10-14-2004

DRAFT RULE LANGUAGE FOR § 50.46 ECCS LOCA REDEFINITION RULE

NOTE: Redline below shows changes from existing 10 CFR Part 50 regulations

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. 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 of this part for facilities for which construction permits may be issued after December 28, 1974, but before [EFFECTIVE DATE OF RULE]. Such analyses must be performed in accordance with the requirements of § 50.46 and § 50.46(b) for facilities for which construction permits may be issued after [EFFECTIVE DATE OF RULE], and design approvals and standard design certifications under part 52 of this chapter issued after [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. Analysis and evaluation of ECCS cooling performance following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of §§ 50.46 or 50.46a, and 50.46b for facilities for which a license to operate may be issued after December 28, 1974, but before [EFFECTIVE DATE OF RULE]. Such analyses must be performed in accordance with the requirements of § 50.46 and 50.46(b) for facilities for which construction permits may be issued after [EFFECTIVE DATE OF RULE], and design approvals and standard design certifications under part 52 of this chapter issued after [EFFECTIVE DATE OF RULE].

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3. In § 50.46, paragraph (a) is revised by adding an introductory paragraph, and paragraph (a)(1)(i) is revised 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). Reactors whose operating licenses were issued before [EFFECTIVE DATE OF RULE] must be designed in accordance with the requirements of either this section or § 50.46a. Reactors whose construction permits were issued prior to, but have not received operating licenses as of [EFFECTIVE DATE OF RULE], and those reactors whose construction permits are issued after [EFFECTIVE DATE OF RULE] must be designed in accordance with this section.

(1)(i) The ECCS system must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents 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 loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents 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 loss-of-coolant accident. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the 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.* 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 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.

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

(4) *Transition break size* is a break equivalent in area to a double ended rupture of a pipe whose inside diameter is 14 inches for PWRs and 20 inches for BWRs.

(b) *Applicability and scope.*

(1) The requirements of this section apply to each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircalloy or ZIRLO cladding for which a license to operate was issued prior to [EFFECTIVE DATE OF RULE], but do not apply to such 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 paragraph (d), with cooling performance calculated in accordance with an acceptable evaluation model under paragraph (c), 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) Each nuclear power reactor subject to this section must be provided with an ECCS that must be designed so that its ECCS calculated cooling performance following postulated loss-of-coolant accidents, as demonstrated with ECCS evaluation models, conforms to the criteria set forth in paragraph (d) of this section. The evaluation models must meet the criteria in this paragraph, and must be approved for use by the NRC. 10 CFR Part 50, Appendix K, Part II, sets forth the documentation requirements for evaluation models.

(1) *ECCS evaluation for LOCAs involving breaks at or below the transition break size.* ECCS cooling performance at or below the transition break size 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 (d)(1) are satisfied. The evaluation model must be utilized for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient

to provide assurance that the most severe postulated loss-of-coolant accidents involving breaks at or below the transition break size 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 loss-of-coolant accident. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (d)(1) of this section, there is a high level of probability that the criteria would not be exceeded.

(2) *ECCS evaluation for LOCAs involving breaks larger than the transition break size.* ECCS cooling performance for LOCAs involving breaks larger than the transition break size must be calculated in accordance with an evaluation model and must demonstrate that the acceptance criteria in paragraph (d)(2) are satisfied. The evaluation model must address the most important phenomena in analyzing the course of the accident. The model must be utilized for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents from the transition break size up to the double-ended rupture of the largest pipe in the reactor coolant system are analyzed. The evaluation model must include sufficient supporting justification to show that the analytical technique reasonably describes the behavior of the reactor system during a loss-of-coolant accident from the transition break size up to the double-ended rupture of the largest reactor coolant system pipe. These calculations may take credit for the availability of offsite power. Comparisons to applicable experimental data must be made. When the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (d)(2) of this section, there must be a reasonable level of probability that the criteria would not be exceeded.

(d) *ECCS acceptance criteria.* The following acceptance criteria must be used in determining the acceptability of ECCS cooling performance as determined in accordance with paragraph (c).

(1) *Acceptance criteria for LOCAs involving breaks at or below the transition break size.*

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

(ii) *Maximum cladding oxidation*. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. As used in this subparagraph 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 shall 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 shall 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 shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(iv) *Coolable geometry*. Calculated changes in core geometry shall 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 shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

(2) *Acceptance criteria for LOCAs involving breaks larger than the transition break size.*

(i) *Coolable geometry*. Calculated changes in core geometry shall 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 shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

(e) *Imposition of restrictions.* The Director 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 paragraphs (c) and (d) of this section.

(f) *Changes to facility, technical specifications, and procedures.*

(1) *Submission and approval process*. A licensee may request to make changes to its facility, technical specifications and procedures based upon the analyses of ECCS performance permitted under this section, by submitting an application for a license amendment, provided, however, that changes to the facility and procedures may be made without submitting an application or obtaining NRC review and approval if the requirements of paragraph (f)(6) are met. The application must contain the following information:

(i) A description of the proposed change, and a discussion of how the proposed plant or procedure change will affect the design and licensing basis;

(ii) A discussion demonstrating that the criteria in paragraph (f)(2) have been met, including a discussion of the technical adequacy of all modeling and assessment methodologies used;

(iii) Identification of the structures, systems and components, procedures, and activities affected by the proposed change;

(iv) A description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown (including the plant-specific probabilistic risk assessment (PRA), or other systematic evaluation techniques used to evaluate severe accident vulnerabilities) are adequate for estimating the total core damage frequency (CDF), large early release frequency (LERF), late release frequency (LRF), and changes to CDF, LERF and LRF;

(v) A description of all initiating events and the plant operating modes, an assessment of how each of these events and modes may be affected by the proposed changes, and the bases for this assessment including the manner in which the events and modes are addressed in the risk estimates in paragraphs (f)(3) and (4);

(vi) A list of permanent facility and procedure changes that may have an impact on the PRA but have not been incorporated into the PRA, the bases for not including the changes in the PRA, and a sensitivity study demonstrating that the significant accident sequences or contributors are not affected; and

(vii) An assessment whether the bases for a previous risk-informed change to the facility's structures, systems and components, procedures, or activities may be adversely affected in a significant manner by the change being proposed, together with any actions to be taken by the licensee to address any adverse effect.

(2) *Acceptance criteria for changes to facility, technical specifications and procedures.* The Commission may approve the licensee's request if it determines:

(i) The facility is able to mitigate a LOCA at the limiting location(s) involving breaks greater than the transition break size up to and including a double-ended rupture of the largest pipe in the reactor coolant system, such that:

(A) The analysis performed under paragraph (c)(2) of this section has demonstrated that the acceptance criteria in paragraph (d)(2) of this section are met under all at-power operating configurations.

(B) The integrity of the reactor containment structure, including access openings, penetrations, and its internal compartments, is maintained for the realistically calculated pressure and temperature conditions resulting from any loss of coolant accident greater than the transition break size.

(ii) The frequency of occurrence of pipe breaks larger than the transition break size at the facility, or the uncertainty in the frequency of occurrence of such pipe breaks, is not significantly increased, by assuring that:

(A) New reactor coolant system (RCS) pressure boundary degradation mechanisms are not introduced, nor is the likelihood or effect of known degradation mechanisms significantly increased; and

(B) The likelihood of detecting RCS boundary degradation is not reduced.

(iii) The total increases in CDF, LERF and LRF due to all facility, technical specification and procedure changes permitted as a result of this section are each sufficiently small;

(iv) An appropriate level of defense in depth is also provided by assuring that:

(A) Reasonable balance is provided among prevention of core damage, containment failure (early and late) and consequence mitigation;

(B) Over reliance on programmatic activities to compensate for weaknesses caused by the proposed facility or procedure change is avoided;

(C) System redundancy, independence, and diversity is provided commensurate with the expected frequency of postulated accidents, consequences of postulated accidents, and uncertainties;

(D) Independence of barriers is not degraded;

(E) Defenses against human errors are preserved; and

(F) Common cause failures are addressed.

(v) Sufficient safety margins are retained to account for uncertainties;

(3) *Requirements for Risk Assessment-PRA*. A technically adequate PRA must be used to demonstrate compliance with this section. To the extent that a PRA is used to demonstrate compliance with paragraph (f)(2) of this section, 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) Calculate CDF, LERF, and LRF;

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

(iv) Have sufficient technical adequacy (including consideration of uncertainty) and level of detail to provide confidence that the change in total CDF, LERF and LRF adequately reflects the plant and the effect of the proposed change on risk.

(4) *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, LERF, LRF, a licensee must justify that the methods used produce realistically conservative results.

(5) *Monitoring and Feedback*. Upon implementation of a change to the facility, technical specifications, or procedure under this section, the licensee must periodically re-evaluate and update its PRA required under paragraph (f)(3) to address subsequent changes to the plant, operational practices, equipment performance, plant operational experience, changes in the PRA model, revisions in analysis methods, model scope, data, and modeling assumptions. The re-evaluation and updating must be completed in a timely manner, but no longer than once every two refueling outages. The updated PRA must continue to meet the requirements in paragraph (f)(3). Based upon the PRA, the licensee shall take appropriate action to ensure that all changes accomplished under this section continue to meet the acceptance criteria in paragraph (f)(2). The re-evaluation and updating required by this section, and any necessary changes to the facility, technical specifications and procedures as a result of this re-evaluation and updating, shall not be deemed to be backfitting under any provision of this chapter.

(6) *Facility and procedures changes not requiring NRC review and approval*. A licensee may make changes to its facility and procedures based upon the analysis of ECCS performance permitted under this section without prior NRC review and approval; the provisions of § 50.59 are not applicable.

(i) *Submission and approval process*. A licensee who wishes to make changes to its facility and procedures without prior NRC review and approval must submit an application under § 50.90 to request NRC approval of a process for evaluating the acceptability of such changes. The application must contain the following information:

(A) A description of the licensee's PRA model and risk assessment methods for demonstrating compliance with paragraph (f)(6)(ii); and

(B) A description of the methods and decisionmaking process for evaluating compliance with the risk criteria and defense-in-depth criteria in paragraphs (f)(6)(ii)(D) and (f)(2)(iv).

(ii) *Acceptance criteria*. The NRC may approve a licensee's process for making changes to its facility and procedures without prior NRC review and approval, and a licensee may make such changes following such NRC approval if the process ensures that:

(A) The acceptance criteria in paragraphs (d)(2) and (f)(2) are met;

(B) A change is not made if licensee reporting under paragraph (h) is required;

(C) All risk contributors and initiators such as external events and shutdown are considered in demonstrating that the acceptance criteria are met; and

(D) Each facility or procedure change implemented without prior NRC approval individually constitutes an inconsequential increase in CDF, LERF and LRF, and all facility and procedure changes implemented without prior NRC approval, considered cumulatively, results in inconsequential increases in CDF, LERF and LRF. If the changes in CDF, LERF and LRF can be estimated using the existing plant-specific PRA, the change must be estimated and result in no more than an inconsequential increase. If a change cannot be estimated or some estimates are impractical to perform, a combination of qualitative and quantitative considerations must justify that the cumulative changes result in, at most, an inconsequential increase in CDF, LERF and LRF.

(7) *Operational requirements.* The acceptance criteria in paragraph (d) must not be exceeded under any allowed at-power operating configurations analyzed under paragraph (c), and the plant may not be placed in any at-power operating configuration not addressed under paragraph (c).

(g) *Documentation, change control, and records retention.*

(1) *ECCS analysis change*. The first change to the ECCS analysis performed in conformance with this section must be reflected in the ECCS analysis required by § 50.34(b) of this chapter, but need not include a supporting § 50.59 evaluation of the change. Thereafter, any changes to the ECCS analysis, as described in the FSAR, may be made if the requirements of this section and § 50.59 continue to be met.

(2) *Facility and procedures change*. The licensee shall document the bases for its application under paragraph (f)(1), or (f)(6) of this paragraph, and the bases demonstrating compliance with the acceptance criteria in paragraph (f)(2) and (f)(6). Upon either the approval of the change under paragraph (f)(2) or licensee implementation of the change under paragraph (f)(6), the licensee shall update the FSAR in accordance with § 50.71(e) of this part.

(h) *Reporting.*

(1) Each licensee shall estimate the effect of any change to or error in evaluation models or in the application of such models to determine if the change or error is significant. For this purpose, a significant change or error is:

(i) For LOCAs involving pipe breaks at or below the transition break size, one which results 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; and

(ii) For LOCAs involving pipe breaks at or below the transition break size, one which results in a change in the calculated oxidation, or the sum of the absolute value of the changes in calculated oxidation, equals or exceeds 0.4% oxidation, and

(iii) For LOCAs involving pipe breaks above the transition break size, one which results in a calculated peak fuel cladding temperature different by more than 300°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 300° F.

(2) For each change to or error discovered in an ECCS evaluation model or in the application of such a model that affects the temperature calculation, 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 § 50.46 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 paragraph (d) 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.

(3) As part of the PRA update under paragraph (f)(5), the licensee shall:

(i) compare the revised values of baseline CDF and LERF to those calculated under the last PRA model required by paragraph (f)(5). If the baseline CDF or LERF increases by 20% or more, the licensee shall report the change to the NRC. The report must be filed with the NRC no more than 60 days after completing the PRA update and must include a description of the relevant PRA updates performed by the licensee, and an explanation of the changes in the PRA modeling, plant design, or plant operation that led to the increase(s) in CDF or LERF; and

(ii) Determine the cumulative changes in CDF and LERF for changes in the facillity, technical specifications and procedures implemented under this section using the updated PRA model; and compare the revised values to the CDF and LERF values calculated under the previous PRA model required by paragraph (f)(5). If the cumulative change in CDF increases by $1x10^{-6}$ per year or more, or the cumulative change in LERF increases by $1x10^{-7}$ per year or more, the licensee shall report the change to the NRC. The report must be filed with the NRC no more than 60 days after completing the PRA update. The report must include a description of the relevant PRA updates performed by the licensee, an explanation of the changes in the PRA modeling, plant design, or plant operation that led to the increase(s) in CDF or LERF, a description of any corrective actions required under paragraph (f)(5) of this section, and a schedule for implementation.

(i) [RESERVED]

(j) *Changes to transition break size; changes to the facility, technical specifications and procedures.* If the transition break size specified in this section applicable to a licensee's nuclear power plant is increased, each licensee subject to this section shall perform the evaluations required by paragraph (c) of this section and reconfirm compliance with the acceptance criteria in paragraph (d). If the licensee cannot demonstrate compliance with the acceptance criteria, then the licensee must 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, shall not be deemed to be backfitting under any provision of this chapter.

NOTE: **Redline below shows changes from existing regulations in 10 CFR Part 50**.

BACKFITTING EXCLUSION FOR NRC RULEMAKING CHANGING TRANSITION BREAK SIZE IN 50.46A, OR A NRC ACTION IMPOSING A DIFFERENT TRANSITION BREAK SIZE ON A LICENSEE

[NOTE: 50.46a(j) contains backfitting exclusions for licensee's changes to a facility or procedures necessary as a result of a change to the transition break size. In addition, 50.46a(f)(5) contains a backfitting exclusion for changes to licensee-initiated changes to the facility, procedures and technical specifications necessary as a result of the reevaluation and updating requirements in 50.46a(f).]

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

§ 50.109 Backfitting.

(a) * * * * *
(b) Paragraph (a)(3) of this section shall not apply to:
(1) Backfits imposed prior to October 21, 1985; and
(2) Any changes to the transition break size specified in § 50.46a or as otherwise applied to a licensee.

6. In Appendix A to 10 CFR Part 50, under the heading, "CRITERIA," Criteria 17, 35, 38, 41, 44 and 50 are amended to read as follows:

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

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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, where a single failure of the onsite power supplies and electrical distribution system need not be assumed for plants under § 50.46a.

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 loss-of-coolant accident 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, where a single failure need not 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 loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure, except for loss of coolant accidents involving pipe breaks larger than the transition break size under § 50.46a, where a single failure need not be assumed for plants under § 50.46a.

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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 loss of coolant accidents involving pipe breaks larger than the transition break size under § 50.46a, where a single failure need not be assumed for plants under § 50.46a.

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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 loss of coolant accidents involving pipe breaks larger than the transition break size under § 50.46a, where a single failure need not be assumed for plants under § 50.46a.

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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. For plants under § 50.46, and for plants under § 50.46a with respect to loss of coolant accidents involving pipe breaks at or below the transition break size, 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.

7. In 10 CFR Part 50, Appendix K, paragraph 5 of Section II is revised to read as follows:

APPENDIX K TO PART 50 - ECCS EVALUATION MODELS

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II. REQUIRED DOCUMENTATION

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5. General Standards for Acceptability - Elements of evaluation models reviewed will include technical adequacy of the calculational methods. For models covered by § 50.46(a)(1)(ii), the review of technical adequacy will include compliance with required features of section I of this appendix K; and for models covered by § 50.46(a)(1)(i), assurance of a high level of probability that the performance criteria of § 50.46(b) would not be exceeded. For models covered by § 50.46a(c)(1), the review will include either compliance with the required features of section I of this appendix K, or assurance of a high level of probability that the performance criteria of § 50.46a(d)(1) would not be exceeded. For models covered by § 50.46a(c)(2), the review will include whether there is a reasonable demonstration that the criteria of § 50.46a(d)(2) would not be exceeded.