UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

July 22, 2020

Ms. Margaret M. Doane
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: REGULATORY GUIDE (RG) 1.236, PRESSURIZED-WATER REACTOR CONTROL ROD EJECTION AND BOILING-WATER REACTOR CONTROL ROD DROP ACCIDENTS

Dear Ms. Doane:

During the 674th meeting of the Advisory Committee on Reactor Safeguards, June 3-5, 2020, we reviewed the NRC staff's Draft Regulatory Guide (RG) 1.236, "Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents," April 28, 2020. Our combined Metallurgy and Reactor Fuels and Accident Analysis - Thermal-Hydraulics Phenomena Subcommittees also reviewed an earlier draft version of this RG, Draft Guide (DG) - 1327, during a meeting on October 21, 2016; and, after two rounds of public comment periods, the subject final draft RG 1.236 on May 5, 2020. During these meetings, we had the benefit of discussions with the staff. We also had the benefit of the referenced documents.

CONCLUSIONS AND RECOMMENDATIONS

1. RG 1.236 provides thorough, comprehensive guidance for analyzing reactivity-insertion accidents, and it should be issued.

2. Timely completion of RG 1.183 on source terms is a necessary complement to fully implement the guidance in RG 1.236.

3. Significant effort was made by the staff in reviewing experimental data to define limits in RG 1.236. Key background materials, methods, and rationale used to develop this guide should be captured and published as part of the Commission's Knowledge Management Program.

4. It is anticipated that this guide will be applied on a case-by-case basis as evolutionary changes are made to light water reactor (LWR) fuels. These include changes to fuel pellet and cladding structure, higher enrichments, non-UO_2 fuel forms, and higher burnups. We look forward to reviewing staff actions related to these and similar submittals.

BACKGROUND

Reactivity insertion (or initiated) accidents (RIAs) were recognized early in the development of commercial nuclear power as safety significant because of their potential to challenge fuel rod integrity, fuel bundle geometry and coolability, and the integrity of the reactor coolant pressure boundary. A postulated control rod (assembly) ejection) in a pressurized water reactor (PWR) due to a rupture of the control rod drive housing or nozzle, or a postulated control rod drop in a boiling water reactor (BWR) when a stuck control blade is decoupled from its drive mechanism, are largely considered to be the limiting RIAs for the current fleet.

To provide guidance on the matter, RG 1.77, “Assumptions Used for Evaluating Control Rod Ejection Accidents in Pressurized Water Reactors,” was first issued in 1974. The event sequence of concern was the uncontrolled rapid removal of a single control rod from the core, i.e., a control rod ejection. This results in a positive reactivity insertion that leads to a rapid local power excursion. This transient is initially mitigated by Doppler feedback in the fuel and delayed neutron effects, followed by reactor trip. The resultant rapid fuel pellet expansion, exacerbated by gaseous fission product swelling, can lead to cladding failure, especially at higher burnups when the fission product inventory is greatest. Based on data from Special Power Excursion Reactor Test (SPERT) experiments, RG 1.77 specified a value of 280 cal/g deposited in the fuel pellet as a conservative limit to avoid catastrophic fuel failure and fuel-coolant interaction, and as a basis for retaining core coolability. The guide also specified criteria on primary coolant pressure boundary integrity and limits, i.e., below ASME Boiler and Pressure Vessel Code Service Level C, and radiological dose consequences as well below guidelines in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 100 (later defined as 6.3 Rem TEDE [total effective does equivalent] in RG 1.183).

As more experimental data on the effects of rapid power excursions on fuel rod integrity and performance were developed from SPERT and the Power Burst Facility (PBF), the legacy guidance in RG 1.77 was found to be neither adequate nor conservative. These experiments identified additional failure mechanisms related to thermal stresses in the cladding and fuel pellet-cladding mechanical interaction (PCMI). The uptake of hydrogen by the zirconium cladding during normal operation was found to further exacerbate cladding failure. In addition, exceeding local heat flux limits could cause ductile failure of the cladding by ballooning from high temperatures and internal gas pressures, leading to cladding rupture and release of fission gas inventory to the primary coolant system. A new axial fuel pellet enthalpy limit of 230 cal/g was established as an interim criterion for determining core coolability, and a cladding failure threshold of 140 cal/g was set for irradiated rods, such that in combination, core damage would be minimal and short- and long-term cooling would not be impaired.

In the 1980s and 1990s, international in-pile test programs raised additional concerns about the performance of high burnup fuel under accident conditions. These included empirical databases from tests conducted at PBF and SPERT (US), the CABRI Research Reactor (France), the Nuclear Safety Research Reactor (NSRR – Japan), the Impulse Graphite Reactor (IGR – Russia), and the Fast Pulse Graphite Reactor (BIGR – Russia). In response to these findings, the NRC and industry modified failure threshold limits and evaluation methods to account for this new information, and the need for new regulatory guidance was recognized (RIL-0401).

In 2007, interim criteria and further guidance were issued for evaluating RIAs in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Section 4.2, “Fuel System Design,” Appendix B. Significant data from PBF and international test reactors and considerably improved analytical models prompted further revision to existing criteria and led to development of a new regulatory guide on this topic.

This new guide, RG 1.236, defines fuel cladding failure thresholds for ductile failure, brittle failure, pellet-clad mechanical interaction, and fuel melting, along with their impact on core geometry and coolability. It also describes methods and procedures that the NRC staff consider acceptable for analyzing a postulated PWR control rod ejection accident and a postulated BWR control rod drop accident. Finally, it establishes analytical limits and guidance for demonstrating compliance with applicable regulations. To facilitate implementation, this guide also provides acceptable analytical models for cladding hydrogen uptake and estimating transient fission gas release. The latter is used as an input in assessing radiological consequences, and refers to further guidance contained in RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.” This is a necessary companion regulatory guide that is a critical component of the overall evaluation of RIAs.

DISCUSSION

The staff’s thorough preparation of RG 1.236 reflects a comprehensive effort to incorporate up-to-date analytical methods and experimental data since the original RIA regulatory guide was issued in 1974. Significant effort was made by the staff to review the extensive experimental database and define the limits in RG 1.236. We recommend that the background materials and methods that informed the staff, along with the rationale used by the staff to develop this guide, be captured in a technical basis document and published as part of the Commission’s Knowledge Management Program.

Applicability

Guidance on fuel rod cladding failure thresholds, fission gas release fractions, and allowable limits on damaged core coolability within RG 1.236 is essentially constrained to LWR fuels using slightly enriched uranium dioxide ceramic pellets (up to 5 wt% uranium-235) loaded within cylindrical zirconium alloy-based cladding, with fuel rod average burnups up to a maximum of 68 GWd/MTU.

Fuel Rod Cladding Failure Thresholds

For the accident scenarios under consideration, as a function of the energy deposition level and heat transfer from the rod, the following phenomena can occur: the fuel rod may internally pressurize above local coolant conditions; fuel temperatures may increase and approach melting temperatures; rapid fuel pellet thermal expansion may promote PCMI-induced cladding failure, and local heat flux may exceed critical heat flux conditions, causing fuel cladding temperatures to rise, leading to other potential fuel failure mechanisms. The guide defines thresholds for the following failures: high-temperature cladding failure, PCMI-induced failure, and molten fuel cladding failure. To ensure a conservative assessment of onsite and offsite radiological consequences, each resulting failure mode should be evaluated and the total number of failed fuel rods estimated.

Nuclear and Thermal-Hydraulic Methods and Analysis

Methodology
Accident analyses for these events are traditionally performed using NRC-approved analytical models and application methodologies as found in the guidance in RG 1.203, “Transient and

Accident Methods." RG 1.236 specifies that computer codes used for analyses should be based on a coupled thermal-hydraulic and nuclear kinetics model. Fuel enthalpy calculations should account for burnup-related effects on reactor kinetics and fuel performance. In particular, the importance of two or three-dimensional flux characteristics and changes in flux shapes as they influence reactivity input and feedback, peak energy deposition, total energy, and gross heat transfer to the coolant are to be evaluated. Also, sensitivity studies on variations of the Doppler effect, power distribution, pellet radial power profile, fuel element heat transfer parameters such as fuel thermal and fuel-clad gap conductivity, and other relevant parameters (e.g., moderator coefficients and boron concentration) are also to be included in the analyses.

Initial Conditions
For both PWR and BWR events, the accident analyses should consider the full range of cycle operation from beginning of cycle to end of cycle and the full range of power operation from cold zero power, to intermediate power levels up to hot full-power conditions. The guidance provides extensive detail on assumptions for thermal-hydraulic conditions and neutronic parameters as a function of operating states, and on differential rod worths and timing and rate of scram insertion, as input to analyses.

Predicting Total Number of Rod Failures
The total number of fuel rod failures is then considered in the radiological dose assessment and is defined as the sum of all fuel rods failing any one of the cladding failure thresholds described in the guide.

Allowable Safety Limits

Damaged Core Coolability
Limiting peak radial average fuel enthalpy to prevent catastrophic fuel rod failure and to avoid molten fuel-coolant interaction is acceptable to the staff to demonstrate that there is limited damage to core geometry and that the core remains amenable to cooling. For fresh and low-burnup fuel rods, the guide indicates that the peak radial average fuel enthalpy restriction will likely be more limiting than the fuel melt restriction. However, medium-to high-burnup fuel rods are noted to be more likely to experience fuel melting in the pellet periphery under rapid power excursion conditions. For this case, fuel melting outside the centerline region should be precluded, a restriction likely to be more limiting than the peak radial average fuel enthalpy.

Radiological Consequences
Rather than include detailed methodology on radiological consequences, the guide refers to guidance in RG 1.183 and RG 1.195 to be used in estimating consequences of reactivity insertion accidents. Staff indicated that updates to RG 1.183 are in progress, and it is anticipated that a draft will be issued in 2021. As noted above, RG 1.183 is a necessary complement and a critical component of the overall evaluation of RIAs. We encourage its timely completion.

Reactor Coolant System Pressure
The maximum reactor coolant system pressure should be limited to ensure that system integrity is maintained. For PWRs, the guide states that no credit should be taken for the possible pressure reduction caused by the assumed failure of the control rod drive pressure housing.

Anticipated Future RG Applications

The cladding failure thresholds cited in the guide are conservative bounds for empirical databases that consider the impacts of fuel burnup for the fuel-cladding types tested. It is anticipated that this guide will be applied on a case-by-case basis as evolutionary changes are made to LWR fuels. These include changes to fuel pellet and cladding structure (e.g., doped pellets or coated claddings), higher enrichments (i.e., >5.0 wt% U-235), non-UO_2 fuel forms (e.g., MOX), and higher burnups in LWR fuel-cladding designs (i.e., >68 GWd/MTU). When applying this RG to high burnups or new fuel-cladding types, it is incumbent on the applicant to provide technical justification, and when necessary, submit confirmatory experimental data. We look forward to reviewing staff actions related to these and similar submittals.

SUMMARY

Regulatory Guide 1.236 provides thorough, comprehensive guidance for analyzing reactivity-insertion accidents, such as a postulated PWR control rod ejection or a postulated BWR control rod drop, to meet regulatory requirements on reactivity limits and associated radiation dose consequences. RG 1.236 should be issued, and we encourage timely completion of RG 1.183 as a necessary complement for evaluating RIAs.

We commend the staff's performance in the development of Regulatory Guide 1.236. The resultant product required a deliberate, long-term focus on relevant issue identification and resolution; it involved two public comment periods and significant stakeholder interaction; it derived in part from substantial international collaboration and cooperation; and it tapped into current, applicable NRC engineering and research resources.

We recommend that the background materials and rationale that went into developing this new guide should be captured and published as part of the Commission's Knowledge Management Program. This will aid the staff in their reviews of advanced reactor core designs incorporating new fuel forms with increased accident tolerance, higher enrichments, and increased burnups. We look forward to reviewing staff actions related to these and similar submittals, and completion of RG. 1.183.

Sincerely,

Matthew W. Sunseri Digitally signed by Matthew W. Sunseri
Date: 2020.07.22 18:26:23 -04'00'

Matthew W. Sunseri
Chairman

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