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U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Comments on Proposed Rule 10 CFR Part 50, RIN 3150-AH29, "Risk-Informed Changes to Loss-of-Coolant-Accident Technical Requirements (70 Federal Register 67598, November 7, 2005)

Framatome ANP appreciates the opportunity to comment on the proposed alterations to the rule governing the design of emergency core cooling systems (ECCS) and plant response to loss-of-coolant accidents (LOCA). Framatome ANP is aware of the comments and responses to NRC questions being provided by the Nuclear Energy Institute (NEI). In addition to our own comments and responses, offered in Attachment A, we endorse those of NEI.

Framatome ANP is happy to discuss our comments with the NRC staff or commission. Please contact myself, or Bert Dunn of my staff, for any assistance we can offer.

Sincerely,

Ronnie L. Gardner

Ronnie L. Gardner, Manager
Site Operations and Regulatory Affairs
Framatome ANP, Inc.

Enclosure

cc: G.S. Shukla
Project 728

Template = SECY-067

SECY-02

ATTACHMENT A

Comments, Discussion, Suggested Rule Language

Topics Identified by NRC for Public Comment

The NRC seeks specific public comments on 16 questions or issues. Framatome ANP is providing a specific response only to the first of these.

1. In proposed Sec. 50.46a(b), the Commission specifically precluded the application of the Sec. 50.46a alternative requirements to future reactors. However, future light water reactors might benefit from Sec. 50.46a. The Commission requests specific public comments regarding whether Sec. 50.46a should be made available to future light water reactors.

Comment: § 10CFR50.46a should be available to nuclear plants licensed after the publication of the rule that are of similar design to the current generation of operating boiling and pressurized light water nuclear power reactors. The group of advanced light water designs previously certified (ABWR, System 80+, AP600, AP1000), under design certification review (ESBWR) and in the pre-review process (US EPR) all fit into this category and can realize benefits from the application of § 10CFR50.46a. Licensing under § 50.46a should not, however, be without due consideration, that is, an active decision by the NRC staff that the design is substantially similar to currently operating nuclear power reactors should be required. The following consideration may be useful:

> The applicability, of the frequency of pipe rupture versus break size used as a basis for establishing the transition break size (TBS) in § 10CFR50.46a, to the new design should be established. (This is probably nothing more than ensuring that the piping has been designed to an appropriate code.)

Language to achieve this is provided in "Suggested Rule Language" at the end of these comments.

Comments on Proposed Rule

<u>§ 50.34 paragraphs (a)(4) & (b)(4); § 50.46 paragraph (a); § 50.46a paragraph (b)(1)</u>

As discussed in our response to NRC question 1, § 50.46a should be made available to future light water reactors. The language in the above paragraphs should be modified such that § 50.46a is applicable to future light water reactors.

<u>§ 50.46 paragraph (a) and § 50.46a paragraph (b)(1)</u>

The rule language, as drafted, perpetuates the specific inclusion of only zircaloy and ZIRLO cladding. This would continue the need for Framatome ANP's M5 cladding to be licensed by exemption. M5 is currently being used in 11 nuclear power reactors of varying designs across the US. Each of these plants and others in the near future continues to require the formality of an exemption for their license. It is obvious that M5 is an acceptable and desirable cladding material for use in nuclear power reactors. With a change to the regulations being made, it will serve efficiency to include M5 and eliminate the need for exemptions. The language in the above paragraphs should include M5 cladding.

This could be done in two ways:

> In an April 12, 2000 letter (Project Number: 689, David J. Modeen to David L. Meyer, Chief, Rules and Directives Branch) the Nuclear Energy Institute (NEI) suggested a wording change to § 10CFR50.46 that would apply the rule to any NRC-approved, Zirconium-based cladding. The specific references to Zircaloy and Zirlo would be removed.
>
> or
>
> M5 could simply be added to the acceptable cladding materials listed in § 10CFR50.46.

Changing to "NRC-approved, Zirconium-based cladding" is a better generic solution because it would allow for the use of cladding alloys now under development. However, this path may also involve some added review and development by the NRC. Because zircaloy, Zirlo, and M5 are the only cladding materials widely used in US commercial nuclear power reactors and because a second rule making, impacting § 10CFR50.46, is planned in the near future, it would perhaps be preferable to just add M5 to the list of applicable cladding materials and adopt the more generic language later, when the revised ECCS acceptance criteria are incorporated.

Language to achieve the direct addition of M5 is provided in "Suggested Rule Language" at the end of these comments.

<u>§ 50.46a(e)(2) *ECCS analyses for LOCAs involving breaks larger than the TBS.*</u>

The analysis or case requirements in Section § 50.46a(e)(2) for beyond the transition break size (TBS) evaluations are excessive. The desire, for this portion of the regulation, is to establish, in a reasonable way, that the plant remains able to mitigate a large break loss-of-coolant accident. It is unnecessary and inconsistent to elevate the consideration of break size effects beyond that of other portions or aspects of the evaluation that are to be treated as reasonable values. Under the proposed rule language, a full § 50.46 evaluation will be required for breaks of area less than the TBS. The results for these analyses can be extended to the smaller break sizes in the greater than TBS spectrum with assurance. Combining a reasonable selection of discharge coefficient (0.6) with the use of the 1994 ANS decay heat standard would roughly equate a 14 inch schedule 160 pipe area ($0.7 ft^2$), treated as below the TBS, with a $1.4 ft^2$ break, treated as a beyond TBS break. Similarly, at the upper end of the break spectrum, what used to be considered as an 8 to 9 ft^2 break of the cold leg will be the equivalent of a historical 5 ft^2 break. The requirement to perform sensitivity studies to identify a worst case break between these two limits seems unwarranted. It would be reasonable to just perform the full double area break or at most that break and one intermediate break. The only sensitivity required should be relative to break location. Historically, break location can have a substantial influence on the calculated results. This should be resolved prior to the greater than TBS calculation either by sensitivity studies or by reference to appropriate historical analyses.

The concern can be allayed by either altering the rule such that the identification of the most severe break size is not required or by inserting the concept of reasonable confidence that breaks within the beyond TBS spectrum will not pose consequences substantially more severe than those of the calculations performed. In the last case, the measure of substantial increase in consequence would be similar to that upon which reportability is based.

Language to achieve this is provided in "Suggested Rule Language" at the end of these comments.

Suggested Rule Language

§ 50.34(a)(4) *Preliminary Safety Analysis Report*

(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.

§ 50.34(b)(4) *Final Safety Analysis Report*

(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 LOCAs must be performed in accordance with the requirements of § 50.46 or 50.46a, and 50.46b.

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

(1)(i) Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy, ZIRLO, or M5 cladding must be provided with an emergency core cooling system (ECCS). Reactors must be designed in accordance with the requirements of either this section or § 50.46a.

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

(b)(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, ZIRLO, or M5 cladding.

(e)(2) 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. The evaluation must be performed for postulated LOCAs of different sizes and locations sufficient to provide reasonable confidence that any LOCA within the beyond TBS spectrum (break areas larger than the TBS up to the double-ended rupture of the largest pipe in the reactor coolant system) will not lead to consequences substantially more severe than those calculated. 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. 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 equipment may be assumed if supported by plant-specific data or analysis.