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

December 20, 2021

Mr. Daniel H. Dorman
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: RESEARCH INFORMATION LETTER (RIL) 2021-13 ON INTERPRETATION OF RESEARCH ON FUEL FRAGMENTATION, RELOCATION, AND DISPERSAL AT HIGH BURNUP

Dear Mr. Dorman:

During the 691st meeting of the Advisory Committee on Reactor Safeguards (ACRS), November 30 - December 2, 2021, we completed our review of the Research Information Letter (RIL) for fuel fragmentation, relocation, and dispersal (FFRD) during a loss-of-coolant accident (LOCA). Our Metallurgy and Reactor Fuels Subcommittee reviewed this topic on November 17, 2021. During these meetings, we had the benefit of discussions with the U.S. Nuclear Regulatory Commission (NRC) Research staff. We also had the benefit of the referenced documents.

CONCLUSIONS AND RECOMMENDATIONS

1. The current data set on FFRD has been expanded. However, there remains a significant degree of uncertainty in large part because the problem is multivariate and the experiments from which the data were developed did not always represent actual light water reactor (LWR) conditions. Our letter suggests a number of cautions that should be considered when applying this RIL. They are described in detail in our final thoughts section.

2. The staff recognized, and we agree, that this document would benefit from additional context by identifying the entire scope necessary to resolve the safety issues related to FFRD and describing the role of the RIL as a specific piece of that overall scope.

3. A risk informed approach should be undertaken that examines both the likelihood of expected event conditions combined with a more complete modeling evaluation of FFRD consequences. This activity could add substantial value to future research program development and to the regulatory decision-making process.

BACKGROUND

The RIL evaluates the publicly available experimental data related to fuel fragmentation, relocation, and dispersal at high burnup during a loss-of-coolant accident (LOCA) and is an update to LOCA research that has been ongoing for over 40 years. It began with study of cladding embrittlement and performance during a LOCA and has evolved to include fuel fragmentation and dispersal at high burnup. Several documents related to this topic have been published. These include:

1. NUREG/CR-6967, "Cladding Embrittlement During Postulated Loss-of-Coolant Accidents," July 2008,

2. RIL-0801, "Technical Basis for Revision of Embrittlement Criteria in 10 CFR 50.46," May 2008,

3. NUREG-2121, "Fuel Fragmentation, Relocation, and Dispersal during the Loss-of-Coolant Accident," March 2012,

4. 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 2015, and

5. ACRS Letter "Draft Final Rule 10 CFR 50.46C, 'Emergency Core Cooling System Performance During Loss-of-Coolant Accidents (LOCA)' and Associated Regulatory Guides," February 2016.

The NRC's current research studies on FFRD seek to gain a better understanding of fuel performance, under LOCA conditions, in particular for fuel with burnup greater than the current license limit of approximately 62 GWd/MTU. The RIL's purpose is to provide a current assessment of the results of the latest experimental programs on FFRD and to suggest conservative, empirical thresholds for FFRD-related phenomena.

It is intended that the RIL will serve as a foundation for the next steps in evaluating the effects of FFRD on LOCA performance at high burnup, which may include developing regulatory guidance.

DISCUSSION

The goal of the analysis performed by the Office of Nuclear Regulatory Research staff was to provide important and timely interpretations of a complex technical issue regarding high burnup fuel behavior during a LOCA. To this end the staff sponsored or participated in experimental programs to determine the potential impacts of FFRD on fuel performance at and beyond the current approved limits. These include the Studsvik Cladding Integrity Program (SCIP)-III research conducted at U.S. National Laboratories and the Halden reactor. Experimental results from the literature as well as other international laboratories were also included. The results of the analysis:

1. Suggest an empirical threshold at which fuel pellets become susceptible to fine fragmentation.

2. Identify a local cladding strain threshold above which fuel relocation becomes a concern as a result of a LOCA.

3. Provide a conservative estimate for the mass fraction of "dispersible" fuel as a function of burnup.

4. Provide an estimate for a range of packing fractions of relocated but nondispersed fuel in the balloon region that may affect heat transfer.

5. Evaluate the potential for significant transient Fission Gas Release (tFGR) that may impact ballooning and burst behavior of high-burnup fuel under LOCA conditions.

Studsvik LOCA Accident Testing

From 2009 through 2011, NRC sponsored LOCA accident testing experiments at Studsvik Nuclear Laboratory in Sweden. The experimental program included:

1. Six single-rodlet integral LOCA tests,

2. Four rodlets with segment burnup ranging from 72 to 78 GWd/MTU, and

3. Two rodlets with segment burnup around 60 GWd/MTU.

The pressurized, high-burnup, fueled rod segments were subjected to a temperature transient in a steam environment to induce ballooning, burst, and high-temperature steam oxidation. However, the rodlets were not subject to nuclear heating. The tests were conducted in a hot-cell facility. A 30-cm rodlet was externally heated using an infrared furnace in a flowing steam environment from 300 degrees Centigrade (C) to the target temperature (either 950 degrees C or 1,185 degrees C, depending on the test) at a rate of 5 degrees C per second. Internal pressures bounded a typical end-of-life rod to induce ballooning and burst, with burst hoop strains in the range of 25 to 55 percent (%).

After the LOCA simulation, the rods were subjected to four-point bend tests to measure the residual mechanical behavior of the ballooned and burst region. Subsequently, a "shake test" determined the mobility of fuel fragments that remained within the fuel rod.

Studsvik Cladding Integrity Programs (SCIP)

The NRC also participated in the SCIP-III program. Some of the SCIP-III tests used the same equipment built for the earlier NRC-sponsored work, while other tests used a newly designed test device with similar features. The SCIP-III project generated 18 test results similar to the six NRC-sponsored tests, to further evaluate how various parameters including fuel burnup and microstructure, cladding strain, temperature, internal gas pressure and gas flow at the time of burst, and magnitude of tFGR effect FFRD. The experimental methods developed in the third phase of the SCIP-III are continuing to be used in the SCIP-IV international research project. NRC remains a participant in the project.

Oak Ridge National Laboratory (ORNL)

In 2019, three hot-cell integral LOCA tests were conducted in the Severe Accident Test Station at ORNL. The segment average burnup of the three tests conducted at ORNL ranged from 69 to 77 GWd/MTU. The segments were harvested from parent rods with average burnups ranging from 63 to 68.5 GWd/MTU. Experimental methods were comparable to those used at Studsvik. Following the LOCA simulation, fuel fragments were shaken out of the rods and examined to determine the particle size distribution.

Halden Reactor Project

Thirteen LOCA tests were conducted with segments of fuel rods irradiated in commercial power reactors. Seven tests were on pressurized water reactor (PWR) fuel rodlets (IFA-650.3/4/5/9/10/15/16), four tests were on boiling water reactor (BWR) fuel rodlets (IFA-650.7/12/13/14), and two tests were on Russian VVER reactor fuel rodlets (IFA-650.6/11). Rodlet burnup values ranged from 44 to 92 GWd/MTU. Heatup rates varied from 2 to 6 degrees C per second. For some cases, heater power was slightly adjusted during the transient to obtain the desired target peak cladding temperature (PCT). The overall range of PCT for the test series was from 800 to 1,200 degrees C. A key difference between the Halden and other tests is that, in the Halden tests, nuclear heating was used.

The results of the Halden tests exhibited considerable variability. For example, Halden LOCA test IFA-650.14 was subjected to more prototypical LOCA conditions but did not burst. However, significant tFGR (18.6%) was observed during the test.

Axial fuel relocation and packing were observed during test IFA-650.9 in a high-burnup PWR rod when subjected to LOCA conditions. Post-test gamma scans showed that a significant portion of the fuel stack was missing due to axial fuel relocation and dispersal. The relocated fuel had dropped to the lower portion of the rod near the burst opening.

The results of the Halden tests suggest that it is not clear whether tests using rodlets and a furnace are truly conservative or comparable relative to in-pile LOCA conditions.

General Results-Discussion

The results of the staff's analysis for the experimental conditions of the data base suggested that FFRD manifests itself in regions of the core with the following specific characteristics:

1. It appears that fine fragmentation is limited to regions with burnups above 55 GWd/MTU pellet average burnup.

2. Axial fuel relocation was observed in regions of the fuel rod with a local cladding strain greater than 3%. Relocated fuel fragments could occupy between 60% and 85% of the fuel rod cross-sectional area in the balloon region. The propensity for fuel dispersal was correlated with fuel fragment size and burst opening size. However, cladding burst and fuel relocation were prerequisites.

3. The data suggest that significant quantities of fission gas may be released during a LOCA transient. Transient fission gas release becomes increasingly significant with increasing burnup, with releases as high as 20% percent observed from a fuel rod segment with an average burnup of 70 GWd/MTU.

The above observations provided the bases for the staff to suggest that FFRD should be evaluated at rod average burnup limits below the current limit of 62 GWd/MTU. Moreover, the RIL provided suggestions for models that should be considered in such an evaluation. The staff proposed a conservative approach to FFRD. Fuel fragmentation should be assumed for rods that exceed burnups of 55 GWd/MTU. The subsequent potential for relocation and dispersal should be assessed once a calculated cladding strain of 3% is reached during the LOCA. The staff acknowledges that the model uncertainty is high given limited data but may be considered reasonably conservative for this data set.

The staff also questioned whether the publicly available experiments adequately represent the conditions affecting FFRD in the design basis LOCA event and whether these experimental results are truly conservative. External peer review group members suggested that this concern was not warranted. However, the staff concludes that much uncertainty in performance data and modeling remains.

Final Thoughts on Future Work

A number of cautions should be considered going forward:

1. The conditions of the experiments often differed significantly from conditions that would exist at PWR operating conditions. Depending on the specific test, key variables that were not always prototypic include: linear heat generation rate (low), terminal temperature (high) and heatup rate (low).

2. The fuel/thermal variables influencing fragmentation and relocation phenomena are many. While fuel burnup is easily calculated, and verified by measurement, the cladding strain must be calculated - a calculation fraught with uncertainty. Fuel pellet cracking occurs almost immediately on initial startup. The evolution of fuel microstructure during burnup, including the rim, also introduces uncertainty.

3. Operational variables, including flow-induced vibration (normal and accident) can strongly influence the fragmentation and relocation process as well as dispersal after cladding breach.

4. It is one thing to determine an observed burnup at which fuel fragmentation begins to occur. In this case, it is essential to determine the difference between conditions of a test/examination and the conditions that would exist under actual operating conditions. A careful evaluation of uncertainty will be key. It is quite another thing to determine the point at which FFRD actually influences LOCA performance. The RIL made no claims in this area, one that is critical to determination of practical consequences. There are a number of experimental programs that will produce data that more closely approach PWR conditions including the TREAT tests. These results will likely provide more prototypic data and hence reduce uncertainty.

5. Lastly, the larger picture should be considered. The document would benefit from additional context by identifying the entire scope necessary to resolve the safety issues related to FFRD and describing the role of the RIL as a specific piece of that overall scope. Because this RIL suggests a numerical threshold burnup for the onset of fragmentation that is different than that used presently for licensing actions, the document should identify the purpose for transmitting this different burnup limit to the Office of Nuclear Reactor Regulation and identify plans to provide additional information relative to its potential impact on any future regulatory activity.

   Additional experimental programs and data sets that examine a broader range of transient conditions and fuel types are important to resolve these remaining uncertainties and address these cautions. In addition, we recommend a risk-informed approach be undertaken that examines both the likelihood of expected event conditions combined with a more complete modeling evaluation of FFRD consequences. This activity could add substantial value to future research program development and to the regulatory decision-making process.

We look forward to the results of additional tests (SCIP-IV, TREAT) that will shed further light on the FFRD phenomena and its likely effect on fuel performance under LOCA conditions at high burnup.

Member Rempe recused herself from the deliberation of RIL 2021-13.

Sincerely,

Signed by Sunseri, Matthew on 12/20/21

Matthew W. Sunseri, Chairman

REFERENCES

1. U.S. Nuclear Regulatory Commission (NRC), RIL 2021-13, "Interpretation of Research on Fuel Fragmentation, Relocation, and Dispersal at High Burnup," 2021. (ADAMS Proprietary Accession No. ML21313A110)

2. Framatome Presentation, "Fuel Fragmentation Relocation and Dispersal (FFRD) Status Meeting." (ADAMS Accession Nos. Package ML20335A115 [Non-publicly Available Accession No. ML20335A113], [Publicly Available Accession No.ML20335A087])

3. Framatome Presentation, "Radiological Increased Burnup Meeting," November 16, 2020. (ADAMS Package Accession No. ML20343A028 [Non-publicly Available Accession No. ML20344A437], [Publicly Available Accession No.ML20344A439])

4. Framatome Presentation, "Fuel Particle Transport in Fuel Fragmentation and Dispersal (FFRD) Evaluation for Higher Burnup Fuels," February 24, 2021. (ADAMS Package Accession No. ML21050A328 [Non-publicly Available Accession No. ML21050A331], [Publicly Available Accession No. ML21050A332])

5. Framatome Presentation, "Framatome Fuel Fragmentation Relocation and Dispersal High Burnup Topical Report Meeting," May 25, 2021. (ADAMS Accession No. ML21141A235 [Non-publicly Available Accession No. ML21141A259], [Publicly Available Accession No.ML21144A011])

6. Framatome Presentation, "Meeting for Framatome Increased Burnup Topical Report Discussions on Fuel Fragmentation, Relocation and Dispersal (FFRD), Criticality, and Core Coolability," October 26, 2021. (ADAMS Package Accession No. ML21256A006 [Non-Publicly Available Accession No. ML21256A013], [Publicly Available Accession No. ML21256A007])

7. Bianco, A., et al., 2015. "Experimental investigation on the causes for pellet fragmentation," *Journal of Nuclear Materials,* pp. 260–267.

8. Billone, M., Y. Yan, T. Burtseva, and R. Daum, 2008. NUREG/CR-6967, "Cladding Embrittlement During Postulated Loss-of-Coolant Accidents," Washington, DC, June 2008.

9. Blumberg, M., and M. Smith, 2021. "Revision of Regulatory Guide 1.183, 'Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors'," Presentation on March 5, 2021. (ADAMS Accession No. ML21056A058)

10. Capps, N., et al., 2021. ORNL/SPR-2021/2218, "Engineering Assessment of UO2 and Cladding Behavior under High-Burnup LOCA Conditions," Oak Ridge National Laboratory, Oak Ridge, TN.

11. Capps, N., et al., 2021. "A Critical Review of High Burnup Fuel Fragmentation, Relocation, and Dispersal under Loss-of-Coolant Accident Conditions," *Journal of Nuclear Materials*, Vol. 546, p. 152750.

12. Capps, N., et al., 2020. "Integral LOCA fragmentation test on high-burnup fuel," *Nuclear Engineering and Design,* p. 110811.

13. Chung, H.M., and T.F. Kassner, 1978. NUREG/CR-0344, "Deformation Characteristics of Zircaloy Cladding in Vacuum and Steam under Transient-Heating Conditions: Summary Report," Washington, DC, July 1978.

14. Esmaili, H., 2021. "Letter Report on Evaluation of the Impact of Fuel Fragmentation, Relocation, and Dispersal for the Radiological Design Basis Accidents in Regulatory Guide 1.183," Memorandum from H. Esmaili (NRC) to K. Hsueh (NRC), July 20, 2021. (ADAMS Accession No. ML21197A067)

15. Geelhood, K., et al., 2021. PNNL-31160, "FAST-1.0.1: A Computer Code for Thermal-Mechanical Nuclear Fuel Analysis under Steady-state and Transients," Pacific Northwest National Laboratory, Richland, WA.

16. Geelhood, K.G., and W.G. Luscher, 2019. "Degradation and Failure Phenomena of Accident Tolerant Fuel Concepts," Pacific Northwest National Laboratory, Richland, WA.

17. Jernkvist, L.O., 2019. "Modelling of fine fragmentation and fission gas release of UO2 fuel in accident conditions," *EPJ Nuclear Sciences & Technologies,* Vol. 5, p. 11.

18. Karlsson, J., et al., "SCIP III—Test Method Descriptions Studsvik/N-15/315 Studsvik-SCIP III-196," Studsvik Nuclear AB, Nykoping, Sweden.

19. Khvostov, G., 2020. "Analytical criteria for fuel fragmentation and burst FGR during a LOCA," *Nuclear Engineering and Technology,* 52(10), pp. 2402–2409.

20. Linton, K., Y. Yan, Z. Burns, and Terrani, K., 2017. ORNL/SPR-2017/434, "Hot Cell Installation and Demonstration of the Severe Accident Test Station," Oak Ridge National Laboratory, Oak Ridge, TN.

21. Ma, Z., K. Shirvan, Y. Wu, and G. Su, 2020. "A three-dimension axial fuel relocation framework with discrete element method to support burnup extension," *Journal of Nuclear Materials,* Vol. 541, p. 152408.

22. Magnusson, P., et al., 2016. "A study of transient FGR by integral LOCA tests," *Top Fuel 2016.*

23. Magnusson, P., et al., 2020. "STUDSVIK/N-19/105 STUDSVIK-SCIP III-253—Subtask 1.1: Fuel fragmentation, relocation and dispersal, Final Summary Report," Studsvik Nuclear AB, Nykoping, Sweden.

24. Mileshina, L., and P. Magnusson, 2019. "STUDSVIK-SCIP III-235, Integral LOCA test to study burnup threshold on fuel fragmentation: 09-OL1L04 LOCA2," Studsvik Nuclear AB, Sweden.

25. Mileshina, L., and P. Magnusson, 2019. "STUDSVIK-SCIP III-240, Integral LOCA test to study burnup threshold on fuel fragmentation: WZR0067 LOCA1," Studsvik Nuclear AB, Sweden.

26. Mileshina, L. and P. Magnusson, 2018. "STUDSVIK-SCIP III-216, Integral LOCA test to study cladding strain and non-depressurization effects on fuel fragmentation: R2D5 LOCA3," s.l.: Studsvik Nuclear AB.

27. Mileshina, L. and P. Magnusson, n.d. "Studsvik/N-18/074 Studsvik-SCIP III-225 Integral LOCA test to study effect of pressure on fuel fragmentation: VUL2 LOCA2," s.l.: Studsvik Nuclear AB.

28. Moreno, R.O., I.G. Cabezón, P.J.G. Sedano, and Y.T. Gómez, 2005. "Fuel Relocation Effects in BWR LOCA Conditions," 2005 Water Reactor Fuel Performance Meeting, Kyoto, Japan, s.n.

29. Parsons, P., E. Hindle, and C. Mann, 1986. "The Deformation, Oxidation and Embrittlement of PWR Fuel Cladding in a Loss-of-Coolant Accident: A State-of-the-Art Report," Committee on the Safety of Nuclear Installations, OECD Nuclear Energy Agency, Paris, France.

30. Pontillon, Y., et al., 2004. "Experimental and Theoretical Investigation of Fission Gas Release from UO2 up to 70 GWd/t under Simulated LOCA Type Conditions: The GASPARD Program," *Proceedings of the 2004 International Meeting on LWR Fuel Performance.*

31. Powers, D., and R.O. Meyer, 1980. NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis," Washington, DC, April 1980.

32. Raynaud, P., 2013. "Core-Wide Estimates of Fuel Dispersal During a LOCA," Charlotte, NC, USA, s.n., pp. 636–643.

33. Raynaud, P., and I. Porter, 2014. "Predictions of fuel dispersal during a LOCA," Sendai, Japan, s.n., pp. 14–17.

34. Rest, J., et al., 2019. "Fission gas release from UO2 nuclear fuel: A review," *Journal of Nuclear Materials,* Vol. 513, pp. 301–345.

35. Snead, M., et al., 2015. ORNL/TM-2015/556, "Severe Accident Test Station Design Document," Oak Ridge, TN: Oak Ridge National Laboratory.

36. Studsvik Nuclear AB, n.d. "SCIP III—LOCA test data sheet—09-OL1L04-LOCA -v17," s.l.:s.n.

37. Tejland, P., and C. Sheng, 2019a. “Studsvik-SCIP-III-243, Integral BWR H3 overheating test of the fuel rod 12-R1E10,” Nykoping, Studsvik Nuclear AB, Sweden.

38. Tejland, P., and C. Sheng, 2019b. “Studsvik-SCIP-III-233, Integral overheating test to study TFGR of the fuel rod 09-OL1L04,” Nykoping, Studsvik Nuclear AB, Sweden.

39. Tradotti, R., 2014. “LOCA Testing at Halden, The BWR Fuel Experiment IFA-650.14,” OECD Halden Reactor Project, Halden, Norway.

40. Turnbull, J.A., et al., 2015. “An Assessment of the Fuel Pulverization Threshold During LOCA-Type Temperature Transients,” *Nuclear Science and Engineering,* pp. 477–485.

41. Une, K., S. Kashibe, and K. Hayashi, 2002. “Fission gas release behavior in UO2 fuels with developed rim structure,” *Journal of Nuclear Science and Technology,* 39, pp. 668-674.

42. U.S. Nuclear Regulatory Commission, 2012. NUREG-2121, “Fuel Fragmentation, Relocation and Dispersal During the Loss-of-Coolant Accident,” Washington, DC, March 2012.

43. U.S. Nuclear Regulatory Commission, 2013. NUREG-2160, “Post-Test Examination Results from Integral, High-Burnup, Fueled LOCA Tests at Studsvik Nuclear Laboratory,” Washington, DC, August 2013. (ADAMS Accession No. Publicly Available ML1340A256)

44. U.S. Nuclear Regulatory Commission, 2015. 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),” Washington, DC, November 30, 2015. (ADAMS Accession No. Publicly Available ML15230A200)

45. U.S. Nuclear Regulatory Commission, 2020. Regulatory Guide 1.236, "Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents," Washington, DC, June 2020. (ADAMS Accession No. Publicly Available ML20055F490)

46. Wiesenack, W., 2013. “HRP-380, Summary of the Halden Reactor Project LOCA Test Series IFA-650,” Halden Reactor Project, Halden, Norway.

47. Wiesenack, W., 2015. “HPR-383, Summary and Comparison of LOCA Tests with BWR Fuel in the Halden Reactor Project Test Series IFA-650,” Halden Reactor Project, Halden, Norway.

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