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DRAFT FINAL RULE LANGUAGE

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

After review of the public comments on the proposed rule and consideration of stakeholder views expressed at public meetings on June 28, 2006 and August 17, 2006, the NRC staff has prepared the following draft language for the final rule. Changes were made to the requirements for risk assessments and the operational requirements, along with other miscellaneous and editorial changes.

Additionally, in order to make the § 50.46a process applicable to new reactors which are similar to current operating reactors, the applicability provisions of draft final sections 50.46 and 50.46a, as well as the provisions in sections 50.34(a) and (b) addressing the content of the PSAR and the FSAR with respect to ECCS cooling performance, are written as amendments to the current rules, and do not reflect the proposed Part 52 rule published in March 2006. However, if the Section 50.46a rulemaking is published after a final Part 52 rule, the language of these sections will be conformed in the final § 50.46a rulemaking to reflect the final Part 52 rulemaking.

The draft final language with respect to applicability indicates, among other things, that the ECCS system for reactors designs approved as part of a design certification issued after December 28, 1974 but before the effective date of the rule, "must be designed" in accordance with either § 50.46 or § 50.46a. This language is not intended to require any change to existing design certifications. Rather, the language is intended to allow a design certification applicant to voluntarily seek an amendment of the design to allow for the use of section 50.46a.

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 rule language may be incomplete 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. No stakeholder requests for a comment period will be granted at this stage in the rulemaking process.

Any questions on the requirements may be addressed to the NRC rulemaking project manager, Richard Dudley (301-415-1116; rfd@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.

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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 Part 50.

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).

Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 (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 in accordance with the requirements of § 50.46 or § 50.46a, and § 50.46b for facilities for which construction permits, design approvals, design certifications, or manufacturing licenses may be issued after [EFFECTIVE DATE OF RULE] and are determined by the NRC 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 in accordance with the requirements of § 50.46 and § 50.46b for facilities for which construction permits, design approvals, design certifications, or manufacturing licenses may be issued after [EFFECTIVE DATE OF RULE] and are not determined by the NRC to have designs that are similar to the designs of reactors licensed before [EFFECTIVE DATE OF RULE].

* * * * *

(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 in accordance with the requirements of §§ 50.46 or 50.46a, and 50.46b for facilities whose operating licenses, combined licenses, design approvals, design certifications, or manufacturing licenses were issued after December 28, 1974, but before [EFFECTIVE DATE OF RULE], and for facilities whose operating licenses, combined licenses, design approvals, design certifications, or manufacturing licenses are issued after [EFFECTIVE DATE OF RULE] and are determined by the NRC 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 in accordance with the requirements of §§ 50.46 and 50.46b for facilities whose operating licenses, combined licenses, design approvals, design certifications, or manufacturing licenses are issued after [EFFECTIVE DATE OF RULE] and whose designs are not determined by the NRC to be similar to the designs of reactors licensed before [EFFECTIVE DATE OF RULE].

* * * * *

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 in accordance with 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 or manufacturing licenses are issued after [EFFECTIVE DATE OF RULE] and are determined by the NRC to have designs that are similar to the designs of reactors licensed before [EFFECTIVE DATE OF RULE]; for design approvals and design certifications under part 52 of this chapter issued after December 28, 1974, but before [EFFECTIVE DATE OF RULE]; and for design approvals and design certifications under part 52 of this chapter issued after [EFFECTIVE DATE OF RULE] and whose designs are determined by the NRC to be similar to designs of reactors licensed before [EFFECTIVE DATE OF RULE]. The ECCS system must be designed in accordance with the requirements of this section for facilities whose operating licenses, combined licenses, or manufacturing licenses are issued after [EFFECTIVE DATE OF RULE] and are not determined by the NRC to have designs that are similar to the designs of reactors licensed before [EFFECTIVE DATE OF RULE]; and for design approvals and design certifications which are issued after [EFFECTIVE DATE OF RULE] and whose designs are not determined by the NRC to be 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.

*   *   *   *   *

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) *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.

(2) *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 considered design-basis accidents. LOCAs involving breaks larger than the TBS are considered beyond design-basis accidents.

(3) *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.

(4) *Transition break size (TBS)* is a break of area equal to the cross-sectional flow area of the inside diameter of specified piping for a specific reactor. The specified piping for a pressurized water reactor is the largest piping attached to the reactor coolant system. The specified piping for a boiling water reactor is the larger of the feedwater line inside containment or the residual heat removal line inside containment.

(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 or manufacturing license is issued after [EFFECTIVE DATE OF RULE] and whose design is determined by the NRC to be similar to the designs of reactors licensed before [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 design approval or design certification under part 52 of this chapter was issued after December 28, 1974, but before [EFFECTIVE DATE OF RULE]; and to each light water reactor design approval or design certification under part 52 of this chapter issued after [EFFECTIVE DATE OF RULE] whose design is determined by the NRC 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 licensee of a facility desiring to implement this section shall submit an application for a license amendment under § 50.90 that contains the following information:

(i) A description of the method(s) for demonstrating compliance with the ECCS criteria in paragraph (e) of this section;

(ii) A description of the risk-informed evaluation process to be used in evaluating whether proposed changes to the facility accomplished under paragraph (d)(3) of this section meet the requirements in paragraph (f) of this section; including:

(A) a description of the licensee's PRA model and non-PRA risk assessment methods demonstrating compliance with paragraph (f) of this section, and

(B) a description of the methods and decisionmaking process for evaluating compliance with the risk criteria, defense-in-depth criteria, safety margin criteria, and performance measurement criteria.

(2) In addition to the information required by paragraph (c)(1) of this section, an applicant for a construction permit, operating license, design approval, design certification, manufacturing license, or combined license desiring to implement the requirements of this section shall 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 for why the recommended TBS is consistent with the intent of this section.

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

(i) 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); and

(ii) The risk-informed evaluation process is adequate for determining whether the acceptance criteria in paragraph (f) of this section have been met.

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(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 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) For LOCAs larger than the TBS, operation in an operating configuration not demonstrated to meet the acceptance criteria in paragraph (e)(4) shall not exceed seven days;

(3) Any facility change which requires NRC review and approval must, in addition to meeting applicable NRC requirements, also be evaluated by a risk-informed evaluation. A risk-informed evaluation must also be performed for any change to a structure, system, or component subject to the requirements of § 50.65 and for which no other NRC regulation establishes criteria for accomplishing the change. In either case, the risk-informed evaluation must demonstrate that the acceptance criteria in paragraph (f)(1) of this section are met;

(4) The licensee shall periodically assess the cumulative effect of changes to the plant, operational practices, equipment performance, and plant operational experience. The assessment must be based upon updated PRA and risk assessments. The assessment must be completed in a timely manner, but no less often than once every two refueling outages.

(e) *ECCS Performance.* Each nuclear power reactor 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 involving breaks at or below the TBS must meet the criteria in this paragraph, and must be approved for use by the NRC. Appendix K, Part II, 10 CFR Part 50, sets forth the documentation requirements for evaluation models for LOCAs involving breaks at or below the TBS. The analysis methods for LOCAs involving breaks larger than the TBS must be maintained, must be made available for inspection upon request by the NRC, and must include the analytical approaches, equations, approximations and assumptions.

(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 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 and must demonstrate that the acceptance criteria in paragraph (e)(4) of this section are satisfied. The analysis method must address the most important phenomena in analyzing the course of the accident. Sufficient supporting justification, including the methodology used, must be available to show that the analytical technique reasonably describes the behavior of the reactor system during a LOCA from the TBS up to the double-ended rupture of the largest reactor coolant system pipe. Comparisons to applicable experimental data must be made.

The evaluation must be performed for a number of postulated LOCAs of different sizes and locations 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.

(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 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) Risk-informed acceptance criteria.

(1) The risk-informed evaluation for any changes made under paragraph (d) of this section must demonstrate that, following the change:

(i) The total increases in core damage frequency and large early release frequency are small and the overall risk remains small;

(ii) Defense in depth is maintained, in part, by assuring that reasonable balance is provided among prevention of core damage, containment failure (early and late), and consequence mitigation; system redundancy, independence, and diversity are provided commensurate with the expected frequency of postulated accidents, the consequences of those accidents, and uncertainties; and independence of barriers is not degraded;

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

(iv) 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.

(2) *Requirements for risk assessment - PRA.* To the extent that a PRA is used in the risk-informed evaluation, the PRA must:

(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) Model severe accident scenarios resulting from internal initiating events occurring at full power operation;

(iii) Calculate CDF and LERF;

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

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

(vi) Be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC.

(3) *Requirements for risk assessment other than PRA.* To the extent that risk assessment methods other than PRAs are used to develop quantitative or qualitative estimates of changes to CDF and LERF in the risk-informed evaluation, a licensee shall justify that the methods used produce realistic results.

(g) *Reporting.*

(1) 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. 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:

(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 a change in the calculated oxidation, or the sum of the absolute value of the changes in calculated oxidation, equals or exceeds 0.4 percent oxidation; 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.

(2) For each change to or error discovered in a PRA model or in the application of such a model that results in the facility design and operation exceeding the requirements in paragraph (f)(1)(i) of this section, the licensee shall report the nature of the change or error and its estimated effect on CDF and LERF. Any change or error correction that results in the facility design and operation exceeding the requirements in paragraph (f)(1)(i) of this section is a reportable event as described in §§ 50.55(e), 50.72 and 50.73. The licensee shall report the change or error within 30 days and propose steps and a schedule to demonstrate compliance or bring plant design or operation into compliance with § 50.46a requirements. Any necessary changes to the facility under this paragraph shall not be deemed to be backfitting under any provision of this chapter.

(3) The licensee shall submit, as specified in § 50.4, a report containing a brief description of all changes evaluated with the risk-informed evaluation but implemented without prior staff review which result in increases in CDF or LERF which are greater than very small. The report must include a summary of the risk-informed evaluation of each, and must be submitted at intervals not to exceed 24 months.

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

(i) through (l) - [RESERVED]

(m) *Changes to TBS*. If the NRC increases the TBS specified in this section applicable to a licensee's nuclear power plant, each licensee subject to this section shall 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, must not be deemed to be backfitting under any provision of this chapter.

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.

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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

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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 § 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 § 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 § 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 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 § 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 § 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 § 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 § 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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