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Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants

Comment On: NRC-2021-0179-0001
Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors

Document: NRC-2021-0179-DRAFT-0010
Comment on FR Doc # 2022-08519

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General Comment

See attached file(s)

Attachments

OG-22-117

Program Management Office
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Project 694
Docket 99902037

June 20, 2022

OG-22-117

Office of Administration
Mail Stop: TWFN-7-A60M
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
ATTN: Program Management, Announcements and Editing Staff

Subject: PWR Owners Group, Analysis Committee
PWROG Comments on Draft Regulatory Guide (DG), DG-1389, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," 87 FR 23891; Docket ID NRC-2021-0179

Dear Mr. Michael Eduy and Mr. Mark Blumberg,

The PWR Owners Group has developed on behalf of its members the attached comments on Draft regulatory guide (DG), DG-1389, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors". This letter serves to transmit the PWR Owners Group comments to the NRC. The PWROG appreciates the opportunity to comment on DG-1389.

Attached to this letter please find:

- PWROG Comments on DG-1389 (Non-Proprietary) (Attachment 1)

If you have any questions, please do not hesitate to contact me at (602) 999-2080 or Mr. T. Laubham, Executive Director (Acting) of the PWR Owners Group, Program Management Office at (412) 374-6788.

Sincerely yours,

Michael Powell
Chairman and COO
PWR Owners Group

DRPB

Attachment: PWROG Comments on DG-1389 (Non-Proprietary)

cc: PWROG PMO
PWROG Analysis Committee
PWROG Licensing Committee
B. Mount, PWROG/Dominion Energy
N. Hall, PWROG/STP
M. Powell, PWROG/APS
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L. Fields, US NRC

PWROG DG-1389 Comment 1

Statement in Guidance: "Revision 0 of RG 1.183 will continue to be available for use by licensees and applicants as a method acceptable to the NRC staff for demonstrating compliance with the regulations. A combination of the methods contained in revision 0 or revision 1 of RG 1.183 would need additional justification." (Page 6)
Description of concern: It is not clear if licensees can transition one (or a few) analyses to revision 1, or if full implementation for all design basis accidents is required. In revision 0 of RG 1.183, partial implementation was allowed.
Basis For Concern: Full implementation of DG-1389 could be cost prohibitive. Additionally, vague guidance could result in submittals that do not meet the intent of the Reg Guide.
Proposal: "Revision 0 of RG 1.183 will continue to be available for use by licensees and applicants as a method acceptable to the NRC staff for demonstrating compliance with the regulations. *Selective implementation is acceptable, provided that each accident analysis uses either revision 0 or revision 1*. A combination of the methods contained in revision 0 or revision 1 of RG 1.183 *in a single analysis* would need additional justification."
Basis for Proposal: It is assumed that the NRC intends to allow partial implementation of DG-1389 but did not intend the guidance in the DG to be mixed in a single analysis.

PWROG DG-1389 Comment 2

Statement in Guidance: "For the remaining non-LOCA DBAs which predict fuel rod cladding failure, such as PWR reactor coolant pump locked rotor and fuel handling accident, additional fission product releases may occur as a result of fuel pellet fragmentation (e.g., fracturing of high-burnup rim region) due to loss of pellet-to cladding mechanical constraint or impact loads. TFGR has been experimentally observed under a variety of accident conditions and should be addressed in future applications." (Page 19)
Description of concern: The statement implies that so called "transient" fission gases releases (TFGR) due to high burnup pellet fragmentation must be considered for non-LOCA DBAs that predict fuel failure but does not provide details how this requirement could be met. Additional fragmentation-based accident transient fission gas releases are tied to certain mechanisms, such as increases in fuel temperature, that occur in specific event scenarios. For Non-LOCA, TFGR guidance is currently only available for reactivity insertion accidents (RG 1.236).
Basis For Concern: There is no guidance available to address non-LOCA accidents aside from reactivity insertion accidents and little experimental data to develop a model or support the concern. The mechanism for transient fission gas release and test data supports a fuel temperature dependent relationship. The first phase of release is in the range of 500-800°C (e.g., [1], [2]). 1. Magnusson, P., et al., 2016. "A study of transient FGR by integral LOCA tests," Top Fuel 2016. 1.2. 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.
Proposal: Appropriate additional high burnup-related TFGR guidance for non-LOCA DBAs should be provided. Non-LOCA DBA without fuel overheating are not impacted. For non-LOCA DBA with fuel overheating, guidance should be specific. Identify the specific non-LOCA DBAs that are impacted and include applicable TFGR guidance in the relevant Appendices.
Basis for Proposal: Existing data indicates significant transient gases require the accidental conditions associated with RIA or the higher temperature thresholds associated with LOCA-type events. The Draft Guide should not extrapolate TFGR to other non-LOCA accidents without adequate references.

PWROG DG-1389 Comment 3

Statement in Guidance: "The licensee may take credit for accident mitigation features that are classified as safety related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures." (Page 29)
Description of concern: Akin to the limitation allowance for operator action in the current draft text, the guidance should not exclude all potential use of non-safety related equipment. Consistent with the historic practice in safety analysis, justifiable credit for non-safety grade equipment has been acceptable.
Basis For Concern: The DG should provide additional guidance for crediting non-safety grade equipment during an accident. There is currently no guidance for providing acceptable justifications for crediting non-safety grade equipment to mitigate an accident within the draft guide.
Proposal: Add text to the RG discussing the elements necessary to justify crediting the use of non-safety equipment. Examples of crediting the use of non-safety equipment include the crediting the Hatch 1 & 2 turbine building ventilation system described in ML072910399 and approved via ML081770075 and the Oconeee QA Condition 5 program described in Attachments 4, 4a and 4b of ML15238A066.pdf and accepted in ML16141A933.
Basis for Proposal: The NRC should provide additional guidance to credit non-safety related equipment following an accident to ensure that these justifications are addressed consistently across the LWR fleet.

PWROG DG-1389 Comment 4

Statement in Guidance:
"For the purpose of this RG, a transit dose is considered to be the dose that is accumulated as personnel travel to and from the control room for the duration of an accident once onsite (e.g., dose from site boundary to the control room). Licensees whose licensing basis includes transit dose should include the transit dose to demonstrate compliance with 10 CFR 50.67(b)(2)(iii)." (Page 24)

Description of concern:
Control rooms are designed to maintain habitable and safe conditions under accident conditions, including loss-of-coolant accidents. Control room dose calculations are performed to demonstrate the adequacy of the control room design with respect to radiological protection. Control rooms are not designed to maintain habitable and safe conditions for those external to the boundary of the control room. .

Basis For Concern:
10 CFR 50.67(b)(2)(iii) states "Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident". This echoes the language of 10 CFR 50, Appendix A, General Design Criterion 19. With respect to radiological consequences, 10 CFR 50, Appendix A, General Design Criterion 19 states "Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident."

In accordance with General Design Criteria 19, control rooms are designed to protect individuals occupying an accessible control room. With respect to radiological consequences, a key focus of the design is on HVAC considerations, including a safety-related emergency mode with HEPA filters and charcoal beds, with provisions made for limiting unfiltered inleakage. The physical structure, i.e. walls, ceiling, and floor, of the control room is designed to provide shielding to occupants from a post-accident radiological release. Post-accident access to the control room for the operators is not limited by the control room design or control room habitability system design, except to the extent where the control room must remain accessible. i.e. the operators can enter and exit the control room freely. The existing analysis assumption of an operator remaining in the control room for 30 days (with allowances for occupancy factors and reasonable in-leakage which covers ingress and egress through the control room envelope boundary) is intended to be an overall conservative approach with respect to the expected rotation of personnel. Demonstrating doses less than the design limit confirms the adequacy of the design of the control room (along with the other engineered safety features of the plant design). This is the dose reported in Chapter 15.

Doses accrued by operators when not in the control room are not explicitly accounted for in the design of the control room as the occupancy factor assumptions are intended to provide an overall conservative approach.

In addition, 10CFR50.47(a)(9), (10, and (11) provide sufficient requirements for the Emergency Response organizations to measure and limit doses to emergency response personnel including control room operators.

It is understood that a small subset of licensees include transit dose in their respective Chapter 15 control room doses. This is not a widespread practice. As such, there is no definitive guidance for calculating transit doses.
Proposal: Please remove the discussion of transit dose.
Basis for Proposal: Control room doses should be limited to doses accrued while occupying the control room provided.

PWROG DG-1389 Comment 5

Statement in Guidance: N/A
Description of concern: The holistic impact of transitioning from Revision 0 to Revision 1 is not clear. Evaluations of a sample plant(s) would illustrate the impacts of transitioning from Revision 0 to Revision 1 on plant analyses and plant maintenance.
Basis For Concern: DG-1389 represents a significant change in methodology compared to RG 1.183. The difference is comparable to the transition from TID to AST circa 1999. Major changes have been made to the release fraction, timing, and accident definition. It is not clear how the proposed changes will impact existing PWR dose analyses or plant maintenance requirements based on those analyses. For the transition from AST to TID, sample plant evaluations were developed to illustrate the impact. For example, the MHA release duration in DG-1389 (4.72 hours) is extended compared to RG 1.183 Revision 0 LOCA (1.8 hours). This increase in duration could result in additional duration-of-operation requirements on containment sprays.
Proposal: Please provide evaluation(s) of sample plant(s) to illustrate the impacts of transitioning from Revision 0 to Revision 1.
Basis for Proposal: The impact of transitioning from Revision 0 to Revision 1 should be illustrated by example, similar to the transition from TID to AST (SECY-98-154).

PWROG DG-1389 Comment 6

Statement in Guidance: Multiple locations in Table 7 and Appendices.
Description of concern: A pre-accident (also called pre-incident) iodine spike occurs when a plant enters a time-limited action statement related to the limiting condition for operation (LCO) for specific RCS activity. The abnormal operation condition occurs independent of and prior to the design basis event (e.g. MSLB or SGTR). Thus a MSLB or SGTR with a pre-accident iodine spike constitutes two independent overlapping design basis conditions, the combination of which is typically considered beyond design basis. This is unique to the dose analyses. While EAB and LPZ dose limits are higher for this scenario, the Control Room dose limit is unchanged.
Basis For Concern: A pre-accident iodine spike occurs when a plant enters a time-limited action statement related to the limiting condition for operation (LCO) for specific RCS activity. The inclusion of this abnormal operation condition in the licensing basis analyses is unique. Typically, abnormal operation conditions are not included in the licensing basis analyses. For example, for plants that normally require all atmospheric dump valves (ADVs) to be operable, when one ADV is declared inoperable, there is a time-limited action statement to restore that ADV to operable status. However, it is not required to include cases in the licensing basis analysis with one ADV out of service. A pre-accident iodine spike occurs independent of and prior to the design basis event (e.g. MSLB or SGTR). Thus a MSLB or SGTR with a pre-accident iodine spike constitutes two independent overlapping design basis conditions, the combination of which is typically considered beyond design basis. For example, the probability of a Steam Generator Tube Rupture occurring during the LCO period of a DEI excursion is beyond the scope of a plant's licensing basis. Based on the industry average parameter estimates in NUREG/CR-6928, the mean frequency of a SGTR is 1.78E-3/year. This frequency was developed by Idaho National Laboratory and is based on data as recent as 2020. This value is used in the NRC SPAR models and industry PRA models. NRC: Industry Average Parameter Estimates (inl.gov) The likelihood of this event occurring coincident with the plant being in a DEI Limiting Condition is extraordinarily low. The Standard Technical Specification 3.4.16 in NUREGs-1430, -1431, and -1432 reports this LCO period is only 48 hours or 0.55% of the year for all PWR types. Fuel performance indicators would typically direct a plant to shut down well before it even enters this LCO. Considering that the SGTR accident is independent of the chemistry excursion that leads to the DEI exceeding the maximum steady-state value (typically 1.0 μCi/g), the probability of a SGTR occurring during this LCO period is then only 9.75E-6. Including a subsequent LOOP (mean frequency on the order of E-02) in this scenario leads to an even lower frequency (on the order of E-07).

This accident frequency is well below the Design Basis Accident – Limiting Fault range of 1E-4/year to 1E-6 described in Annex II of IAEA Publication SSG-2, Rev .1 "Deterministic Safety Analysis for Nuclear Plants" 2019. As such, the SGTR case assuming a pre-existing spike should <u>not</u> be included in the plant design bases. This also applies to Main Steamline Break. It is recommended that the pre-accident iodine spike be removed as a requirement for licensing basis dose analyses.
Proposal: Eliminate the pre-accident iodine spike as a requirement for licensing basis analyses.
Basis for Proposal: The inclusion of an abnormal operation condition in the licensing basis analyses is unique to the pre-accident iodine spike scenarios. These unique requirements should be either removed or justified.

PWROG DG-1389 Comment 7

Statement in Guidance: "The nuclides used for modeling dose from airborne radioactivity inside the control room may not be conservative for determining the dose from radioactivity outside the control room." (Page 26, Footnote 15)
Description of concern: This is a new footnote that was not included in RG 1.183 Rev. 0. The intent of this new footnote is unclear.
Basis For Concern: There are multiple interpretations for this new footnote. Within context, it is understood to mean that the same set of nuclides should be used in the dose calculations both internal and external to the control room, with recognition that the relative dose significance of a given nuclide in each calculation may vary. Clarification by the NRC will ensure proper application by licensees.
Proposal: Please clarify the footnote. Suggested text is provided in italics: "15. The nuclides used for modeling dose from airborne radioactivity inside the control room may not be conservative for determining the dose from radioactivity outside the control room. *Recognizing that the relative significance of a given nuclide may vary, the same set of nuclides should be used in the dose calculations both internal and external to the control room.*
Basis for Proposal: Guidance should clearly communicate expectations.

PWROG DG-1389 Comment 8

Statement in Guidance:
"3.3 Timing of Release Phases

Table 5 provides the onset and duration of each sequential release phase for LOCA DBAs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel release phase immediately follows the gap release phase. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase.[11] For non-LOCA DBAs in which fuel damage is projected, the release from the fuel gap and the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage.

The applicability of Table 5 is consistent with the applicability of Tables 1 and 2.

Table 5. MHA LOCA Release Phases

Phase	PWRs		BWRs	
	Onset	Duration	Onset	Duration
Gap Release	0.5 minutes	0.22 hours	2 minutes	0.16 hours
Early In-Vessel	0.22 hours	4.5 hours	0.16 hours	8.0 hours

The early in-vessel release phase begins immediately following the gap release phase. For facilities licensed with leak-before-break methodology, the licensee may assume the onset of the gap release phase to be 10 minutes. A licensee may propose an alternative time for the onset of the gap release phase based on facility-specific calculations using suitable analysis codes or on an accepted topical report shown to be applicable to the specific facility. In the absence of approved alternatives, the licensee should use the gap release phase onsets in Table 5. Regardless of delays in the onset, the duration of the gap release phase is 0.5 hours."

Description of concern:
It is not clear in the wording of Table 5 and surrounding text whether licensees should treat the times listed at the end of each phase as "duration" or "end times". If they are treated as "duration" as described, there would be overlap between phases. However, the text states the early in-vessel release phase begins immediately following the gap release phase.

Basis For Concern:
The DG should be clear in the intent of the modeling of release phases, so all licensees model the accident consistently and correctly.

Proposal:
Please revise the word "duration" in Table 5 and surrounding text to state "end time" or similar wording, to clarify there is no overlap of the phases, and the final times listed are the end of any releases (4.5 hours or 8.0 hours).

Basis for Proposal:
Guidance should clearly communicate expectations.

PWROG DG-1389 Comment 9

Statement in Guidance: Appendix E, Position E-6.1 states: "The primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the TSs. The leakage should be apportioned between affected and unaffected steam generators in a manner that maximizes the calculated dose."
Description of concern: It is overly conservative to model the maximum Technical Specification leak rate for the duration of the event, especially in the later stages of the accident where the pressure differential across the steam generator tubes may be significantly reduced. This results in unrealistic activity release to the secondary system and the environment in events that model these releases (MSLB, SGTR, locked rotor, rod ejection).
Basis For Concern: In later stages of the event(s), the pressure differential across the steam generator tubes is significantly reduced from event initiation. For instance, the TS leak rate is intended to bound the pressure differential during instances of a MSLB when the faulted steam generator blows down to atmospheric (or near atmospheric) conditions while the primary system remains at higher pressures (NOP down to ~1000 psia). Later in the transient, residual heat removal cut-in conditions are reached, with the primary system pressure reduced to ~375 psig. MSLB conditions (and other secondary release events) are not anticipated to change the tube defect level in the steam generators; therefore, the maximum primary-to-secondary leakage should decrease significantly from initial conditions.
Proposal: The NRC should add the following statement to Position E-6.1 (and related statements in other appendices) to allow for a licensee to credibly justify lower leakage rates during later stages of the events. "Licensees may provide adequate technical justification to consider lower primary-to-secondary leakage rates at later stages of the transient when pressure differentials across the steam generator tubes are reduced from those representing maximum leakages." This statement should also be added to the other appendices modeling primary to secondary leakage (Appendix F for MSLB, Appendix G for Locked Rotor, and Appendix H for Rod Ejection).
Basis for Proposal: It is overly conservative to model the maximum Technical Specification leak rate for the duration of the event, especially in the later stages of the accident where the pressure differential across the steam generator tubes may be significantly reduced.

PWROG DG-1389 Comment 10

Statement in Guidance: Table 7, PWR Steam Generator Tube Rupture category "Fuel Damage or Preincident Spike" implies that a fuel damage case would take the place of the Pre-incident spike. This is not consistent with guidance in E-2.2 (page E-3), "A concurrent iodine spike need not be considered if fuel damage is postulated." There is a similar issue with F-2.2.
Description of concern: The guidance in E-2.2 and F-2.2 appear to be inconsistent with Table 7.
Basis For Concern: Guidance should be self-consistent throughout. Additionally comment 6 requests that the pre-accident spike be removed.
Proposal: Remove Pre-Accident spike from Table 7 and delete E-2.1 and F-2.1.
Basis for Proposal: This preserves intent of E-2.2 and F-2.2 and is consistent with the rest of the comment package.

PWROG DG-1389 Comment 11

Statement in Guidance: ***4.2.7 Dose Conversion Factor*** The licensee should calculate control room doses using the dose conversion factors identified in Regulatory Position 4.1 for use in offsite dose analyses. The calculation should consider all radionuclides, including progeny from the decay of parent radionuclides that have significant dose consequences, and the released radioactivity. The EDE from photons may be corrected for the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression may be used to correct the semi-infinite cloud dose, $EDEX_{\infty}$, to a finite cloud dose, $EDEX_{finite}$, where the control room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the control room (Ref. 34). Equation 1: $EDEX_{finite} = (EDEX_{\infty} * V^{0.338})/1173$
Description of concern: This equation, from the Murphy-Campe paper, is for Noble Gases. Use of AST will have many more isotopes in the control room than the Noble Gases (with the associated photon differences). Other equations could apply to other isotopes in the control room.
Basis For Concern: The requirement to apply the equation provided in the RG is overly restrictive and may prevent a licensee from pursuing other justifiable conversion factors.
Proposal: The RG should include an allowance for a licensee to proposal alternate conversion factors with appropriate justification, "Licensees may choose to create their own finite volume correction factor based upon the isotopic mix (versus time) developed in their dose analysis. The Licensee needs to provide adequate justification supporting the alternate conversion factors."
Basis for Proposal: To assure that appropriate credit for shape factor is considered/used while calculating control room dose.

PWROG DG-1389 Comment 12

Statement in Guidance: F-6.4, and similar guidance in E-6.3 The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100 degrees Celsius (212 degrees Fahrenheit). The release of radioactivity from unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.
Description of concern: In PWRs, if forced circulation is not available (i.e. loss of off-site power), once the shutdown cooling system is placed into service, flow in the faulted loop may stall with the steam generator isolated. This means that although the active portions of the RCS will cool down via the shutdown cooling system and by dumping steam from the active SGs, the local fluid conditions in the faulted loop outside of the RHR cooling circuit could remain above 212F for an extended period of time.
Basis For Concern: There may be inconsistent interpretations of how to determine when the radiological release from the faulted steam generator has terminated for the purpose of radiological analysis.
Proposal: Update item F-6.4 to state that the release of radioactivity from faulted and unaffected steam generators may be analytically assumed to terminate when the bulk of the primary system is less than 212°F.   Recommended F-6.4 text is below, with the purposed change in italics. "The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature *in the bulk of the primary system* is less than 100 degrees Celsius (212 degrees Fahrenheit). The release of radioactivity from unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated." Note that similar guidance in E-6.3 for the unaffected SGs should be updated similarly.
Basis for Proposal: Once shutdown cooling is placed in service, RCS pressures and temperatures are significantly reduced. In addition, the local masses of fluid that would not be in communication with the active portions of the RCS are finite. The radiological release rate and consequence during this condition is very low. Explicit accounting for additional radiological release during this condition would add analysis complexity and burden without a significant change to the overall radiological consequence result.