIVE DATE OF RULE], the proposed reactor design is similar to the designs of reactors licensed before [EFFECTIVE DATE OF RULE] and the applicant's proposed TBS is consistent with the technical basis of this section.

(d) *Requirements during operation.* A licensee whose application under paragraph (c) of this section is approved by the NRC shall comply with the following requirements as long as the facility is subject to the requirements in this section until the licensee submits the certifications required by § 50.82(a):

(1) The licensee shall maintain ECCS model(s) and/or analysis method(s) meeting the requirements in paragraphs (e)(1) and (e)(2) of this section;

(2) The licensee shall have leak detection systems available at the facility and shall implement actions as necessary to identify, monitor and quantify leakage to ensure that adverse safety consequences do not result from primary pressure boundary leakage from piping and components that are larger than the transition break size.

(3) Changes made under this section must, in addition to meeting other applicable NRC requirements, be evaluated by a risk-informed evaluation demonstrating that the acceptance criteria in paragraph (f) of this section, are met.

(4) The licensee shall periodically maintain and upgrade, as necessary, its risk assessments to meet the requirements in paragraph (f)(4) and (f)(5) of this section. The maintenance and upgrading shall be consistent with NRC-endorsed consensus standards on PRA and must be completed in a timely manner, but no less often than once every four years. Based upon a re-evaluation of the risk assessments after the periodic maintenance and upgrading are completed, the licensee shall take appropriate action to ensure that the acceptance criteria in paragraph (f) of this section, as applicable, are met. The PRA maintenance and upgrading required by this section, and any necessary changes to the facility, technical specifications and procedures as a result of this re-evaluation, shall not be deemed to be backfitting under any provision of this chapter.

(5) For LOCAs larger than the TBS, operation in a plant operating configuration not demonstrated to meet the acceptance criteria in paragraph (e)(4) of this section may not exceed a short time. A short time is either a total of fourteen (14) days in any 12 month period or an alternative proposed by the licensee and approved by the NRC.

(6) The licensee shall perform an evaluation to determine the effect of all planned facility changes and shall not implement any facility change that would invalidate the evaluation performed pursuant to § 50.46a(c)(1)(i) demonstrating the applicability to the licensee’s facility of the generic results in NUREG-1829 and NUREG-1903.

(e) *ECCS Performance*. Each nuclear power reactor or nuclear power reactor design subject to this section must be provided with an ECCS that must be designed so that its calculated cooling performance following postulated LOCAs conforms to the criteria set forth in this section. The evaluation models for LOCAs must meet the criteria in this paragraph, and must be approved for use by the NRC. Appendix K, Part II, to 10 CFR Part 50, sets forth the documentation requirements for evaluation models.

(1) *ECCS evaluation for LOCAs involving breaks at or below the TBS*. ECCS cooling performance at or below the TBS must be calculated in accordance with an evaluation model that meets the requirements of either section I to Appendix K of this part, or the following requirements, and must demonstrate that the acceptance criteria in paragraph (e)(3) of this section are satisfied. The evaluation model must be used for a number of postulated LOCAs of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated LOCAs involving breaks at or below the TBS are analyzed. The evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a LOCA. 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 criteria set forth in paragraph (e)(3) of this section, there is a high level of probability that the criteria would not be exceeded.

(2) *ECCS analyses for LOCAs involving breaks larger than the TBS*. ECCS cooling performance for LOCAs involving breaks larger than the TBS must be calculated in accordance with an evaluation model that meets the requirements of either section I to Appendix K of this part, or the following requirements, and must demonstrate that the acceptance criteria in paragraph (e)(4) of this section are satisfied. The evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior

of the reactor system during a LOCA. 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 criteria set forth in paragraph (e)(4) of this section, there is a high level of probability that the criteria would not be exceeded. The evaluation model must be used for a number of postulated LOCAs of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated LOCAs larger than the TBS up to the double-ended rupture of the largest pipe in the reactor coolant system are analyzed. These calculations may take credit for the availability of offsite power and do not require the assumption of a single failure. Realistic initial conditions and availability of safety-related or non safety-related equipment may be assumed if supported by plant-specific data or analysis, and provided that onsite power can be readily provided through simple manual actions to equipment that is credited in the analysis.

(3) *Acceptance criteria for LOCAs involving breaks at or below the TBS*. The following acceptance criteria must be used in determining the acceptability of ECCS cooling performance:

(i) *Peak cladding temperature.* The calculated maximum fuel element cladding temperature must not exceed 2200°F.

(ii) *Maximum cladding oxidation*. The calculated total oxidation of the cladding must not at any location exceed 0.17 times the total cladding thickness before oxidation. As used in this paragraph, total oxidation means the total thickness of cladding metal that would be locally converted to oxide if all the oxygen absorbed by and reacted with the cladding locally were converted to stoichiometric zirconium dioxide. If cladding rupture is calculated to occur, the inside surfaces of the cladding must be included in the oxidation, beginning at the calculated time of rupture. Cladding thickness before oxidation means the radial distance from inside to outside the cladding, after any calculated rupture or swelling has occurred but before significant oxidation. Where the calculated conditions of transient pressure and temperature lead to a prediction of cladding swelling, with or without cladding rupture, the unoxidized cladding thickness must be defined as the cladding cross-sectional area, taken at a horizontal plane at the elevation of the rupture, if it occurs, or at the elevation of the highest cladding temperature if no rupture is calculated to occur, divided by the average circumference at that elevation. For ruptured cladding the circumference does not include the rupture opening.

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

(iv) *Coolable geometry*. Calculated changes in core geometry must be such that the core remains amenable to cooling.

(v) *Long term cooling*. After any calculated successful initial operation of the ECCS, the calculated core temperature must be maintained at an acceptably low value and decay heat must be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

(4) *Acceptance criteria for LOCAs involving breaks larger than the TBS*. The following acceptance criteria must be used in determining the acceptability of ECCS cooling performance:

(i) *Coolable geometry*. Calculated changes in core geometry must be such that the core remains amenable to cooling.

(ii) *Long term cooling*. After any calculated successful initial operation of the ECCS, the calculated core temperature must be maintained at an acceptably low value and decay heat must be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

(5) *Imposition of restrictions*. The Director of the Office of Nuclear Reactor Regulation or 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 paragraph (e) of this section.

(f) *Changes to facility, technical specifications, or procedures*. The holder of construction permit, licensee, or other entity who wishes to make changes enabled by this section, to the facility, facility design, or procedures or to the technical specifications shall perform a risk-informed evaluation.

(1) The licensee may make changes enabled by this section, other than changes to the technical specifications, without prior NRC approval if:

(i) The change is permitted under § 50.59 for holders of operating licenses, combined licenses that do not reference a design certification, design approval, or manufacturing license (per § 52.98(b)), or combined licenses that reference a design approval; permitted under § 52.98(c) for holders of combined licenses that reference a design certification; or permitted under § 52.98(d) for holders of combined licenses that reference a manufacturing license,

(ii) The risk informed evaluation process described in paragraph (c)(1)(iv) of this section demonstrates that any increases in the estimated risk are minimal and the criteria in paragraph (f)(3) of this section are met, and

(iii) The change does not invalidate the evaluation performed pursuant to paragraph (c)(1)(i) of the applicability of the results in NUREG-1829 and NUREG-1903 to the licensee's facility.

(2) For implementing changes which are not permitted under paragraph (f)(1) of this section, the licensee must submit an application for license amendment under § 50.90. The application must contain:

(i) The information required under § 50.90;

(ii) Information from the risk-informed evaluation demonstrating that the total increases in core damage frequency and large early release frequency are very small and the overall risk remains small and the criteria in paragraph (f)(3) of this section are met;

(iii) If previous changes have been made under § 50.46a, information from the risk-informed evaluation on the cumulative effect on risk of the proposed change and all previous changes made under this section. If more than one plant change is combined; including plant changes not enabled by this section, into a group for the purposes of evaluating acceptable risk increases; the evaluation of each individual change shall be performed along with the evaluation of combined changes; and

(iv) Information demonstrating that the criteria in paragraphs (e)(3) and (e)(4) of this section are met.

(v) Information demonstrating that the proposed change will not increase the LOCA frequency of the facility (including the frequency of seismically-induced LOCAs) by an amount that would invalidate the applicability to the facility of the generic studies (NUREG-1829,"Estimating Loss-of-Coolant Accident (LOCA) Frequencies through the Elicitation Process", March 2008 and NUREG-1903, "Seismic Considerations for the Transition Break Size", February 2008") that support the technical basis for this section.

(3) All changes made under this rule must meet the following criteria:

(i) Adequate defense in depth is maintained;

(ii) Adequate safety margins are retained to account for uncertainties;

(iii) Adequate performance-measurement programs are implemented to ensure the risk-informed evaluation continues to reflect actual plant design and operation. These programs shall be designed to detect degradation of the system, structure or component before plant safety is compromised, provide feedback of information and timely corrective actions, and monitor systems, structures or components at a level commensurate with their safety significance, and

(iv) For applicants or licensees referencing a certified design, will not result in a significant decrease in the level of safety otherwise provided by the certified design.

(4) *Requirements for risk assessment - PRA.* Whenever a PRA is used in the risk-informed evaluation, the PRA must, with respect to the area of evaluation which is the subject of the PRA:

(i) Address initiating events from sources both internal and external to the plant and for all modes of operation, including low power and shutdown modes, that would affect the regulatory decision in a substantial manner;

(ii) Reasonably represent the current configuration and operating practices at the plant;

(iii) Have sufficient technical adequacy (including consideration of uncertainty) and level of detail to provide confidence that the total risk estimate and the change in total risk estimate adequately reflect the plant and the effect of the proposed change on risk; and

(iv) Be determined, through peer review, to meet industry standards for PRA quality that have been endorsed by the NRC.

(5) *Requirements for risk assessment other than PRA*. Whenever risk assessment methods other than PRAs are used to develop quantitative or qualitative estimates of changes to risk in the risk-informed evaluation, an integrated, systematic process must be used. All aspects of the analyses must reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operating experience.

(g) *Reporting*.

(1) Licensees. (i) Each licensee shall estimate the effect of any change to or error in evaluation models or analysis methods or in the application of such models or methods to determine if the change or error is significant. For each change to or error discovered in an ECCS evaluation model or analysis method or in the application of such a model that affects the calculated results, the licensee shall report the nature of the change or error and its estimated effect on the limiting ECCS analysis to the Commission at least annually as specified in §§ 50.4 or 52.3. If the change or error is significant, the licensee shall provide this report within 30 days and include with the report a proposed schedule for providing a reanalysis or taking other action

as may be needed to show compliance with § 50.46a requirements. This schedule may be developed using an integrated scheduling system previously approved for the facility by the NRC. For those facilities not using an NRC-approved integrated scheduling system, a schedule will be established by the NRC staff within 60 days of receipt of the proposed schedule. Any change or error correction that results in a calculated ECCS performance that does not conform to the criteria set forth in paragraphs (e)(3) or (e)(4) of this section is a reportable event as described in §§ 50.55(e), 50.72 and 50.73. The licensee shall propose immediate steps to demonstrate compliance or bring plant design or operation into compliance with § 50.46a requirements. For the purpose of this paragraph, a significant change or error is:

(A) For LOCAs involving pipe breaks at or below the TBS, one which results either in a calculated peak fuel cladding temperature different by more than 50°F from the temperature calculated for the limiting transient using the last acceptable 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

(B) For LOCAs involving pipe breaks larger than the TBS, one which results in a significant reduction in the capability to meet the requirements of paragraph (e)(4) of this section.

(ii) As part of the PRA maintenance and upgrading under paragraph (d)(4) of this section, the licensee shall report to the NRC if the re-evaluation results in exceeding the acceptance criteria in paragraph (f) of this section, as applicable. The report must be filed with the NRC no more than 60 days after completing the PRA re-evaluation. The report must describe and explain the changes in the PRA modeling, plant design, or plant operation that led to the increase(s) in risk, and must include a description of and implementation schedule for any corrective actions required under paragraph (d)(4) of this section.

(iii) Every 24 months, the licensee shall submit, as specified in §§ 50.4 or 52.3, a short description of each change involving minimal changes in risk made under paragraph (f)(1) of this section after the last report and a brief summary of the basis for the licensee's determination pursuant to § 50.46a(f)(2)(vi) that the change does not invalidate the applicability evaluation made under § 50.46a(c)(1)(i).

(2) *Design certifications; applicants for and holders of design approvals*. Each design certification applicant and each applicant for and holder of a design approval shall estimate the effect of any change to or error in evaluation models or analysis methods or in the application of such models or methods to determine if the change or error is significant. For each change to or error discovered in an ECCS evaluation model or analysis method or in the application of such a model that affects the calculated results, the applicant or holder shall report the nature of the change or error and its estimated effect on the limiting ECCS analysis to the Commission and to any applicant or licensee referencing the design approval or design certification at least annually as specified in § 52.3. If the change or error is significant, the applicant or holder shall provide this report within 30 days and include with the report a proposed schedule for providing a reanalysis or taking other action as may be needed to show compliance with § 50.46a requirements. A design certification applicant's duty to report under this paragraph continues until the later of either the termination or expiration of the design certification; or the termination of the last license directly or indirectly referencing the design certification. For the purpose of this paragraph, a significant change or error is:

(i) For LOCAs involving pipe breaks at or below the TBS, one which results either in a calculated peak fuel cladding temperature different by more than 50 °F from the temperature calculated for the limiting transient using the last acceptable 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) For LOCAs involving pipe breaks larger than the TBS, one which results in a significant reduction in the capability to meet the requirements of paragraph (e)(4) of this section.

(h) *Documentation*. Following implementation of the § 50.46a requirements, each entity subject to this section shall maintain records sufficient to demonstrate compliance with the requirements in this section in accordance with § 50.71.

(i) through (l) - [RESERVED]

(m) *Changes to TBS*. If the NRC increases the TBS specified in this section, affected entities shall take the following actions.

(1) O*perating licenses under Part 50, combined licenses under Part 52, and manufacturing licenses*. Each licensee subject to this section (other than a licensee referencing a design certification rule or a design approval complying with the requirements of this section) shall re-perform the evaluations required by paragraphs (e)(1) and (e)(2) of this section and reconfirm compliance with the acceptance criteria in paragraphs (e)(3) and (e)(4) of this section. If the licensee cannot demonstrate compliance with the acceptance criteria, then the licensee shall change its facility, technical specifications or procedures so that the acceptance criteria are met. The evaluation required by this paragraph, and any necessary changes to the facility, technical specifications or procedures as the result of this evaluation, are not to be deemed to be backfitting under any provision of this chapter or a violation of any finality provision in Part 52.

(2) Design certifications and referencing combined licenses under Part 52. Changes to a TBS for a design certification must be accomplished by rulemaking, in accordance with 10 CFR 52.63(a). Holders of combined licenses referencing a design certification rule shall re-perform the evaluations required by paragraphs (e)(1) and (e)(2) of this section and reconfirm compliance with the acceptance criteria in paragraphs (e)(3) and (e)(4) of this section. If the licensee cannot demonstrate compliance with the acceptance criteria, then the licensee shall change its facility, technical specifications or procedures so that the acceptance criteria are met. These actions are deemed to be in conformance with applicable finality provisions in Part 52.

(3) Design approvals and referencing combined licenses under Part 52. Holders of combined licenses referencing a design approval rule shall re-perform the evaluations required by paragraphs (e)(1) and (e)(2) of this section and reconfirm compliance with the acceptance criteria in paragraphs (e)(3) and (e)(4) of this section. If the licensee cannot demonstrate compliance with the acceptance criteria, then the licensee shall change its facility, technical specifications or procedures so that the acceptance criteria are met. The evaluation required by this paragraph, and any necessary changes to the facility design, technical specifications or procedures as the result of this evaluation, are not to be deemed to be backfitting under any provision of this chapter or a violation of any finality provision in Part 52.

5. In § 50.109, paragraph (b) is revised to read as follows:

§ 50.109 Backfitting.

* * * * *

(b) Paragraph (a)(3) of this section shall not apply to:

(1) Backfits imposed prior to October 21, 1985; and

(2) Any changes made to the TBS specified in § 50.46a or as otherwise applied to a licensee.

* * * * *

6. In Appendix A to 10 CFR Part 50, under the heading, “CRITERIA,” Criterion 17, 35, 38, 41, 44, and 50 are revised to read as follows:

APPENDIX A TO PART 50 -GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

* * * * *

CRITERIA

*   *   *   *   *

Criterion 17--Electrical power systems. An on-site electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power supplies, including the batteries, and the onsite electrical distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure, except for loss of coolant accidents involving pipe breaks larger than the transition break size under 10 CFR 50.46a, where a single failure of the onsite power supplies and electrical distribution system need not be assumed for plants under 10 CFR 50.46a. For those pipe breaks only, neither a single failure nor the unavailability of offsite power need be assumed.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a LOCA to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

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

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure, except for loss of coolant accidents involving pipe breaks larger than the transition break size under 10 CFR 50.46a. For those pipe breaks only, neither a single failure nor the unavailability of offsite power need be assumed.

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

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure, except for analysis of loss of coolant accidents involving pipe breaks larger than the transition break size under 10 CFR 50.46a. For those pipe breaks only, neither a single failure nor the unavailability of offsite power need be assumed.

* * * * *

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

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure, except for analysis of loss of coolant accidents involving pipe breaks larger than the transition break size under 10 CFR 50.46a. For those pipe breaks only, neither a single failure nor the unavailability of offsite power need be assumed.

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Criterion 44--Cooling water. A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure, except for analysis of loss of coolant accidents involving pipe breaks larger than the transition break size under 10 CFR 50.46a. For those pipe breaks only, neither a single failure nor the unavailability of offsite power need be assumed.

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Criterion 50--Containment design basis. The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by

10 CFR 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

For licensees voluntarily choosing to comply with 10 CFR 50.46a, the structural and leak tight integrity of the reactor containment structure, including access openings, penetrations, and its internal compartments, shall be maintained for realistically calculated pressure and temperature conditions resulting from any loss of coolant accident larger than the transition break size.

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

7. 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 (42U.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. 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.

(i) Analysis and evaluation of emergency core cooling system (ECCS) cooling performance and the need for high-point vents following postulated loss-of-coolant accidents may be performed under the requirements of either § 50.46 or § 50.46a and § 50.46b of this chapter for designs certified after [EFFECTIVE DATE OF RULE] and demonstrated under § 50.46a(c)(2) of this chapter to be similar to reactor designs licensed before [EFFECTIVE DATE OF RULE], or

(ii) Analysis and evaluation of ECCS cooling performance and the need for high-point vents following postulated loss-of-coolant accidents must be performed under the requirements of §§ 50.46 and 50.46b of this chapter for designs that are not demonstrated under § 50.46a(c)(2) of this chapter to be similar to reactor designs licensed before [EFFECTIVE DATE OF RULE].

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9. In § 52.54, paragraph (b) is revised to read as follows:

§ 52.54 Issuance of standard design certification.

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(b) The design certification rule must specify the site parameters, design characteristics, and any additional requirements and restrictions of the design certification rule. A design certification rule which was reviewed and approved as meeting the requirements of 10 CFR 50.46a must specify the criteria governing departures that a referencing combined license must meet. The criteria must ensure that the safety bases for the NRC's approval of the certified design's compliance with § 50.46a (including applicability of the TBS) continue to apply despite the departure.

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

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

(a)   *   *   *

(5) An analysis and evaluation of the design and performance of structures, systems, and components with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents.

(i) Analysis and evaluation of ECCS cooling performance and the need for high-point vents following postulated loss-of-coolant accidents must be performed under the requirements of either § 50.46 or § 50.46a and § 50.46b of this chapter for facilities licensed after [EFFECTIVE DATE OF RULE] and demonstrated under § 50.46a(c)(2) of this chapter to be similar to reactor designs licensed before [EFFECTIVE DATE OF RULE], or

(ii) Analysis and evaluation of ECCS cooling performance and the need for high-point vents following postulated loss-of-coolant accidents must be performed under the requirements of §§ 50.46 and 50.46b of this chapter for facilities licensed after [EFFECTIVE DATE OF RULE] and not demonstrated under § 50.46a(c)(2) of this chapter to be similar to reactor designs licensed before [EFFECTIVE DATE OF RULE].

* * * * *

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

§ 52.137 Contents of applications; technical information.

(a) * * *

(4) An analysis and evaluation of the design and performance of SSCs with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of SSCs provided for the prevention of accidents and the mitigation of the consequences of accidents.

(i) Analysis and evaluation of ECCS cooling performance and the need for high-point vents following postulated loss-of-coolant accidents must be performed under the requirements of either § 50.46 or § 50.46a and § 50.46b of this chapter for designs approved after [EFFECTIVE DATE OF RULE] and demonstrated under § 50.46a(c)(2) of this chapter to be similar to reactor designs licensed before [EFFECTIVE DATE OF RULE], or

(ii) Analysis and evaluation of ECCS cooling performance and the need for high-point vents following postulated loss-of-coolant accidents must be performed under the requirements of §§ 50.46 and 50.46b of this chapter for designs that are not demonstrated under

§ 50.46a(c)(2) of this chapter to be similar to reactor designs licensed before [EFFECTIVE DATE OF RULE].

* * * * *

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

§ 52.157 Contents of applications; technical information in 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.

(i) Analysis and evaluation of ECCS cooling performance and the need for high-point vents following postulated loss-of-coolant accidents must be performed under the requirements of either § 50.46 or § 50.46a and § 50.46b of this chapter for facilities licensed after [EFFECTIVE DATE OF RULE] and demonstrated under § 50.46a(c)(2) to be similar to reactor designs licensed before [EFFECTIVE DATE OF RULE], or

(ii) Analysis and evaluation of ECCS cooling performance and the need for high-point vents following postulated loss-of-coolant accidents must be performed under the requirements of §§ 50.46 and 50.46b of this chapter for facilities licensed after [EFFECTIVE DATE OF

RULE] and not demonstrated under § 50.46a(c)(2) of this chapter to be similar to reactor designs licensed before [EFFECTIVE DATE OF RULE].

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Dated at Rockville, Maryland, this September day of 2, 2010.

For the Nuclear Regulatory Commission.

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R. W. Borchardt,
Executive Director for Operations