framatome

February 23, 2023
NRC:23:006

Secretary
U.S. Nuclear Regulatory Commission
Washington, DC 20555–0001
ATTN: Rulemakings and Adjudications Staff

Framatome Inc. Response to NRC Request for Public Comment on Petition for Rulemaking Requesting that the NRC Revise Regulations Regarding the Licensing Safety Analysis for Loss-of Coolant Accidents (Docket ID: NRC–2022–0178) (Federal Register Notice 87FR71531)

Dear Ms. Brooke P. Clark,

Framatome Inc. (Framatome) submits the enclosed comments for consideration by the U.S. Nuclear Regulatory Commission (NRC). These comments are in response to the Petition for Rulemaking (PRM), Docket No. PRM–50–124, (Docket ID: NRC-2022-0178). The PRM requests that the NRC amend the regulations of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," which limits peak cladding temperature and maximum cladding oxidation to satisfy General Design Criterion No. 35 of Appendix A to Part 50, "Emergency core cooling."

There are no commitments within this letter or its enclosures.

If you have any questions related to this information, please contact me by telephone at (434) 832-3347, or by e-mail at gayle.elliott@framatome.com.

Sincerely,

Gayle Elliott, Director
Licensing & Regulatory Affairs
Framatome Inc.

cc: Ngola Otto
Project 689

Attachments:
1. Framatome Comments on PRM-50-124 (Docket ID: NRC–2022–0178)

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Attachment 1

Framatome Comments on PRM-50-124 (Docket ID: NRC–2022–0178)

Framatome Inc. (Framatome) has reviewed the proposed rule change in PRM-50-124 and believes that the proposed rule change should not be pursued. The proposal is not well founded technically as discussed in the text below. A full replacement of the existing acceptance criteria is not simple nor justifiable when 10 CFR 50.46 provides adequate safety to the fleet. Research is continuing regarding the Fuel Fragmentation, Relocation, and Dispersal (FFRD) phenomena and when that research is more mature a rule change may be appropriate as part of a revision to the current draft rule change to 10 CFR 50.46c. Industry initiatives on high burnup and accident tolerant fuel designs may provide motivation for modification of 10 CFR 50.46c.

The PRM challenges the adequacy and the safety of the current fleet at moderate and high burnups, particularly due to the FFRD phenomena. The range of burnups meant by the Petitioner's term "moderate" is not clear. Fuel fragmentation occurs throughout the cycle and the general aspect of relocation and possible dispersal via the cladding rupture opening have been known since the 1980s. Due to the fragmentation characteristics and realistic expectations of the core-wide cladding response, widespread dispersal that could impact the overarching goal of 10 CFR 50.46 was not expected to be significant. More recent tests revealed that at very high burnups, there is a potential for the fragments to be much smaller (i.e., fine fragmentation). With the smaller size, the potential for dispersal increases. This led to more exhaustive industrywide FFRD testing and evaluations that the NRC continues to support. In the interim, the NRC determined in SECY-15-0148 (Reference [1]) that the plants continue to operate safely with existing burnup limits (~60 GWd/MTU) and current styles of operation.

The PRM relies on the position of the German regulator, RSK. The position appears to be based on a misinterpretation of the German licensing basis. In fact, while the criteria are not identical worldwide, they are generally consistent with 10 CFR 50.46, using surrogate peak cladding temperature (PCT) and maximum local oxidation (MLO) limits on the single worst rod ("hot rod") as a means to ensure that plant's emergency core cooling system (ECCS) design will adequately maintain core coolability following a LOCA.

- The original PCT and MLO limits of 2200°F and 17% equivalent cladding reacted (ECR) are surrogate criteria to guarantee that the fuel rod cladding exhibits a sufficient residual ductility after oxidation and quench. The underlying idea is that the mechanical loading, during and after LOCA, is difficult to assess and that a residual ductility implies a sufficient mechanical strength. It was recognized in the 1970s that the requirement for residual ductility cannot be met in the ballooned and burst region but limiting the ECR enables the rod to retain sufficient strength. As the PRM states, this is not straight-forward, but the criteria and their application to the single worst rod remain suitably conservative with respect to the overarching goal of core coolability. This relationship is well-understood by those responsible for developing the evaluation models (EMs) and ensuring the adequacy of a plant's ECCS design.

- The U.S. NRC has proposed draft rule 10 CFR 50.46c. It contains some modifications to better align the criteria to the degradation mechanisms (e.g., operational hydrogen-based ECR limit, breakaway oxidation) but retains the same underlying rationale. The surrogate measures of PCT and local oxidation are retained and the same treatment for the rupture region is justified. With current fuel products, the draft rule does not challenge plant safety. The French nuclear industry is also evaluating a modification to the historical LOCA criteria. It is similar in form and application to 10 CFR 50.46c but the values are derived from a strength-based approach, as opposed to residual ductility.
- In Germany (Reference [2]), the PCT limit is 1204°C (2200°F). The ECR limit is provided by either the U.S. NRC ECR(H) curve or the GRS ECR(H) curve. The GRS curve uses the hydrogen that originated under LOCA conditions, taking hydrogen content and secondary hydrogen uptake into account. RSK states that both are suitable for preventing fragmentation of the cladding. However, rather than disputing the NRC approach as the PRM suggests, it is the GRS approach that RSK stated is "currently not possible". The PRM also supports its proposed change to a core-wide rupture criterion on the basis that "safety analyses in Germany are required to show that no more than 1% of the fuel rods in a core would rupture during a small-break LOCA and no more than 10% of fuel rods in a core would rupture during a large-break LOCA." These criteria are not for core coolability assessments: the 10% failure rate (2A leak) and 1% (0.1A leak) are radiological criteria, as described in tab. 3.1c. of Reference [3].

The PRM proposes a limitation on the percent of core-wide ruptures for coolability, but there is no attempt to justify the values: "The 10% limit presumes that the core would remain coolable if 90% of the rods remained intact, but this could be confirmed or adjusted later if results from ongoing fuel fragmentation, relocation, and dispersal (FFRD) studies indicate a need." Using a percentage of the entire core does not make sense when only a limited portion of the core will be at dispersal-susceptible burnup levels. The distribution of ruptured rods in the core is also important: ruptured rods well distributed in the core have a different impact on core cooling than a cluster of ruptured rods. Additionally, this type of criteria is not universal to the 10 CFR 50.46 applicable light water reactor (LWR) fleet. Unlike the pressurized water reactor (PWR) fuel assemblies, the boiling water reactor (BWR) fuel bundle is enclosed in a fuel channel to direct coolant up through the fuel assembly and act as a bearing surface for the control rod. The means of core cooling also differ. The impact of swelling and rupture without dispersal is different between PWRs and BWRs. In the case of potential dispersal into the reactor coolant system (RCS), the consequences for local and core-wide coolability differ even more.

Rather than a replacement, criteria similar to the proposed criteria may be valuable as alternative or supplementary criteria to enhance the demonstration of core-coolability for specific scenarios and to provide a consistent standard for review. Even as an alternative though, it would not be simple: rule changes, methodology developments, and plant implementation all take significant time and effort. Major aspects that would need to be defined include the level of probability, the key features of an EM, and justification of the metric to core coolability. The metric for core coolability may depend on plant type, LOCA scenario, and evolving fuel design innovations (e.g., Accident Tolerant Fuel (ATF)). Introducing the proposal as a replacement would affect the entire industry, requiring the NRC staff, fuel vendors, and licensees to change the existing licensing basis, without increasing safety.

SECY-15-0148 (Reference [1]), which concluded that the industry was operating safely in light of FFRD, supported the NRC's position that 10 CFR 50.46c could proceed without explicitly addressing FFRD. The conclusion was strongly tied to the fuel management styles of 2015. The industry is moving toward fuel designs beyond existing operational limits, pursuing enrichment increases and burnup increases. Furthermore, while 10 CFR 50.46c was intended to be technology neutral, it is based on the cladding failure mechanisms associated with double-sided Zirconium based alloys, which differ for ATF fuels. The PCT and MLO surrogate measure and means to address them do not directly apply to ATF designs. The more evolutionary the design, the greater the difference in the failure mechanism, but even the near-term ATF designs would have to pursue exemption requests and large NRC reviews in order to fully capture their benefit. To encompass these current industry initiatives along with the NRC-initiative to be more risk-informed, the draft 10 CFR 50.46c should be updated. If the FFRD research indicates necessary regulatory action, a consistent approach could be a valuable addition for high burnup fuel designs, if appropriately correlated to the level of risk and specifically tied to coolability.

References

[1] NRC SECY-15-0148 "EVALUATION OF FUEL FRAGMENTATION, RELOCATION AND DISPERSAL UNDER LOSS-OF-COOLANT ACCIDENT (LOCA) CONDITIONS RELATIVE TO THE DRAFT FINAL RULE ON EMERGENCY CORE COOLING SYSTEM PERFORMANCE DURING A LOCA (50.46c)" November 30, 2015. ML15230A200.

[2] RSK, 2015. "Demonstration of residual ductility/residual strength by means of an ECR limit curve." Recommendation to the 476th meeting of the Reactor Safety Commission (RSK) of the Federal Office for Radiation Protection (Germany) (June 24, 2015).

[3] RS-Handbuch "Sicherheitsanforderungen an Kernkraftwerke" vom 22. November 2012, Neufassung vom 3. März 2015 (BAnz AT 30.03.2015 B2).