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MEMORANDUM TO: Joseph Donoghue, Director
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FROM: Michael Case, Director ***/RA K. Webber for/***
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SUBJECT: APPLICABILITY OF SOURCE TERM FOR ACCIDENT TOLERANT FUEL, HIGH BURN UP AND EXTENDED ENRICHMENT

Consistent with the Office of Nuclear Reactor Regulation (NRR) Use Need Requests (UNRs), NRR-2019-0009, “Regulatory Research Supporting Licensing of Burnup and Enrichment Extensions in Near-Term Accident Tolerant Fuel (ATF)” (Agencywide Documents Access and Management System (ADAMS) Accession Number ML19171A205) and NRR-2019-010, “Regulatory Research Supporting Licensing of Near-Term Accident Tolerant Fuel (ATF)” (ADAMS Accession Number ML19171A381), staff from the Fuel and Source Term Code Development Branch (FSCB) has reviewed recent analysis and evaluated its applicability to the subject UNRs. Specifically, FSCB staff has evaluated the applicability of Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors” for:

- Burnups up to 68 GWd/MTU excluding potential impacts related to fuel fragmentation, relocation, and dispersal;
- Enrichment between 5-8 percent;
- Chromium-coated cladding (Cr-coated); and,
- Chromia-doped fuel.

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For these concepts, FSCB staff recommends using accident source terms from Sandia National Laboratories (SNL) report SAND2011-0128 and non-loss-of-coolant accident (non-LOCA) source terms based on FAST calculations (similar to those calculated in the proposed update to RG 1.183, Draft Guide 1199) to serve as a basis for a future RG 1.183 update. The rationale for this determination is discussed below.

FSCB staff is also evaluating whether there would be significant impacts to the fractional releases and release timings in the Regulatory Guide for FeCrAl cladding as discussed below.

This letter does not supersede the scope of work defined in NRR-2019-0009 and NRR-2019-0010. The staff will continue with the following work defined in the UNR:

- Update MELCOR for FeCrAl material properties and conduct verification and validation (V&V)/code assessment;
- Complete an assessment of isotopic calculational capabilities and assessment of available validation data;
- Complete source term calculations for representative pressurized water reactors (PWR) and boiling water reactors (BWR) for Cr-coated cladding;
- Complete source term calculations for representative PWR and BWR for FeCrAl cladding;
- Complete a severe accident literature review, expert elicitation, and final phenomena identification and ranking table (PIRT) report; and
- Complete confirmatory calculations to support vendor topical reports for high burnup fuel.

Note that the target dates for the high burnup, Cr-coated cladding, and FeCrAl cladding source term calculations will likely be adjusted relative to the dates in the UNRs based on current industry plans. In particular, FSCB is looking to accelerate the high burnup work based on public statements made by the Nuclear Energy Institute on expected batch loading schedules for higher burnup fuel. These efforts are not expected to challenge the conclusion in the applicability of RG 1.183 or the high burnup source terms proposed in the SNL report SAND2011-0128 but, if they do, the FSCB staff will notify your staff. All of these efforts could serve as a future basis for an update to the source term tables in Regulatory Guide 1.183 at a later time.

Discussion

Burnups up to 68 GWd/MTU

In 2011, Sandia National Laboratories (SNL) issued SAND-2011-0128, “Accident Source Terms for Light Water Nuclear Power Plants Using High-Burnup of MOX Fuel” (ADAMS Accession Number ML20093F003). This report documents a series of MELCOR calculations to compare source terms for low burnup fuel (26-38 GWd/MTU core average discharge burnup, which varied depending on the plant analyzed) vs. high burnup (HBU) fuel in BWRs and PWRs (59

GWd/MTU maximum assembly-averaged burnup corresponding to 62 GWd/MTU peak rod-average burnup). The calculations accounted for cycle-specific information, fuel assembly design, core inventories, and decay heat. They also accounted for higher fission product diffusivity for the HBU fuel based on experimental results from VERCORS program in France.[1] The diffusion coefficient is based on VERCORS test RT-6, which used a UO_2 pellet irradiated to 72 GWd/MTU in a commercial PWR.[2]

The work performed and documented in SAND2011-0128 indicated little difference between the results of the low burnup and high burnup cases, even with the higher decay heat and fission product diffusivities in the high burnup case. Tables 10 and 11 in SAND2011-0128 provide comparison for BWRs and PWRs, respectively. Based on the limited impact of burnup effects between 38 GWd/MTU and 62 GWd/MTU, it is reasonable to extrapolate the conclusion for fuel with 68 GWd/MTU peak rod-average discharge burnup. Core inventories and decay heat at extended burnups (up to 80 GWd/MTU) are being assessed with the SCALE code through our existing ATF activities.

It is important to point out that the results from SNL's calculations for both low and high burnup fuel are somewhat different from the source terms in NUREG-1465 (and thus the timing of each fractional release group stated in Tables 1 and 2 of Regulatory Guide 1.183). There is a difference in the time-duration of the in-vessel release period that is significantly longer in SNL's proposed source term compared to NUREG-1465. Release group fractions are comparable for most chemical classes, but there are some differences, the most notable being a higher iodine release fraction for BWRs. See Tables 12 and 13 of SAND2011-0128 for details of the comparison to the results from NUREG-1465. It is unclear how these differences would impact the radiological consequence analyses performed by the NRR staff.

The SNL results were peer-reviewed by a panel led by Energy Research, Inc. The panel concluded that the "proposed source terms in [SAND2011-0128] are technically justified and appropriate" (per the Executive Summary of the peer review report), but it recommended SNL provide additional documentation of the methods used for the calculations and of the accident progression results. The panel also noted the significant uncertainties in the iodine release fraction results and recommended SNL provide further discussion of iodine chemistry. For further details, please see the peer review report available in ADAMS under Accession Number ML12005A043. At the time, the NRC and SNL did not address the peer review panel recommendations due to staff availability related to post-Fukushima work.

In summary, Sandia's results showed little difference between low burnup (LBU) and HBU fuel but did demonstrate noticeable differences due to using a more modern severe accident code

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[1] The VERCORS program studied release of fission products from irradiated UO_2 pellets in a furnace under simulated severe accident conditions. For more information on this program and its results, please refer to G. Ducros, et al., "Fission product release under severe accidental conditions: general presentation of the program and synthesis of VERCORS 1–6 results," *Nuclear Engineering and Design* 208.2 (2001): 191-203.

[2] See SAND2010-1633, "Synthesis of VERCORS and Phebus data in severe accident codes and applications" (ADAMS Accession Number ML20093F204) for further information.

instead of the older Source Term Code Package used for NUREG-1465. FSCB staff recommends using the source terms from Tables 10 and 11 of SAND2011-0128 for high burnup fuel up to 68 GWd/MTU peak rod-average discharge burnup. These source terms are more appropriate than Tables 1 and 2 of Regulatory Guide 1.183 because they represent significant improvements in severe accident and source term modeling since the development of NUREG-1465. As indicated in the peer review report, the proposed source terms from SAND2011-0128 are technically justifiable and meet the attributes of an acceptable source term as defined in Section C.2 of Regulatory Guide 1.183. FSCB staff believes the proposed source term still meets the criteria of an acceptable source term for burnup up to 68 GWd/MTU because the incremental increase in burnup is not expected to significantly impact the source term results. This is consistent with the work documented in SAND2011-0128 showing little difference in source term between the low and high burnup cases.

The staff has work planned in-house and at SNL to perform additional analysis for HBU fuel using the latest version of MELCOR because many improvements have been made to the code since the SAND2011-0128 study. In addition, improvements have been made in the MELCOR input models as part of the more recent State-of-the Art Reactor Consequence Analysis (SOARCA) uncertainty analysis and post-Fukushima regulatory analysis (e.g., NUREG-2206). These efforts to recalculate the accident source term are not expected to challenge the conclusion regarding the acceptability of the source terms from SAND2011-0128 Tables 10 and 11 but, if they do, RES/DSA staff will notify your staff. In parallel, FSCB staff will provide further justification for why the proposed source term meets Regulatory Position 2 from Regulatory Guide 1.183.

SNL's work documented in SAND2011-0128 covers the release fractions in RG 1.183 Tables 1 and 2 and release phases in Table 4 for LOCA events. RG 1.183 Table 3 provides recommended gap fractions to use for source term calculations for non-LOCA events. A footnote to Table 3 limits its applicability to fuel with a peak burnup of 62 GWd/MTU; this includes a restriction on the peak linear heat generation rate for fuel with burnup between 54 GWd/MTU and 62 GWd/MTU.

NRC staff proposed an update to RG 1.183 Table 3 in Draft Guide 1199, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." The updated tables were generated using a methodology that included the FRAPCON-3.3 fuel performance code and the ANS 5.4 (2011), "Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel," the standard for fission gas release. More recently, NRC staff performed updated calculations with the Fuel Analysis under Steady-state and Transients (FAST) fuel performance code to estimate the fission product inventories in the gap at the start of a non-LOCA event (ADAMS Accession Number ML19154A226).

The staff calculations to support an update to Table 3 did not consider fuel with burnup above 62 GWd/MTU. However, FAST shows good agreement with experimental fission gas release data up to a burnup of about 70 GWd/MTU. The FAST assessment is described in a Pacific Northwest National Laboratory (PNNL) report titled "FAST-1.0: Integral Assessment" available in

ADAMS under Accession Number ML20099A089. Further information on FAST assessment for HBU fuel is given in the PNNL report titled "Fuel Performance Considerations and Data Needs for Burnup above 62 GWd/MTU" (ADAMS Accession Number ML19317D098).

Given these assessment results, FAST could be used to determine fission gas release fractions for fuel up to this burnup with one caveat related to the cesium release fraction. The cesium release calculated by FAST is based on the assumption that the diffusion coefficient for cesium within the fuel grains is twice the diffusion coefficient of krypton. Otherwise, FAST treats cesium release the same as Kr-85 release. This approach is based on provisions in American Nuclear Society (ANS) standard ANS 5.4 (2011). When applied to high burnup fuel, FAST predicts cesium release fractions on the order of 50 percent as discussed in "Release Fractions in Non-LOCA Accidents in Draft Regulatory Guide 1.183 DG-1199" (ADAMS Accession Number ML19094A336).

However, cesium is expected to behave very differently than krypton once it reaches the grain boundaries. At this point, it can react with other constituents in the fuel to form less volatile compounds that may then accumulate on the grain boundaries as solids or liquids. Cesium released from the fuel may also react with the zirconium in the cladding to form more stable (i.e., non-gaseous) compounds. These effects tend to decrease the inventory of gaseous cesium available for release in the event of a cladding breach during a non-LOCA event. Indeed, experimental evidence shows that cesium releases are much lower than noble gas releases during normal operations and power ramps.[3] To further this, staff is considering performing experimental work to understand fission product (FP) distribution, quantity, and chemical form that would provide further measurement data. Thus, one would need to consider the effects of cesium chemistry to more accurately model its release to the gap.

In summary, using existing models in FAST to determine the cesium release fraction yields results that are excessively conservation for HBU fuel as demonstrated by the inconsistency between the proposed cesium release fractions in Table 3 of Draft Regulatory Guide 1199 and experimentally observed cesium behavior. Based on the limited data available, it seems likely that the cesium gap inventory would be much closer to 0.12 (i.e., the value in the current version of Regulatory Guide 1.183) than 0.48 (i.e., the proposed gap inventory in DG-1199). Therefore, FSCB staff recommends using the existing cesium gap inventory from Table 3 of Regulatory Guide 1.183.

Enrichment between 5-8 percent

Enrichment is not expected to significantly impact accident source term release fractions or release phase durations. However, enrichment above 5 percent will impact fission product inventories used in source term calculations. The staff is currently working with Oak Ridge National Laboratory (ORNL) in assessing the capabilities of the SCALE computer code to

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[3] For more information on the subject of cesium releases during normal operations and abnormal operational occurrences (AOOs), see C.T. Walker, C. Bagger, and M. Mogensen, "Observations on the release of cesium from UO_2 fuel," *Journal of Nuclear Materials* 240 (1996) 32-42.

predict inventories in this area, which would also be used to initialize fuel performance and severe accident analyses. Increased enrichment will necessitate increased use of burnable poisons and will possibly change the assembly geometry, which in turn will increase core heterogeneity and harden the neutron spectrum (due to more thermal absorption by U-235 and burnable absorbers). These changes will impact the fission product isotopic inventory present at the start of an accident. Licensees will need to account for these effects in their calculations of fission product inventories used for source term calculations.

Moreover, enrichment is not expected to significantly impact results from the FAST fuel performance code. Increased enrichment may have a minor impact on the fuel radial power profile, which in turn could impact the fuel temperature profile. The model used to calculate the radial power profile in FAST is applicable up to 12 percent enrichment, so FAST could be used to evaluate the impacts (if any) of increased enrichment on fuel-cladding gap fission product inventories.

Chromium-coated cladding

Staff internal discussions with Dana Powers, formally a senior scientist from SNL and member of the Advisory Committee on Reactor Safeguards, evaluated the potential impacts of adding a thin chromium coating to the zirconium alloy cladding on the accident source term. The staff's overall conclusion is that the chromium coating will have little impact on the source term for reasons described below.

Chromium-coated zirconium alloy cladding is expected to have some impact on accident sequence timing due to the reduction in high-temperature oxidation rate (and thus, heat generation) of the coating layer compared to bare zirconium alloys. However, the Zr-Cr system has a eutectic point around 1330˚C; above this temperature, the coating is expected to lose its protective quality, resulting in rapid oxidation of the underlying zirconium-alloy substrate. This has been demonstrated by transient testing up to 1500˚C at Karlsruhe Institute of Technology, albeit with non-prototypic coatings. See "Oxidation and Quench Behavior of Cold Spraying Cr-coated Zircaloy Fuel Cladding under Severe Accident Scenarios,"[4] which was presented by KIT researchers at the 2019 Top Fuel conference.

The Electric Power Research Institute attempted to quantify the effects of chromium-coated cladding on severe accident progression in PWRs and BWRs as described in the 2019 Technical Report, "Accident Tolerant Fuel Valuation: Safety and Economic Benefits."[5] This study used the industry-led Modular Accident Analysis Program (MAAP). The MAAP calculation results showed only limited delays (< 1 hour) in the onset of core damage and fission product release for the scenarios they analyzed. Therefore, it is reasonable to conclude that Cr-coated

[4] C. Tang, M. Große, M. Steinbrück, and K. Shirvan, "Oxidation and Quench Behavior of Cold Spraying Cr-Coated Zircaloy Fuel Cladding Under Severe Accident Scenarios," *TopFuel 2019*, Seattle, WA, USA, September 2019. This paper is publicly available through researchgate.net.
[5] This report is publicly available at https://www.epri.com/#/pages/product/3002015091/.

cladding has a minimal impact on the onset and duration of fission product releases such that the release timings from SAND2011-0128 are still applicable for coated cladding.

Note that the presence of chromium could have some impact on fission product speciation and release from a corium pool. However, the proposed coating layer thicknesses are less than 50 microns, which is a small fraction of the overall cladding thickness. Thus, the small increase in chromium that would be present in a molten corium pool is not expected to have a significant impact on the in-vessel release fractions given in Tables 10 and 11 of SAND2011-0128. The outcome of the severe accident PIRT is not expected to challenge the conclusion in the acceptability of the source terms from SAND2011-0128 Tables 10 and 11 for chromium-coated zirconium alloy cladding. If it does, RES/DSA staff will notify your staff.

With regards to inventories, the chromium-coated fuel pins are expected to have a minimal impact on the neutron spectrum and thus, inventory calculations. However, a need will exist to assess the gamma production from the neutron-induced reactions in chromium. This effect may have an impact on decay heat analyses. Currently, these phenomena are being assessed with the SCALE code through our existing ATF activities.

Chromia-doped fuel

Staff internal discussions with Dana Powers evaluated the potential impacts of adding chromia (Cr_2O_3) dopant to UO_2 fuel on the accident source term. The staff's overall conclusion is the chromia dopant will have little impact on the source term for reasons described below.

Adding chromia to UO_2 fuel increases the fuel grain size, which increases the fission product diffusion path length. On the other hand, the dopant increases the vacancy concentration in the fuel lattice, which increases the fission product diffusion coefficient and somewhat counteracts the fission product retention benefits of increased grain size. The net effect is a modest reduction in fission gas release during normal operations and power ramps as demonstrated by limited test data from the Halden Reactor Project.[6] The existing fission gas release models in FAST yield conservative results compared to the available data as described in a paper from the Top Fuel 2018 conference, "Expanded Assessment of FAST for Power Ramp Cases with Short Hold Times and Advanced UO_2 Fuel with Various Dopants."

With that said, doped fuel is expected to have little impact on the accident source term. As demonstrated by SNL's high burnup calculations (described above), modest changes to the fission product diffusion rate have little impact on releases from the fuel during a severe accident. Moreover, the small concentration of dopants is expected to have little impact on fission product speciation. Finally, adding dopants has little impact on fuel thermal properties and is not expected to impact accident progression. Thus, it is reasonable to use the source

[6] For discussion of the results of Halden Reactor Project test IFA-677.1 involving chromia-doped fuel, see D.J. Richmond and K.J. Geelhood, "Expanded assessment of FAST for power ramp cases with short hold times and advanced UO_2 fuel with various dopants," *TopFuel 2018*, Prague, Czech Republic, 2018. This paper is publicly available at https://www.euronuclear.org/archiv/topfuel2018/proceedings.htm.

term in SAND2011-0128 for doped fuel. The outcome of the severe accident PIRT is not expected to challenge the conclusion in the acceptability of the source terms from SAND2011-0128 Tables 10 and 11 for chromia-doped UO_2 fuel. If it does, RES/DSA staff will notify your staff.

With regards to inventories, as with the chromia-coated clad, an integral chromia dopant is not expected to have significant impact on the neutron spectrum. Currently, these phenomena are being assessed with the SCALE code through our existing ATF activities.

FeCrAl cladding

MELCOR model development for accident-tolerant fuels has focused primarily on development of models for FeCrAl cladding. A new default core material, FeCrAl, has been introduced into MELCOR, allowing users to perform accident tolerant fuel analyses. The material can be initialized as cladding and will undergo melt relocation, refreezing, and oxidation. A comparison of the default parabolic rate constant for FeCrAl oxidation was made to the other default materials. The reduced reaction prior to breakaway oxidation shows potential advantage in reaction rates in addition to reducing exothermic energy generation. This new modeling approach is currently present in the official MELCOR code released in November 2019. More recently, the FeCrAl modeling was modified to take advantage of a new capability for modeling material composition and oxidation in a more generic way. The generalized oxidation model is now used to specify the oxidation behavior for the FeCrAl cladding.

Areas of active development are concerned with possible eutectics associated with this new material and assessment of the models with the QUENCH-19 experiment conducted at Karlsruhe Institute of Technology (KIT) in Germany. Sandia National Laboratories is also exploring the integration of a chemistry Gibbs free energy minimizer code (Thermochemica®) into MELCOR for future capability to assist in the assessment of eutectic modeling. In addition, modifications to the molten core chemistry models are being considered to include these new materials. Moreover, additional data may be required to assess the clad failure behavior. For source term characterization, additional calculations and analysis will be required to synthesize the duration of the gap release and in-vessel release.

With regards to inventory analysis, the use of FeCrAl is expected to have more significant neutronic impacts relative to the other ATF designs mentioned earlier. These phenomena are being assessed with the SCALE code through our existing ATF activities.

This response was discussed with Paul Clifford, Mark Blumberg, and Elijah Dickson of your staff.

SUBJECT: APPLICABILITY OF SOURCE TERM FOR ACCIDENT TOLERANT FUEL, HIGH BURN UP AND EXTENDED ENRICHMENT DATED MAY 13, 2020

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