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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

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General Comment

Please see attached letter BWROG22-1-403r0, Denver Atwood (BWR Owner's Group) to Office of Administration (NRC), "Comments on Draft Regulatory Guide DG-1389, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," June 17, 2022.

Attachments

001 BWROG22-1-403r0 BWROG Comments on DG-1389 2022-06

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BWROG22-1-403r0
June 17, 2022

To: 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: Comments on Draft Regulatory Guide DG-1389

The BWR Owners' Group (BWROG) is providing this letter for the Staff's consideration with comments on draft regulatory guide DG-1389, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" as requested in Federal Register 87 FR 23891.

This draft regulatory guide is important to BWRs since it extends the exposure range beyond the 62 GWd/MTU limit in Revision 0 of Reg Guide 1.183. The comments in Attachment 1 were developed by a committee of industry professionals with significant expertise in BWR radiological analyses.

Any questions or requests for clarification on these comments can be directed to the undersigned or Greg Broadbent (Entergy), the BWROG Reg. Guide 1.183 Committee Chairman, at gbroadb@entergy.com or (601) 497-2683.

Sincerely,

Denver Atwood
Chairman
BWR Owners' Group

cc: L. Martins, BWROG Senior Services Director
BWROG Primary Representatives

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Comment 1

Statement in Guidance:
Table 1 BWR Core Inventory Fraction Released into Containment Atmosphere

Description of Concern:
A comparison of the changes in maximum hypothetical accident (MHA) release fractions between BWRs (Table 1) and PWRs (Table 2) identified a significant increase in the BWR halogen release fractions with no indication in either SAND2011-0128 or DG-1389 as to the cause. This increase will adversely affect the ability of BWRs to comply with Reg Guide 1.183 Rev. 1. A comparison to the PWR analyses suggests that the accident sequences may be responsible for this impact.

The original alternative source term in NUREG-1465 was based on a range of accident sequences. As described in Section 3.1 of NUREG-1465, these sequences were based on the accident sequence data in NUREG-1150, which assessed the core damage risks for five representative plants using the PRA methods and severe accident modeling tools available in the late 1980s. The dominant sequences were evaluated with the Staff's Source Term Code Package (STCP) and the MELCOR code and are listed in Table 3.1 of NUREG-1465. An updated version of the core damage risk to the entire fleet is NUREG-1560, which was based on industry-prepared Individual Plant Evaluations (IPEs). As shown in Figure 3.6 of NUREG-1560 (reproduced below) for the majority of operating BWR plants (BWR/3 and BWR/4), there continues to be a broad diversity in the accident scenarios leading to core damage.

SAND2011-0128 updates NUREG-1465 with higher core exposures utilizing the latest NRC's MELCOR methodology. The accident sequences that were analyzed to develop the PWR release fractions are listed in Table 5 of the Sandia report and include a variety of accident types. However, for the BWR release fractions, Table 3 of the Sandia report indicates that nearly all of the evaluations were based on station blackout (SBO) sequences. As illustrated in Figure 3.6 of NUREG-1560, the core damage risk from SBO is similar to that from other accident sequences. Considering the recent modifications to address Fukushima SBO risk, the core damage risk from SBO is considerably lower than the NUREG-1560 data taken from the late 1990s.

Section A-2.1 of DG-1389 characterizes these release fractions as airborne releases into the BWR drywell. It is also unclear from the Sandia report whether suppression pool scrubbing was credited in determining the release fractions from a BWR SBO. Credit for pool scrubbing can significantly decrease the airborne activity since the SBO-related releases would be released via spargers underwater in the suppression pool.

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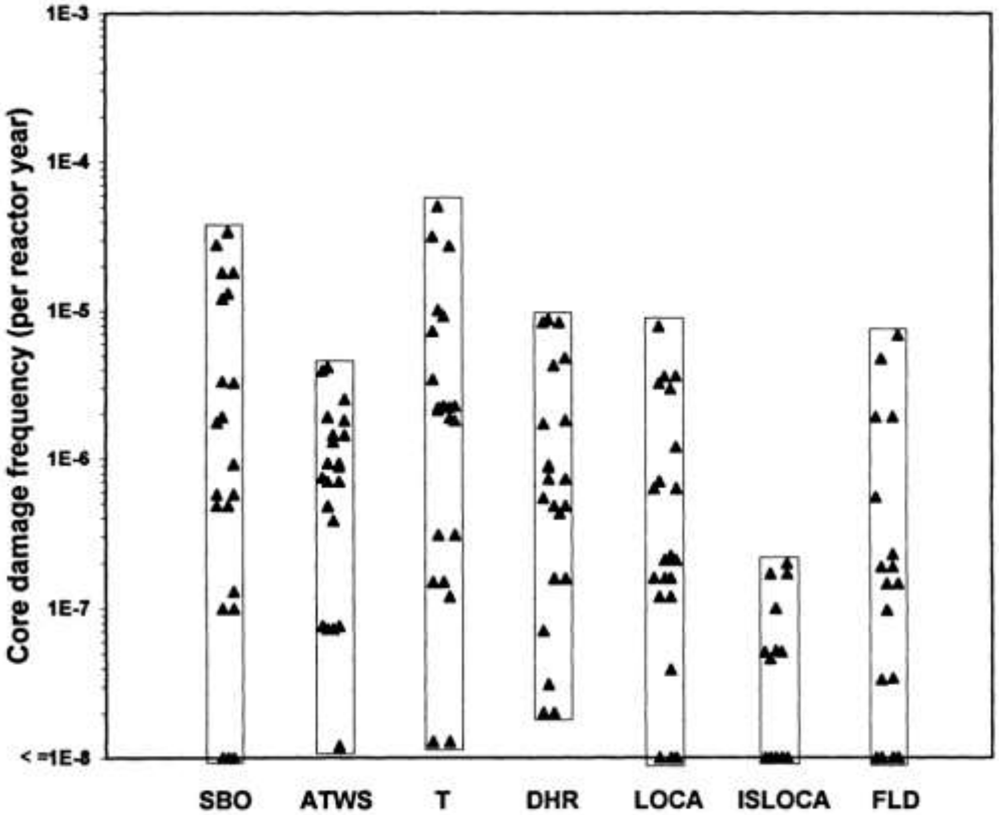

Figure 3.6 Reported IPE accident sequence CDFs for BWR 3/4 plants with RCIC.

NUREG-1560 3-18

Proposal:

Consistent with the PWR analysis in SAND2011-0128, the BWR release fractions should be re-evaluated based on accident sequences that are a more representative of BWR core damage risks. Any sequences that involve releases through the pool spargers should take credit for suppression pool scrubbing.

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Comment 2

Statement in Guidance: Table 3. BWR Steady-State Fission Product Release Fractions Residing in the Fuel Rod Plenum and Gap
Description of Concern: The Clifford memo entitled "TECHNICAL BASIS FOR DRAFT RG 1.183 REVISION 1 (2021) NON-LOCA FISSION PRODUCT RELEASE FRACTIONS" (ML21209A524) states that the BWR gap fractions are based on a BWR 10x10 fuel design; however, the design specifics or model name is not provided. Since there are at least 6 different BWR 10x10 designs, clarification is needed to ensure which designs are addressed by the listed DG-1389 non-LOCA release fractions. The listed release fractions may also be applicable to more advanced fuel designs such as 11x11 designs.
Proposal: State the BWR 10x10 fuel design applied to develop the BWR release fractions in Table 3. Indicate whether the results are applicable to BWR 11x11 designs that meet the power history inputs in Figure 1.

Comment 3

Statement in Guidance: Figure 1. Maximum Allowable Power Operating Envelope for Steady-State Release Fractions
Description of Concern: The BWR part-length rods tend to experience somewhat more aggressive power profiles than the full-length rods. Did the NRC analysis that developed the release fractions Table 3 apply the reported BWR power profile to the PLRs or was a more aggressive history assumed?
Proposal: Confirm applicability of Figure 1 power history to BWR PLRs or provide applied power history.

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Comment 4

Statement in Guidance:
Section 3.2
If BWR part-length fuel rods are treated as full-length fuel rods with respect to overall quantity of fission products, then Table 3 steady-state fission product release fractions apply to these part-length fuel rod designs.

Description of Concern:
BWR fuel designs are increasingly utilizing part-length fuel rods (PLRs) to optimize fuel exposure and shutdown margin. As the number of PLRs increases, the DG-1389 approach of treating PLRs as full-length rods tends to significantly over-estimate the bundle source term.

The DG-1389 approach to the PLR release fraction is just one approach to considering the PLR gap release. Another approach is to apply the guidance in DG-1389 Appendix J to calculate the actual release fraction in the PLRs.

Proposal:
Acknowledge that the DG-1389 approach to BWR part-length rods being treated as full-length rods with respect to overall quantity of fission products is one approach to the PLR issue and that other approaches are acceptable.

Comment 5

Statement in Guidance:
Section A-2.7
N/A

Description of Concern:
Section 3.7 of Appendix A to Reg Guide 1.183 Rev. 0 contained a paragraph providing specific guidance regarding mixing in Mark III containments, including uniform mixing between the drywell and containment after 2 hours. All BWR/6 Mark-III units currently apply this guidance. This paragraph has been deleted in DG-1389.

In the absence of any guidance, what is an acceptable approach to mixing in Mark-III containments? Are the previous Mark-III mixing assumptions still acceptable for amendments prepared under the proposed revision? Is the 2-hour timing for uniform mixing from Rev. 0 still applicable considering the new release timing in Table 5?

As reported in Item 2 in RIS 2006-04, are mixing approached based on thermal-hydraulic conditions still acceptable? If so, what would be the appropriate accident to apply for the MHA analysis?

Should mixing in the drywell be considered substantial enough in the MHA such that the reactor vessel volume can be included in the drywell volume?

Proposal:
NRC should confirm the continued applicability of the previous guidance or provide alternate guidance to BWR mixing between the drywell and wetwell/containment.

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Comment 6

Statement in Guidance: Section *A-5.6.1* *SAND 2008-6601 Model: Section 6.4 of Reference A-11 describes an acceptable model for estimating the aerosol deposition between closed MSIVs and downstream of the MSIVs.*
Description of Concern: This section and the removal coefficients in Table A-1 do not permit credit for the inboard MSL piping. The basis for this lack of credit seems to originate from Section 6.3 of SAND2008-6601 which states that “at times in the simulation the temperature of portions of the in-board MSL piping are predicted to be high enough to vaporize fission products that had been previously deposited.” However, many plants already credit this volume and RIS 2006-04 specifically permits this credit. Section 2 of RIS 2006-04 states “modeling of MSL piping may include volumes <u>between the reactor pressure vessel and the inboard MSIV (inboard volume),</u> between the inboard and outboard valves (in-between volume), and outside of the outboard valve (outboard volume).” These inboard steam lines represent significant lengths of safety-related horizontal surfaces that are amenable for source term deposition. It is recognized that these volumes would need to be subtracted from the overall drywell volume in performing the dose analysis. DG-1389 should allow licensees to take credit for this piping if analyses show temperatures that will not vaporize the deposited source term. Alternatively, re-vaporization of the deposited source terms can be modeled.
Proposal: Add statement allowing licensees to take credit for the inboard MSL piping with appropriate supporting analysis.

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Comment 7

Statement in Guidance: *Section A-5* *Three methods are presented below to compute aerosol deposition within main steamlines. Each method computes similar removal coefficients suitable for radiological consequences calculations, however, these methods are not valid if credit has been taken for aerosol removal from drywell sprays.*
Description of Concern: Multiple BWRs currently have credit for aerosol removal from drywell sprays as well as aerosol deposition within in the main steam lines in their current licensing basis. Section A-5 presents three acceptable methods for calculating aerosol deposition within the main steam lines, but caveats the methodology with, "…however, these methods are not valid if credit has been taken for aerosol removal from drywell sprays." Given the prevalence of credit for both sprays and steam line deposition, there should be a model presented that the Staff finds acceptable for crediting both, or modifications to the presented models if the licensee wants to credit spray removal (e.g., different aerosol size distribution)?
Proposal: Provide additional guidance on MSL deposition for BWRs that credit drywell sprays.

Comment 8

Statement in Guidance: *Table A-1. BWR Main Steamline and Condenser Removal Coefficients*
Description of Concern: The time periods in DG-1389 Table A-1 do not correspond to those in Table 6-1 of SAND 2008-6601. Specifically, SAND2008-6601 has a 0-2 hour time interval where higher removal coefficients can be applied. Then, the next time step is 2-12 hours with a third time period of 12+ hours. DG-1389 only contains a 0-10 hour period and one for 10+ hours.
Proposal: Correct Table A-1 with the time periods and removal coefficients from the underlying reference.

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Comment 9

Statement in Guidance: Section D-2.1 *The concentration that is the maximum value (typically 4.0 microcuries per gram (µCi/g) dose equivalent (DE) iodine (I)-131 (DE I-131)) permitted and corresponds to the conditions of an assumed pre-accident spike.*
Description of 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, which require a time-limited action statement to restore the system to operable status, are not included in the licensing basis analyses. A pre-accident iodine spike occurs independent of and prior to the main steam line break (MSLB). Thus, the MSLB with a pre-accident iodine spike constitutes two independent overlapping design basis conditions, the combination of which is typically considered beyond design basis. As supporting evidence, the probability of a MSLB occurring during the LCO period of a DEI excursion can be shown to be 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 BWR steam line break outside containment is 2.20E-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. Standard Technical Specifications 3.4.7 and 3.4.8 in NUREGs-1433 and 1434 respectively report this LCO period is only 48 hours or 0.55% of the year for all BWR types. Fuel performance indicators would typically direct a plant to shut down well before it even enters this LCO. Considering that the MSLB accident is independent of the chemistry excursion that leads to the DEI exceeding the maximum steady-state value (typically 0.2 µCi/g), the probability of a MSLB occurring during this LCO period is then only 1.21E-5. 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 MSLB case assuming a pre-existing spike should not be included in the plant design bases.
Proposal: Eliminate the pre-accident iodine spike as a requirement for licensing basis analyses. 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.

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