NUREG-1777,
REGULATORY EFFECTI

From: Brian Magnuson <magnuson28@msn.com>
Sent: Sunday, April 24, 2022 10:34 PM
To: Wang, Weidong <Weidong.Wang@nrc.gov>
Cc: Karen Gray <kareng@whistleblower.org>; Jack Kolar <jackk@whistleblower.org>; Stephani Ayers <stephani@whistleblowerdefenders.com>; Thad Guyer <thad@guyerayers.com>; Dickson, Elijah <Elijah.Dickson@nrc.gov>; Frank Newell <frank@loevy.com>; Snodderly, Michael <Michael.Snodderly@nrc.gov>; Widmayer, Derek <Derek.Widmayer@nrc.gov>; Meighan, Sean <Sean.Meighan@nrc.gov>; Blumberg, Mark <Mark.Blumberg@nrc.gov>; Smith, Micheal <Micheal.Smith@nrc.gov>; Burkhart, Larry <Lawrence.Burkhart@nrc.gov>; Petti, David <David.Petti@nrc.gov>; Vasavada, Shilp <Shilp.Vasavada@nrc.gov>; Jones, Steve <Steve.Jones@nrc.gov>; Parillo, John <John.Parillo@nrc.gov>; Chang, Helen <Helen.Chang@nrc.gov>; Bladey, Cindy <Cindy.Bladey@nrc.gov>; Foli, Adakou <Adakou.Foli@nrc.gov>; Berrios, Ilka <Ilka.Berrios@nrc.gov>; Firth, James <James.Firth@nrc.gov>; Burnell, Scott <Scott.Burnell@nrc.gov>; Shepherd, Jill <Jill.Shepherd@nrc.gov>; Moore, Scott <Scott.Moore@nrc.gov>; OIGHOTLINE Resource <OIGHOTLINE.Resource@nrc.gov>
Subject: [External_Sender] RE: RE: ACRS Subcommittee Meeting - Regulatory Guide 1.183 (DG-1389) "Alternative Radiological Source Terms for Evaluating Design Basis Accident at Nuclear Power Reactors" - Regulatory Effectiveness

Dear ACRS Members and NRC Staff:

Please let me know when my March 28 public comments are posted in ADAMS.

Attached is NUREG-1777, "*Regulatory Effectiveness Assessment of Option B of Appendix J.*" This assessment concluded that the voluntary Option B of 10 CFR 50, Appendix J, *Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors*, was effective.

Notably, while making this conclusion, NUREG-1777 unapologetically explains that the gross ineffectiveness of the mandated *General Design Criteria for Nuclear Power Plants, Criterion—16 Containment Design* was, in part, the basis for establishing the voluntary Option B that relaxed the containment leak testing requirements of Appendix J.

NUREG-1777 states:

> *"Reactor containments constitute one of the principal lines of defense in the defense-in-depth design philosophy embodied in the current generation of light water power reactors."*
>
> *And,*
>
> *"10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, Criterion 16, mandates that the primary containment provide an essentially leak-tight barrier to protect against uncontrolled release of radioactivity to the environment following postulated accidents."*
>
> Regardless,

"Several mechanisms can cause releases to the environment. These include gross failure of containment due to the pressure forces resulting from an accident, containment base-mat melt-through, failure of containment isolation systems, interfacing system loss-of-coolant accidents, steam generator tube ruptures, and releases as a result of containment leakage."

And,

"Containment leakage is a small contributor to overall accident risk. At the lower end of changes in leakage rates, any uncertainties associated with the calculated leakage contribution are minuscule in comparison with other uncertainties, (e.g., prediction of containment failure mode probabilities and magnitudes of fission product source terms.)"

NUREG-1777 quotes:

- *"The effect of containment leakage is small since risk is dominated by accident sequences that result in failure or bypass of containment."* —NUREG/CR-4330, "Review of Light Water Reactor Regulatory Requirements" (1986)

- *". . . the overall levels of risk due to containment leakage are less than previous studies because accident risks are dominated by scenarios where the containment fails or is bypassed. -A major finding is that maintaining containment structural integrity post-accident is much more important than containment leak tightness."* —NUREG-1150, "Severe Accident Risk: An Assessment of Five U.S. Nuclear Power Plants" Vols. 1 and 2 (1990)

- "Risk is dominated by containment failure and bypass following severe accidents." –NUREG-1560, Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance" (1997)

These, and other, reactor accident studies; the catastrophic containment failures at Fukushima (U.S. designed BWR plants) and; NRC Order EA-13-109, clearly establish that it is physically and economically infeasible to prevent containment failures in credible nuclear accidents. The best that can be done to prevent catastrophic containment failures is to intentionally release the radiation they were inadequately designed to contain. Unfortunately, this only option is not in the best interest of public safety or clean environment.

It is important to recognize that intentionally releasing large quantities of highly radioactive material to the environment is a poor *"defense-in-depth design philosophy."* This current strategy does not defend or protect people or the environment from credible nuclear accidents. Instead, it concedes that intentionally polluting the environment with large quantities of highly radioactive material is better than polluting the environment with larger quantities of highly radioactive material. It concedes that overexposing people to radiation is better than overexposing larger populations.

Not only does NRC Order EA-13-109 confirm that GDC-16 was ineffective, it negates GDC-16. It still seems that some form of regulatory assessment would be prudent. What other nuclear safety regulations are negated or known to be ineffective?

The NRC would not have issued Order EA-13-109 if they knew reactor containments would not fail in credible nuclear accidents. By requiring that containments be vented during credible severe accidents, the NRC essentially created a "*new or different kind of accident from any*

accident previously evaluated" that "*involve a significant reduction in a margin to safety*" (§50.92).

If the 10 CFR 50.67 is to be effective, it seems that any revision to RG 1.183 must account for the source terms of credible nuclear accidents that require the use of Hardened Containment Vents or result in containment failures.

Sincerely,
Brian Magnuson

References:

(1989) GENERIC LETTER 89-16, *Installation of A Hardened Wetwell Vent:*

As a part of a comprehensive plan for closing severe accident issues, the staff undertook a program to determine if any actions should be taken, on a generic basis, to reduce the vulnerability of BWR Mark I containments to severe accident challenges. At the conclusion of the Mark I Containment Performance Improvement Program, the staff identified a number of plant modifications that substantially enhance the plants' capability to both prevent and mitigate the consequences of severe accidents. The improvements that were recommended include (1) Improved hardened wetwell vent capability, (2 improved reactor pressure vessel depressurization system reliability,
(3 an alternative water supply to the reactor vessel and drywell sprays, and (4 updated emergency procedures and training. The staff as part of that effort also evaluated various mechanisms for implementing of these plant improvements so that the licensee and the staff efforts would result in a coordinated coherent approach to resolution of severe accident issues in accordance with the Commission's severe accident policy.

After considering the proposed Mark I Containment Performance Program (described in SECY 89-017, January 1989), the Commission directed the staff to pursue Mark I enhancements on a plant-specific basis in order to account for possible unique design differences that may bear on the necessity and nature of specific safety improvements. Accordingly, the Commission concluded that the recommended safety improvements, with one exception, that is, hardened wetwell vent capability, should be evaluated by licensees as part of the Individual Plant Examination (IPE) Program. With regard to the recommended plant improvement dealing with hardened vent capability, the Commission, in recognition of the circumstances and benefits-associated with this modification, has directed a different approach.

The staff believes that the available information for installation provides strong incentive of a hardened vent. First, it is recognized that all affected plants have in place emergency procedures directing the operator to vent under certain circumstances (primarily to avoid exceeding the primary containment pressure limit) from the wetwell airspace. Thus, incorporation of a designated capability consistent with the objectives of the emergency procedure guidelines is seen as a logical and prudent plant improvement. Continued reliance on pre-existing capability (non-pressure-bearing vent path) which may jeopardize access to vital plant areas or other equipment is an unnecessary complication that threatens accident management strategies. Second, implementation of reliable venting capability and procedures can reduce the likelihood of core melt from accident sequences involving loss of long-term decay heat removal by about a factor of 10. Reliable venting capability is also beneficial, depending on plant design and capabilities, in reducing the likelihood of core melt from other accident initiators, for example, station blackout and anticipated transients without scram. As a

mitigation measure, a reliable wetwell vent provides assurance of pressure relief through a path with significant scrubbing of fission products and can result in lower releases even for containment failure modes not associated with pressurization (i.e., liner meltthrough). Finally, a reliable hardened wetwell vent allows for consideration of coordinated accident management strategies by providing design capability consistent with safety objectives. For the aforementioned reasons, the staff concludes that a plant modification is highly desirable and a prudent engineering solution of issues surrounding complex and uncertain phenomena.

(2013) NUCLEAR REGULATORY COMMISSION EA-13-109, *Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions*:

The events at the Fukushima Dai-ichi nuclear power plant following the March 2011 earthquake and tsunami highlight the possibility that events such as rare natural phenomena could challenge the traditional defense-in-depth protections related to preventing accidents, mitigating accidents to prevent the release of radioactive materials, and taking actions to protect the public should a release occur. At Fukushima Dai-ichi, limitations in time and unpredictable conditions associated with the accident significantly hindered attempts by the operators to prevent core damage and containment failure.

The requirements in this Order, in addition to providing a reliable HCVS to assist in preventing core damage when heat removal capability is lost (the purpose of EA-12-050), will ensure that venting functions are also available during severe accident conditions. Severe accident conditions include the elevated temperatures, pressures, radiation levels, and combustible gas concentrations, such as hydrogen and carbon monoxide, associated with accidents involving extensive core damage, including accidents involving a breach of the reactor vessel by molten core debris.

Ensuring that the venting functions are available under severe accident conditions will support the strategies in the Mark I and Mark II severe accident management guidelines for the protection or recovery of the containment, which serves as a barrier to the release of radioactive materials.

In EA-12-050, the Commission determined that, in light of the events at Fukushima Dai-ichi and consistent with the NRC's defense-in-depth strategy, installation of reliable hardened containment vents to help prevent core damage in BWRs with Mark I and Mark II containments was necessary to provide reasonable assurance of adequate protection of public health and safety.

Although venting the containment during severe accident conditions could result in the release of radioactive materials, venting could also prevent containment structural and gross penetration leakage failures due to overpressurization that would hamper accident management (e.g., continuing efforts to cool core debris) and ultimately result in larger, uncontrolled releases of radioactive material.

Under the backfit provisions of 10 CFR 50.109, "Backfitting," the NRC may require plant improvements beyond those needed to provide reasonable assurance of adequate protection of public health and safety when engineering approaches are available to provide a cost-justified substantial safety improvement.

The staff performed a detailed regulatory analysis of possible improvements to Mark I and Mark II reliable hardened containment vents, including the option of installing severe accident capable vents. That analysis is available in the NRC's Agencywide Documents Access and Management

System (ADAMS) at Accession No. ML12312A456. A summary of the staff's cost-benefit evaluation was provided in SECY-12-0157.

As discussed in SECY-12-0157, the NRC's determination that a venting system should be available during severe accident conditions considered both quantitative assessments of costs and benefits, as well as, various qualitative factors. Among the qualitative factors, one of the more important is enhancing the defense-in-depth characteristics of Mark I and Mark II containments by addressing the relatively high probabilities that those containments would fail should an accident progress to melting the core.

Other qualitative factors supporting installation of severe accident capable vents include addressing uncertainties in the understanding of severe accident events, supporting severe accident management and response, improving the control of hydrogen generated during severe accidents, improving readiness for external and multi-unit events, and reducing uncertainties about radiological releases and thereby improving emergency planning and response.

The Commission has determined that requiring BWR facilities with Mark I and Mark II containments to make the necessary plant modifications and procedure changes to provide a reliable hardened venting system that is capable of performing under severe accident conditions is a cost-justified substantial safety improvement. These modifications are needed to protect health and to minimize danger to life or property because they will give licensees greater capabilities to respond to severe accidents and limit the uncontrolled release of radioactive materials. In such situations, the Commission may act in accordance with its statutory authority under Section 161 of the Atomic Energy Act of 1954, as amended, to require Licensees to take appropriate action to reduce the risks posed to the public from the operation of nuclear power plants.

For Mark I containments, the preferred venting path is from the wetwell portion of containment because the water in the suppression pool provides a degree of decontamination before release to the environment. The benefits of the suppression pool in the scrubbing of possible releases when using the wetwell vents for pressure control were described in Generic Letter 89-16, "Installation of a Hardened Wetwell Vent." In addition, the wetwell venting path has been incorporated into other parts of the mitigating strategies to address lessons learned from the Fukushima Dai-ichi accident.

During severe accidents involving molten core debris breaching the reactor vessel, mitigating strategies include injecting water into the containment to help prevent drywell liner [containment] melt-through, which would result in a release pathway directly into the reactor building. However, water injection can eventually increase the water level in the suppression pool to a point where venting from the wetwell would no longer be possible. Without venting, containment pressure would continue to increase, threatening containment failure.

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Sent: Sunday, April 24, 2022 6:48:13 PM
To: Wang, Weidong <Weidong.Wang@nrc.gov>
Cc: Karen Gray <kareng@whistleblower.org>; Jack Kolar <jackk@whistleblower.org>; Stephani Ayers <stephani@whistleblowerdefenders.com>; Thad Guyer <thad@guyerayers.com>; Dickson, Elijah <Elijah.Dickson@nrc.gov>; Frank Newell <frank@loevy.com>; Snodderly, Michael <Michael.Snodderly@nrc.gov>; Widmayer, Derek <Derek.Widmayer@nrc.gov>; Meighan, Sean <Sean.Meighan@nrc.gov>; Blumberg, Mark <Mark.Blumberg@nrc.gov>; Smith, Micheal <Micheal.Smith@nrc.gov>; Burkhart, Larry <Lawrence.Burkhart@nrc.gov>; Petti, David

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Subject: RE: RE: ACRS Subcommittee Meeting - Regulatory Guide 1.183 (DG-1389) "Alternative Radiological Source Terms for Evaluating Design Basis Accident at Nuclear Power Reactors" - Inquiries

Dear Mr. Burkhart and Mr. Wang:

I appreciate the information.

Thank you,
Brian

From: Wang, Weidong
Sent: Monday, April 18, 2022 7:15 AM
To: Brian Magnuson
Cc: Karen Gray; Jack Kolar; Stephani Ayers; Thad Guyer; Dickson, Elijah; Frank Newell; Snodderly, Michael; Widmayer, Derek; Meighan, Sean; Blumberg, Mark; Smith, Micheal; Burkhart, Larry; Petti, David; Vasavada, Shilp; Jones, Steve; Parillo, John; Chang, Helen; Bladey, Cindy; Foli, Adakou; Berrios, Ilka; Firth, James; Burnell, Scott; Shepherd, Jill; Moore, Scott
Subject: RE: RE: ACRS Subcommittee Meeting - Regulatory Guide 1.183 (DG-1389) "Alternative Radiological Source Terms for Evaluating Design Basis Accident at Nuclear Power Reactors" - Inquiries

Dear Mr. Magnuson,

The transcripts of the March 16 ACRS meeting, with your written comments attached, is published in ACRS/NRC web page with the following link:

https://www.nrc.gov/docs/ML2209/ML22095A095.pdf

Feel free to let me know any questions.

Weidong Wang
ACRS/NRC

From: Burkhart, Larry <Lawrence.Burkhart@nrc.gov>
Sent: Saturday, April 16, 2022 6:26 PM
To: Brian Magnuson <magnuson28@msn.com>
Cc: Karen Gray <kareng@whistleblower.org>; Jack Kolar <jackk@whistleblower.org>; Stephani Ayers <stephani@whistleblowerdefenders.com>; Thad Guyer <thad@guyerayers.com>; Dickson, Elijah <Elijah.Dickson@nrc.gov>; Frank Newell <frank@loevy.com>; Snodderly, Michael <Michael.Snodderly@nrc.gov>; Widmayer, Derek <Derek.Widmayer@nrc.gov>; Wang, Weidong <Weidong.Wang@nrc.gov>; Meighan, Sean <Sean.Meighan@nrc.gov>; Blumberg, Mark <Mark.Blumberg@nrc.gov>; Smith, Micheal <Micheal.Smith@nrc.gov>; Petti, David <David.Petti@nrc.gov>; Vasavada, Shilp <Shilp.Vasavada@nrc.gov>; Jones, Steve <Steve.Jones@nrc.gov>; Parillo, John <John.Parillo@nrc.gov>; Chang, Helen <Helen.Chang@nrc.gov>; Bladey, Cindy <Cindy.Bladey@nrc.gov>; Foli, Adakou <Adakou.Foli@nrc.gov>; Berrios, Ilka <Ilka.Berrios@nrc.gov>; Firth, James <James.Firth@nrc.gov>; Burnell, Scott <Scott.Burnell@nrc.gov>; Shepherd, Jill <Jill.Shepherd@nrc.gov>; Moore, Scott <Scott.Moore@nrc.gov>
Subject: RE: RE: ACRS Subcommittee Meeting - Regulatory Guide 1.183 (DG-1389) "Alternative Radiological Source Terms for Evaluating Design Basis Accident at Nuclear Power Reactors" - Inquiries

Dear Mr. Magnuson,

I will be responding to you regarding the portion of your email that pertains to the ACRS (I am not an ACRS member but I am the Chief of the Technical Support Branch for the ACRS). Thank you for observing the ACRS activities and providing your comments to the ACRS. Although the ACRS does not respond to specific public comments, the information you provided will be taken into account, along with the NRC staff and other stakeholder input, as the Committee decides on any advice it will provide to the Commission. Please watch the following website for future meetings concerning these topics: https://www.nrc.gov/reading-rm/doc-collections/acrs/agenda/2022/index.html

Regarding the transcripts of the March 16 ACRS meeting, Mr. Weidong Wang of my staff will keep you informed on that status. Your written comments should be attached to those transcripts and slides for that meeting.

Separate from ACRS activities, the NRC staff may respond to your comments and questions regarding the technical merit of those comments and may provide you information on the status of your petition.

Thank you for your continued interest in the ACRS activities.

Larry Burkhart

From: Brian Magnuson <magnuson28@msn.com>
Sent: Saturday, April 16, 2022 10:21 AM
To: Meighan, Sean <Sean.Meighan@nrc.gov>; Wang, Weidong <Weidong.Wang@nrc.gov>; Burkhart, Larry <Lawrence.Burkhart@nrc.gov>; Blumberg, Mark <Mark.Blumberg@nrc.gov>; Smith, Micheal <Micheal.Smith@nrc.gov>; Petti, David <David.Petti@nrc.gov>; Dickson, Elijah <Elijah.Dickson@nrc.gov>; Vasavada, Shilp <Shilp.Vasavada@nrc.gov>; Jones, Steve <Steve.Jones@nrc.gov>; Parillo, John <John.Parillo@nrc.gov>; Chang, Helen <Helen.Chang@nrc.gov>; Bladey, Cindy <Cindy.Bladey@nrc.gov>; Foli, Adakou <Adakou.Foli@nrc.gov>; Berrios, Ilka <Ilka.Berrios@nrc.gov>; Firth, James <James.Firth@nrc.gov>; Burnell, Scott <Scott.Burnell@nrc.gov>; Shepherd, Jill <Jill.Shepherd@nrc.gov>; Shepherd, Jill <Jill.Shepherd@nrc.gov>
Cc: Karen Gray <kareng@whistleblower.org>; Jack Kolar <jackk@whistleblower.org>; Stephani Ayers <stephani@whistleblowerdefenders.com>; Thad Guyer <thad@guyerayers.com>; Frank Newell <frank@loevy.com>
Subject: [External_Sender] RE: ACRS Subcommittee Meeting - Regulatory Guide 1.183 (DG-1389) "Alternative Radiological Source Terms for Evaluating Design Basis Accident at Nuclear Power Reactors" - Inquiries

April 16, 2022

Dear NRC Staff and ACRS Members:

Please let me know when the transcript of the March 16, 2022 meeting is available and all of my March 2022 public comments are posted in ADAMS.

Also, when should I expected documented responses to my December 2020 public comments?

Attached is NRC Regulatory Issue Summary 2001-19: *Deficiencies in the Documentation of Design Basis Radiological Analyses Submitted in Conjunction with License Amendment Requests.* Please recognize that it substantiates my March 28 public comments copied here.

"Ironically, the design-basis contravention must deviate from design-basis, because even using the contravention, nuclear power plants cannot comply with GDC-19. This is why RG 1.183 and DG-1389 lowers design-basis standards. They credit the use of non-safety related and non-seismically qualified systems and components—that are not credited in design-basis analyses. Proffered inspections of non-safety related equipment (e.g., piping, condensers) do not satisfy legitimate design-basis analyses that can only credit safety-related equipment. RG-1389's "seismically rugged" is simply an artifice; legitimate design-basis seismic analyses refer to these components as "seismically unqualified.""

Additionally, please provide the current status of my 10 CFR 2.803 Petition for Rulemaking, "*Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*," (PRM-50-122; NRC 2020 0150).

Sincerely,
Brian Magnuson

From: Brian Magnuson <magnuson28@msn.com>
Sent: Monday, March 28, 2022 10:30:07 PM
To: Meighan, Sean <Sean.Meighan@nrc.gov>; weidong.wang@nrc.gov <Weidong.Wang@nrc.gov>; Burkhart, Larry <Lawrence.Burkhart@nrc.gov>; Blumberg, Mark <Mark.Blumberg@nrc.gov>; Smith, Micheal <Micheal.Smith@nrc.gov>; david.petti@nrc.gov <david.petti@nrc.gov>; elijah.dickson@nrc.gov <elijah.dickson@nrc.gov>; shilp.vasavada@nrc.gov <shilp.vasavada@nrc.gov>; steve.jones@nrc.gov <steve.jones@nrc.gov>; john.parillo@nrc.gov <john.parillo@nrc.gov>
Subject: RE: ACRS Subcommittee Meeting - Regulatory Guide 1.183 (DG-1389) "Alternative Radiological Source Terms for Evaluating Design Basis Accident at Nuclear Power Reactors"

RE: ACRS Subcommittee Meeting on Revision of Regulatory Guide 1.183 "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"

March 28, 2022

Dear NRC Staff and ACRS Members:

I appreciate the opportunity to attend and provide public comments during the ACRS Subcommittee on March 16, 2022. Nonetheless, I do not believe the NRC has been responsive to my prior RG 1.183 (DG-1199) public comments. The NRC posted my public comments (and questions) in ADAMS (ML20343A064 and ML20351A321), but has not provided documented responses. In effect, this negates my efforts and the intent of NRC public meetings.

Below (and attached) are my post-meeting public comments.

Prior RG 1.183 Public Meeting Comments

I resubmit my December 2020 public comments and understand that documented responses are now forthcoming.

I submit my 10 CFR 2.803 Petition for Rulemaking (PRM-50-122), "*Accident Source Term Methodologies and Corresponding Release Fractions*" (NRC-2020-0150) and my subsequent

public comments (Comment (2) from Brian Magnuson on FR Doc # 2020-17645, posted by the NRC on November 24, 2020.).

Additional ACRS Public Meeting Comments

I submitted my pre-meeting public comments to the NRC and ACRS in email dated March 15, 2022.

Additionally, I submit the following documents [attached] that are available to the public:

(1) 2010.01.06 BWROG DG-1199 Comments (ML100081013)
(2) 2020.05.03 Petition to Amend 10 CFR 50.67, Accident Source Term – Magnuson PRM 50-122
(3) 2020.11.08 Magnuson Comments on PRM 50-122
(4) 2009.06.11 Response to Non-Concurrence to DG-1199 (ML091520056)

My prior public comments referenced SAND2008-6601 which determined the BWR MSIV source term methodologies provided in RG 1.183 (Revision 0) are "*non-conservative and conceptually inaccurate*" in 2008. Additionally, my prior comments expounded on SAND2008-6601 and identified other examples in which RG 1.183 methodologies violate the laws of physics. RG 1.183 allows nuclear power plants (NPPs) to ignore the laws of physics in accident dose calculations that are used to demonstrate compliance with nuclear safety regulations, including General Design Criterion-19 (Appendix A to 10 CFR Part 50). In other words, the errors in RG 1.183 financially benefit nuclear power plants at the expense of public safety.

It appears DG-1389 may correct a few of the technical errors in Revision 0 of RG 1.183; however, any corrections would be negated because it states:

> *"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."*

RG 1.183 Revision 0 has a broad range of safety ramifications. Until the NRC has reconciled the errors that SAND2008-6601 identified and I reported in prior public comments, it seems reckless (and disrespectful) of the NRC to claim it is an acceptable method for demonstrating compliance with regulations. In effect, the errors identified in RG 1.183 Revision 0 provide a means for nuclear power plants to falsify accident dose calculations and feign compliance with federal nuclear safety regulations.

The (Beyond) Design-Basis Accident Contravention

For the reasons stated in my pre-meeting public comments, I am opposed to using the DG-1389 term "*maximum hypothetical accident (MHA) loss-of-coolant accident (LOCA)."* An NRC Regulatory/Draft guide cannot legally be used to redefine *"the accident described in the applicable regulations."* For example, the applicability of Appendix A to Part 50, General Design Criterion—19 cannot be limited. Nevertheless, the apparent attempt drew attention to the most egregious contravention of RG 1.183 (and DG-1389).

To begin, the NRC acknowledged: "*In 1971 Appendix A, "General Design Criteria for Nuclear Power Plants," was added to 10 CFR Part 50. General Design Criterion 19 (GDC-19) specified that adequate protection shall be provided to permit access and occupancy of the control room for the duration of an accident without exceeding a radiation exposure of 5 rem whole body or its equivalent to any part of the body. <u>From its inception, GDC-19 became the limiting dose criteria in almost all radiological dose consequence analyses</u>."*

To be clear, GDC-19 was not "limiting" by the late 1970s. By then, the NRC discovered that BWR MSIV leakage was a significant contributor to control room operator doses. Despite this disturbing discovery, the NRC neglected to require nuclear power plants to add this contribution to their accident dose calculations. However, in 2000, the NRC suggested that some nuclear power plants might wish to add MSIV leakage dose contributions to their accident dose calculations if they wanted to reap the "*cost-beneficial licensing actions*" provided by RG 1.183.

Despite Sandia National Laboratories (SAND2008-6601) and my reports, the NRC continues to allow nuclear power plants to exploit the RG 1.183 errors for "*cost-beneficial licensing actions.*" The NRC allowed nuclear power plants to exploit the errors to (1) increase MSIV technical specification allowable leakage; (2) increase reactor thermal power (electrical generation); (3) increase fuel burnup times and; (4) extend (sometimes twice) the licensing life of old nuclear power plants—that have been violating GDC-19 since its inception.

Based on the timeline of NRC actions since 1971, it appears an underlying purpose of RG 1.183 is to evade the minimum design criteria set forth in Appendix A to Part 50.

Despite the many significant design modifications that have been required in the last 50 years, nuclear power plants cannot be made <u>legally</u> safe. They were not designed based on the "maximum credible accident" as required by TID-14844. Instead, they were mistakenly designed on its poor (but conscientious) example. Because of the complex ways in which nuclear plants structures, and components can fail during credible nuclear accidents, it is economically infeasible to redesign or retrofit old nuclear power plants to comply with GDC-19. This motivates the industry to circumvent the deterministic requirements of GDC-19.

In 2019, the NRC further acknowledges that "*the control room accident dose criterion has proven to be challenging to demonstrate with most plants having very little margin to the regulation* [Appendix B to Part 50, GDC-19]. Does "*very little margin*" to GDC-19 "*provide sufficient safety margins with adequate defense in depth to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameter inputs*" as is apparently required by RG 1.183?

As documented in my pre-meeting comments, the NRC clearly knows that the "*Protection by Multiple Fission Product Barriers*" are grossly inadequate. They cannot protect people (and the environment) from severe nuclear accidents as required by 10 CFR 100.11 and 10 CFR 50.67. In fact, the inferior design of these barriers will cause them to overheat, create explosive gases, and catastrophically self-destruct during credible accidents.

Because many of us watched the containment barriers at Fukushima explode, the NRC was compelled to require similar nuclear power plants to install Hardened Containment Vents. In recognition that containment barriers will fail, the NRC now requires nuclear power plant operators to use the Hardened Containment Vents, during credible accidents, to release large quantities of highly radioactive material directly to the environment to prevent their containment barriers from self-destructing and releasing much more radioactive material.
After studying severe accidents for decades—admitting that containment barriers will fail in credible nuclear accidents and watching the containment barriers at Fukushima catastrophically fail, the NRC wrongly allows nuclear power plants to assume that containments will not fail in accident dose calculations using RG 1.183.

Footnote[1] of DG-1389 states:

> "*These evaluations assume containment integrity with <u>offsite</u> hazards evaluated based on design basis containment leakage*."

Footnote[2] of DG-1389 states:

"*The purpose of this approach would be to test the adequacy of the containment and other safety-related systems.*"

These footnotes give rise to a circular position. DG-1389 proffers that containment barriers can be adequately tested by using design-basis evaluations that assume they will not fail. This fallacy epitomizes the NRC's design-basis contravention.

As was the case of the inadequate sea wall at Fukushima, U.S. designed nuclear power plants were/are not designed to protect people and the environment from credible nuclear accidents. The NRC's design-basis contravention attempts to legitimize the safety (and economic viability) of nuclear power plants by using the inferior standards to which they were designed as the basis for complying with regulations, instead of using the legitimate standards to which they should have been designed. It seems the contravention simply and wrongly truncated the AEC's requirement to perform Design Basis Accidents, Transients, and Events analyses.

Whether specious efforts are taken to limit the applicability of regulations to design-basis accidents or explicit claims are made to exclude credible accidents from the applicability of regulations because they are beyond (not) design-basis accidents, the end result is the same; containments and every other fission product barrier will fail in credible nuclear accidents.

Ironically, the design-basis contravention must deviate from design-basis, because even using the contravention, nuclear power plants cannot comply with GDC-19. This is why RG 1.183 and DG-1389 lowers design-basis standards. They credit the use of non-safety related and non-seismically qualified systems and components—that are not credited in design-basis analyses. Proffered inspections of non-safety related equipment (e.g., piping, condensers) do not satisfy legitimate design-basis analyses that can only credit safety-related equipment. RG-1389's "seismically rugged" is simply an artifice; legitimate design-basis seismic analyses refer to these components as "seismically unqualified."

It appears, the NRC's RG 1.183 efforts are narrowly focused on obscuring known design deficiencies—providing nuclear power plants with questionable and unscientific methods to perform accident dose calculations so they can feign compliance with regulations; circumvent deterministic regulations; bring them into compliance or; otherwise increase their profit margins. It seems as though, RG 1.183 and DG-1389 provide the means to depart from the truth.

Sincerely,
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REFERENCES:

(November 2011) NUREG-0654 FEMA-REP-1, Rev. 1 Supplement 3: Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants
Note 1: Rapidly Progressing Severe [Accident] Incident
A rapidly progressing severe incident is a General Emergency (GE) with rapid loss of containment integrity (emergency action levels indicate containment barrier loss) and loss of

ability to cool the core. This path is used for scenarios in which containment integrity can be determined as bypassed or immediately lost during a GE with core damage.

(November 2012) NEI 99-01 (Revision 6) Development of Emergency Action Levels for Non-Passive Reactors

For severe core damage events, uncertainties exist in phenomena important to accident progressions leading to containment failure. Because of these uncertainties, predicting the status of containment integrity may be difficult under severe accident conditions. This is why maintaining containment integrity alone following sequences leading to severe core damage is an insufficient basis for not escalating to a General Emergency.

PSAs [probabilistic safety assessments - also known as probabilistic risk assessment, PRA] indicated that leading contributors to latent fatalities were sequences involving a containment bypass, a large Loss of Coolant Accident (LOCA) with early containment failure, a Station Blackout lasting longer than the site-specific coping period, and a reactor coolant pump seal failure. The generic EAL methodology needs to be sufficiently rigorous to address these sequences in a timely fashion.

(November 23, 2019) PRM-50-121, Re-Submittal - 10 CFR 2.802 Petition for rulemaking Accident Dose Criteria

The proposed rule would allow licensees to adopt revised accident dose acceptance criteria as an alternative to the accident dose criteria specified in § 50.67 Accident source term. The revised accident dose criteria would be described in a separate voluntary rule § 50.67(a) specifying a uniform value of 100 milli Sieverts (10 rem) for the off-site locations and for the control room.

Problem Description:

The U.S. Nuclear Regulatory Commission's (NRC's) design basis accident (DBA) dose criteria and the resulting design of accident mitigation systems could be perceived to emphasize protection of the control room operator over protection of the public. The control room criterion restricts the calculated 30-day accident dose to the annual occupational limit of five rem while the off-site dose criteria allows for a calculated dose of 25 rem in two hours. The off-site dose criteria were derived from the siting practices of the earliest reactors and are not reflective of current health physics knowledge or modern plant construction. As a result, the design of accident mitigation systems may not be optimized in the best interest of NRC's mission of protecting public health and safety. The control room accident dose criterion has proven to be challenging to demonstrate with most plants having very little margin to the regulation.

Proposed Solution:

The proposed voluntary rule would allow licensees to adopt revised accident dose criteria that will; (1) be reflective of modern health physics recommendations and modern plant designs, (2) provide a better balance between protection of the control room operator and protection of the public, and (3) relieve the unnecessary regulatory burden associated with meeting the current control room dose criterion.

The attached petition includes the history of the current dose criteria, proposed changes to § 50.67 Accident source term and General Deign Criterion 19, corresponding revisions to Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, as well as other supporting information.

SUMMARY:

During the 1950s, applicants for reactor construction permits submitted Hazards Summary Reports to the Atomic Energy Commission (AEC) describing the potential dose consequences from what was considered the "maximum credible accident."[1] These evaluations contained wide variations in both the assumed source terms as well as the proposed dose acceptance criteria. In response to the recognition that more definitive siting criteria was needed, the AEC developed a procedural methodology to define reactor siting criteria that was generally consistent with the siting practices in effect at the time. There was a concern within the AEC that it was premature to codify these criteria so early in the development of the nuclear power industry. Notwithstanding this concern,

in 1962, the AEC published 10 CFR Part 100, "Reactor Site Criteria", specifying dose acceptance criteria of 25 rem whole body and 300 rem thyroid for a 2 hour period at the Exclusion Area Boundary (EAB) and for the accident duration at the outer boundary of the Low Population Zone (LPZ).

Control Room Dose Criterion: Objectives

In 1971 Appendix A, "General Design Criteria for Nuclear Power Plants," was added to 10 CFR Part 50. General Design Criterion 19 (GDC-19) specified that adequate protection shall be provided to permit access and occupancy of the control room for the duration of an accident without exceeding a radiation exposure of 5 rem whole body or its equivalent to any part of the body. From its inception, GDC-19 became the limiting dose criteria in almost all radiological dose consequence analyses.

The 5 rem control room dose criterion is limiting for many licensees and this raises the question regarding whether a slightly higher value could still satisfy the objective of providing a comfortable environment for the operators while reducing regulatory burden by increasing the small margin many licensees have relative to the current acceptance criterion.

There are no footnotes or notes in criterion 19 to define the accident condition tobe analyzed as is the case in 10 CFR 100.11[33]. By guidance, licensees are directed to analyze the control room radiological habitability with the same conservative assumptions and MCA source term used in the evaluation of the off-site reference values.

Additional Challenges to Meeting the Requirements of GDC-19

As can be seen by examination of representative MCA results shown in Appendix E[38] of this petition, many licensees' evaluations have a relatively small margin to the control room acceptance value. With the adoption of the TEDE dose criterion many licensees have gained operational flexibility over the previous use of a thyroid dose criterion. The current thyroid dose weighting factor being used in the calculation of TEDE is 0.03 per 10 CFR 20.1003. The International Commission of Radiation Protection (ICRP) Publication 103 has recommended the use of a thyroid weighting factor of 0.04. The NRC's Office of Nuclear Regulatory Research completed a study entitled, "Control Room Dose Evaluation Using ICRP 103 Dose Conversion Factors," letter report (ADAMS Accession No. ML17156A603), which concludes that: "Application of the ICRP 103 DCFs will result in an increase in the range of 23 to 25% in the TEDE doses for the control room." The degree of impact will depend on the amount of credit taken for various iodine removal mechanisms both natural and engineered. However, if the ICRP recommendations are ever incorporated into NRC's regulations and guidance, the incorporation of a thyroid weighting factor of 0.04 will decrease the already small margin many licensees have in their control room dose consequence analysis.

GDC-19 requires that, "*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.*" The NRC has not emphasized the issue of control room access in any of the regulatory guides dealing with control room habitability. As such most licensees do not include an evaluation of access dose in their control room dose consequence analysis.

Including access dose in the calculation of the total control room would decrease the already small margin most licensees have in their control room dose consequence analysis.

NUREG-0625 and 10 CFR 50.34

In August 1978, the Nuclear Regulatory Commission directed the staff to develop a general policy statement on nuclear power reactor siting which resulted in NUREG-0625[39], "Report of the Siting Policy Task Force." NUREG-0625 recommended that fixed distances should be required for the EAB and the LPZ.

ABSTRACT [From NUREG-0625]

In August 1978, the Nuclear Regulatory Commission directed the staff to develop a general policy statement on nuclear power reactor siting. A Task Force was formed for that purpose and has prepared a statement of current NRC policy and practice and has recommended a number of changes to current policy. The recommendations were made to accomplish the following goals:

> To strengthen siting as a factor in defense in depth by establishing requirements for site approval that are independent of plant design consideration. The present policy of permitting plant design features to compensate for unfavorable site characteristics has resulted in improved designs but has tended to deemphasize site isolation.
>
> To take into consideration in siting the risk associated with accidents beyond the design basis (Class 9) by establishing population density and distribution criteria. Plant design improvements have reduced the probability and consequences of design basis accidents but there remains the residual risk from accidents not considered in the design basis. Although this risk cannot be completely reduced to zero, it can be significantly reduced by selective siting.
>
> To require that sites selected will minimize the risk from energy generation. The selected sites should be among the best available in the-region where new generating capacity is needed. Siting requirements should be stringent enough to limit the residual risk of reactor operation but not so stringent as to eliminate the nuclear option from large regions of the country. This is because energy generation from any source has its associated risk, with risks from some energy sources being greater than that of the nuclear option.

The concern was that siting practices were not providing enough emphasis on site isolation as an important contributor to defense in depth because ESF systems such as iodine filters, containment sprays, and double containment structures could be designed to make almost any site acceptable from an accident dose calculation point of view.

*

In the late 1970s there were concerns within the NRC that siting practices were not providing enough emphasis on site isolation as an important contributor to defense-in-depth because engineered safety feature (ESF) systems could be designed to make almost any site acceptable from an accident dose calculation point of view. In August 1978, the NRC directed the staff to develop a general policy statement on nuclear power reactor siting which resulted in NUREG- 0625, “Report of the Siting Policy Task Force,” recommending that fixed distances should be required for the EAB and the LPZ in lieu of dose consequence analyses. After numerous comments objecting to a proposed rule (57 FR 47802), which was based on NUREG-0625 recommendations, the commission decided to retain source term and dose calculations by relocating a new single dose criterion based on total effective dose equivalent (TEDE) in 10 CFR 50.34 (61 FR 65157 December 11, 1996).

The new TEDE criterion is applicable to all new reactors and existing reactors that choose to adopt the alternative source term (AST) methodology. Depending on the contribution to TEDE dose from iodine in the released source term, <u>the 25 rem TEDE criterion allows for the associated thyroid dose to substantially exceed the previously controlling 300 rem thyroid limitation</u>. Therefore, new reactors are being sited with a less restrictive dose criterion than the earliest reactors.

DISCUSSION:

Hazard Summary Reports issued in the 1950's included the dose consequences from a maximum credible accident (MCA) also referred to as a maximum hypothetical accident (MHA) or a maximum probable accident (MPA). Such evaluations were based on the assumption that the plant experienced a substantial core melt releasing appreciable quantities of fission products into the containment atmosphere. These evaluations assumed containment integrity with offsite hazards evaluated based on design basis containment leakage. Applicants then evaluated the off-site radiological conditions for such an event and proffered various suggestions for dose acceptance criteria. The AEC evaluated these applications on a case by case basis without the benefit of a prescribed set of assumptions regarding the degree of core damage or defined dose acceptance criteria. There was a considerable effort in the AEC and the advisory committee on reactor safeguards (ACRS) during the time from 1958 through 1962 to devise a more systematic method to evaluate the licensee's MCA determinations. These concerns were described in an AEC report to the General Manager[4] by the Director of Licensing and Regulation on Reactor Site Criteria[5] as shown below:

> "The hazards reports as presented by the various applicants have shown a wide variation in estimating the magnitude of the maximum credible accident and in the dose calculational methods and, consequently, in the calculated exposure doses that might result to the offsite public in case of an accident. This situation is due partly to the differences in reactor plant design but even more to the different engineering judgments that can be made in analyzing possible consequences of accidents. AEC and ACRS review has emphasized evaluation of the safety factors that have been included in the plant design and evaluation of the conservatism represented in the analytical procedures as well as the numerical values derived. This subjective manner of arriving at judgment on site suitability has led to requests to have the AEC make more definitive the basis upon which the data are evaluated and to make more specific the safety criteria which govern the AEC's consideration of site suitability."

The promulgation of 10 CFR Part 100 and its basis document TID-14844 served to reduce the amount of subjectivity involved to the evaluation of reactor site suitability by defining the degree of core damage to be assumed in the MCA and by prescribing dose acceptance criteria.

Formally Stated Objectives of 10 CFR Part 100

The AEC first published a Notice of Proposed Rule Making regarding site criteria in 1959 (24 Federal Register Notice (FRN) 4184 1959)[6] announcing that:

> "The Commission is considering the formulation of an amendment to its regulations to state site criteria for the evaluation of proposed sites for nuclear power and test reactors and is publishing for comment safety factors which might be a basis for the development of site criteria."
>
> "In view of the complex nature of the environment, the wide variation in environmental conditions from one location to another and the variations in reactor characteristics and associated protection which can be engineered into a reactor facility, definitive criteria for general application to the siting problems have not been set forth."

The FRN went on to describe in general terms the need to show that, "the occurrence of any credible accident, will not create undue hazard to the health and safety of the public." The FRN described the general concept of an exclusion area under the complete control of the licensee as well as an area of low population density immediately outside the exclusion area.

In 1961, the AEC published 10 CFR Part 100, Reactor Site Criteria, Notice of Proposed Guides, (26 FRN 1224 1961)[7]. These guides were more descriptive and included specific dose criteria as well as an appendix detailing an example calculation of reactor siting distances. This FRN also included a more definitive set of objectives stating that:

"The basic objectives which it is believed can be achieved under the criteria set forth in the proposed guides, are:

Serious injury to individuals off-site should be avoided if an unlikely, but still credible, accident should occur;

Even if a more serious accident (not normally considered credible) should occur, the number of people killed should not be catastrophic;
The exposure of large numbers of people in terms of total population dose should be low. The Commission intends to give further study to this problem in an effort to develop more specific guides on this subject. Meanwhile, in order to give recognition to this concept the population center distances to very large cites may have to be greater than those suggested by these guides."

There were numerous comments[8] received on the proposed Part 100 Site Criteria published for comment on February 11, 1961. There was general agreement that the proposed site criteria represented a distinct improvement over the criteria published on May 23, 1959. There was a concern over the inclusion of the Appendix which was felt to be too descriptive to include in a rule. In addition, there were several comments that objected to the wording of the objectives especially in paragraph (b), "Even if a more serious accident (not normally considered credible) should occur, the number of people killed should not be catastrophic."

The objectives stated in the proposed guides published on February 11, 1961 were not repeated in the final rule which was published on April 13, 1962. The final rule (27 FRN 3509 1962)[9] included the following discussion concerning the objective of the population center distance described in 10 CFR Part 100:

"One basic objective of the criteria is to assure that the cumulative exposure dose to large numbers of people as a consequence of any nuclear accident should be low in comparison with what might be considered reasonable for total population dose. Further, since accidents of greater potential hazard than those commonly postulated as representing an upper limit are conceivable, although highly improbable, it was considered desirable to provide for protection against excessive exposure doses to people in large centers, where effective protective measures might not be feasible. Neither of these objectives were readily achievable by a single criterion. Hence, the population center distance was added as a site requirement when it was found for several projects evaluated that the specification of such a distance requirement would approximately fulfill the desired objectives and reflect a more accurate guide to current siting practices. In an effort to develop more specific guidance on the total man-dose concept, the Commission intends to give further study to the subject. Meanwhile, in some cases where very large cities are involved, the population center distance may have to be greater than those suggested by these guides."

Background on the Development of 10 CFR Part 100 – Reactor Site Criteria

The minutes of the ACRS subcommittee held on August 23, 1960[12], contained a draft of site criteria which defined the basis for an Exclusion Area, an Evacuation Area (later termed the Low Population Zone, and a City Distance (later termed Population center distance) as follows:

Exclusion Area -- An area whose radius is not less than the distance at which total radiation doses received by an individual fully exposed for two hours to the radioactive consequences of the maximum credible accident would be above 25 R (or equivalent). The area should be under the full control of the applicant. Residents subject to ready evacuation are allowed.

Evacuation Area -- An area whose radius is not less than the distance at which total radiation doses received by an individual fully exposed for the entire maximum credible accident would be above 25 R (or equivalent). Total

population not to exceed 10,000 people and no more than 2,000 in any 45° sector.

City Distance -- Distance from reactor to nearest fringe of high density population of a substantial city (above 10,000) which must not be less than distance at which total radiation doses received by a person exposed for the entire maximum credible accident would be above 10 R or equivalent. The real basis, however, for this criterion is an uncontained "puff" release" resulting in a LD-50 dose at the city boundary.

This statement by Dr. Beck that, “The real basis, however, for this criterion is an uncontained puff release of radioactivity resulting in an LD-50 [50 percent chance of death without medical intervention] dose at the city boundary,” relates to the objective stated in the proposed rule that, “Even if a more serious accident (not normally considered credible) should occur, the number of people killed should not be catastrophic.” This statement indicates that the actual criterion in mind was that the distance to the nearest city would be large enough that if the core melted, the **containment failed**, and all the volatile fission products were released with the wind blowing toward the city, the dose at the city boundary would be that which was estimated to kill half the people exposed to its full effect.[13] The severe accident analysis at the time was WASH 740 which predicted 3,400 acute early fatalities for a worst case reactor accident. [14]

In his testimony at the JCAE Hearings, on Radiation Safety and Regulation, June 12-15, 1961, Mr. Robert Loewenstein, Acting Director, AEC Division of Licensing and Regulations specifically discussed the population center distance as follows[15]:

> "If one could be absolutely certain that no accident greater than the "maximum credible accident" would occur, then the 'exclusion area' and 'low population' zone would provide reasonable protection to the public under all circumstances. There does exist, however, a theoretical possibility that substantially larger accidents could occur. It is believed prudent at present, when the practice of nuclear technology does not rest on a solid foundation of extended experience, to provide protection against the most serious consequences of such theoretically possible accidents. Consideration of a 'population center distance' is therefore prescribed: This is a distance by which the reactor would be so removed from the nearest major concentration of people that lethal exposures would not occur in the population center **even from an accident in which the containment is breached**[16]."

Regulatory Guide 1.3.(Revision ***2) “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling”***

Section 50.34 of 10 CFR Part 50 requires that each applicant for a construction permit or operating license provide an analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility. The design basis loss of coolant accident (LOCA) is one of the postulated accidents used to evaluate the adequacy of these structures, systems, and components with respect to the public health and safety.

After reviewing a number of applications for construction permits and operating licenses for boiling water power reactors, the AEC Regulatory staff has developed a number of appropriately conservative assumptions, based on engineering judgment and on applicable experimental results from safety research programs conducted by the AEC and the nuclear industry, that are used to evaluate calculations of the radiological consequences of various postulated accidents.

This guide lists acceptable assumptions that may be used to evaluate the design basis LOCA of a Boiling Water Reactor (BWR). It should be shown that the offsite dose consequences will be within the guidelines of 10 CFR Part 100. (During the construction

permit review, guideline, exposures of 20 rem whole body and 150 rem thyroid should be used rather than the values given in § 100.11 in order to allow for (a) uncertainties in final design details and meteorology or (b) new data and calculational techniques that might influence the final design of engineered safety features or the dose reduction factors allowed for these features.)

C. REGULATORY POSITION

1. The assumptions related to the release of radioactive material from the fuel and containment are as follows:

a. Twenty-five percent of the equilibrium radioactive iodine inventory developed from maximum full power operation of the core should be assumed to be immediately available for leakage from the primary reactor containment. Ninety-one percent of this 25 percent is to be assumed to be in the form of elemental iodine, 5 percent of this 25 percent in the form of particulate iodine, and 4 percent of this 25 percent in the form of organic iodides.

b. One hundred percent of the equilibrium radioactive noble gas inventory developed from maximum full power operation of the core should be assumed to be immediately available for leakage from the reactor containment.

c. The effects of radiological decay during holdup in the containment or other buildings should be taken into account.

d. The reduction in the amount of radioactive material available for leakage to the environment by containment sprays, recirculating filter systems, or other engineered safety features may be taken into account, but the amount of reduction in concentration of radioactive materials should be evaluated on an individual case basis.

e. The primary containment should be assumed to leak at the leak rate incorporated or to be incorporated in the technical specifications for the duration of the accident. The leakage should be assumed to pass directly to the emergency exhaust system without mixing in the surrounding reactor building atmosphere and should then be assumed to be released as an elevated plume for those facilities with stacks.

f. No credit should be given for retention of iodine in the suppression pool.

Bases for Withdrawal -2016

The NRC is withdrawing RG 1.3 because it is outdated. The guidance contained in RG 1.3 has been updated and incorporated into RG 1.183 and RG 1.195. The information in RG 1.183 provides guidance for new and existing LWR plants that have adopted the AST, and RG 1.195 provides guidance for those LWR plants that have not adopted the AST.

(June 11, 2009) RESPONSE TO A NON-CONCURRENCE ON DRAFT REGULATORY GUIDE DG-1199, "ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS"

Since the publication of TID-14844 in 1962, significant advances have been made in the understanding of radioactivity released from severe nuclear power plant accidents. In 1995, the NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." NUREG-1465 uses updated research from the 1980's that provides a more realistic estimate of the accident source term, including its mix, magnitude, chemical and physical form, and timing of release.

The NRC staff anticipated that some licensees, who used TID-14844 to design their facilities, may wish to update their design bases using the NUREG-1465 source term to take advantage of the more realistic information it provides. The NRC staff, therefore, initiated several actions to provide a regulatory basis for these licensees to use an alternative source term (AST) in design basis analyses. These initiatives resulted in the development and issuance of Title 10 of the Code of Federal Regulation (10 CFR) Section 50.67 (50.67), "Accident source term."

The NRC, via regulations such as the performance-based 10 CFR 50.67, regulates all U.S. commercial nuclear power plants. 10 CFR 50.67 is an alternative voluntary regulation that allows licensees to revise the accident source term. This source term is used in the radiological analyses for designing their plant. This analysis is often referred to as a "design basis" analysis

and the hypothetical or postulated events used to test the facility are known as “design basis accidents” (DBAs).

NUREG/CR-7155 - SAND2012-10702P, State-of-the-Art Reactor Consequence Analyses Project - Uncertainty Analysis of the Unmitigated Long-Term Station Blackout of the Peach Bottom Atomic Power Station

The U.S. Nuclear Regulatory Commission (NRC), the nuclear power industry, and the international nuclear energy research community have devoted considerable research over the last several decades to examining severe reactor accident phenomena and offsite consequences. The NRC initiated the State-of-the-Art Reactor Consequence Analyses (SOARCA) project to leverage this research and develop current estimates of the offsite radiological health consequences for potential severe reactor accidents for two pilot plants: the Peach Bottom Atomic Power Station, a boiling-water reactor (BWR) in Pennsylvania and the Surry Power Station, a pressurized-water reactor in Virginia. By applying modern analysis tools and techniques, the SOARCA project developed a body of knowledge regarding the realistic outcomes of select severe nuclear reactor accidents.

This document describes the NRC’s uncertainty analysis of the SOARCA unmitigated long-term station blackout (LTSBO) severe accident scenario for the Peach Bottom Atomic Power Station.

Performing the source term calculations of the Peach Bottom unmitigated LTSBO uncertainty analysis revealed three groupings of similar accident progression sequences within the Peach Bottom unmitigated LTSBO scenario: (1) early stochastic failure of the cycling SRV, which was the deterministic SOARCA scenario in NUREG-1935; (2) thermal failure of the SRV without main steam line (MSL) creep rupture; and (3) thermal failure of the SRV with MSL creep rupture. The three sequence groups exhibited differences in release magnitude, with MSL failure generally leading to the largest environmental releases.

The SOARCA analyses [2] of station blackout accidents in Peach Bottom were performed several years before the accidents at Fukushima occurred and as such, were anticipatory of the real-world events that occurred in the three accidents at Fukushima as evident from comparisons highlighted in the following The Fukushima accidents were all variants of either the long-term or short-term station blackout scenarios identified in the SOARCA Peach Bottom study.

In the SOARCA LTSBO, after returning to full RPV pressure with SRV’s cycling, one SRV is assumed to seize open [RCS BARRIER FAILURE] causing RPV depressurization and concurrent water level loss and core damage.

These comparisons highlight some of the common system responses modeled by the MELCOR code for the Peach Bottom station blackout analyses and consistently observed in the Fukushima real-world events.

Another difference observed between SOARCA Peach Bottom station blackout (SBO) analyses and the Fukushima accidents is with respect to containment failure mode and hydrogen behavior. The SOARCA analyses of Peach Bottom, a significantly larger reactor compared with the Fukushima reactors, consistently predicted drywell liner [CONTAINMENT] failure following vessel lower head failure and release of core material to the drywell cavity, caused by contact between core materials and the steel liner of the containment. This resulted in containment depressurization and release of hydrogen to the torus room at a low elevation in the reactor building.

These comparisons illustrate remarkable consistency in accident sequence progression and overall system response between MELCOR-SOARCA modeling and real-world observations

from Fukushima. Differences in the signatures are generally understood and due to differences in operator actions as well as better-than-expected durability of the RCIC turbine driven steam system in the Fukushima accidents. The modeled and observed differences in hydrogen release (i.e., drywell liner [CONTAINMENT] failure versus drywell head flange [CONTAINMENT FAILURE] leakage from over-pressurization) are apparently due to modeled differences in corium behavior in the cavity, perhaps attributable to the comparatively larger Peach Bottom core which may have a higher potential to flow and contact the steel liner [CONTAINMENT]. The real-world observations from Fukushima are consistent with phenomenology and system responses modeled by MELCOR, and give confidence to the overall findings in the SOARCA studies.

The purpose of SOARCA is to evaluate the consequences of postulated severe reactor accident scenarios that might cause a NPP to release radioactive material into the environment.

A detailed uncertainty analysis was performed for a single-accident scenario rather than all seven of the SOARCA scenarios documented in NUREG-1935 [1]. This work does not include uncertainty in the scenario frequency. The SOARCA Peach Bottom BWR Pilot Plant Unmitigated LTSBO scenario [2] is analyzed. While one scenario cannot provide a complete exploration of all possible effects of uncertainties in analyses for the two SOARCA pilot plants, it can be used to provide initial insights into the overall sensitivity of SOARCA results and conclusions to input uncertainty. In addition, since station blackouts (SBOs) are an important class of events for BWRs in general, the phenomenological insights gained on accident progression and radionuclide releases may prove useful for BWRs in general.

An accident sequence begins with the occurrence of an initiating event (e.g., a loss of offsite power, a loss-of-coolant accident (LOCA), or an earthquake) that perturbs the operation of the NPP. The initiating event challenges the plant's control and safety systems, whose failure might cause damage to the reactor fuel and result in the release of radioactive material. Because a NPP has numerous diverse and redundant safety systems, many different accident sequences are possible depending on the type of initiating event that occurs, which equipment subsequently fails, and the nature of the operator actions involved, as described in the SOARCA study [1, 2]. Individual accident sequences can be grouped into accident scenarios that represent functionally similar sequences. The SOARCA project analyzed a handful of important scenarios in detail. The scenario selection process for the SOARCA project is described in NUREG-1935 [1]. Three accident scenarios were chosen for analysis for Peach Bottom (the BWR pilot plant) and four accident scenarios were selected for Surry (the PWR pilot plant) [1].

The process for selecting a SOARCA scenario for this uncertainty analysis considered both the magnitude and timing of the offsite radionuclide release, which have major impacts on both early and latent cancer fatality risks. The examination of candidate scenarios considered both the timing of core damage and the timing of containment failure.

SBOs are an important class of events for NPPs, especially BWRs, which pointed to both Peach Bottom LTSBO and STSBO scenarios as good candidates. Although the uncertainty analysis was already under way by March 2011, the events at the Fukushima Daiichi plant re-confirmed the interest in SBOs for BWRs. The STSBO has a more prompt radiological release and a slightly larger release compared to LTSBO over the same interval of time.

A response to a LTSBO would begin with the onsite emergency response organization and would expand as needed to include utility corporate resources, State and local resources, and resources available from the Federal government, should these be necessary. It is most likely that plant personnel would attempt to mitigate the accident before core melt, but if their efforts were unsuccessful the national level response would provide resources to support mitigation of

the [LTSBO] source term [versus the much lower DBA LOCA source term of RG 1.183/DG-1389].

Source term release behavior in terms of the rate and total amount released in-vessel is strongly coupled to in-vessel melt progression behavior owing to the strong temperature dependence of fission product release. The onset of volatile fission product release is set by the time that fuel is heated to a temperature above about 1500 K (about 1227°C), and this is tightly coupled to cladding oxidation rate. Total release of both volatile and less volatile species is affected by the time at which fuel remains at elevated temperatures and the state of the fuel (rods or debris). Therefore, many of the parameters that affect [FUEL] cladding [BARRIER] oxidation and hydrogen generation also affect fission product release.

The parameters selected in the study were considered in terms of both melt progression and fission product release and transport. This includes important phenomena taking place following vessel lower head melt-through such as melt attack of the drywell liner [CONTAINMENT], containment behavior issues, such as uncertainty in onset of drywell head flange leakage CONTAINMENT FAILURE], and uncertainties in radioactive aerosol transport mechanics.

The dominant mechanism of containment failure in accident sequences involving the drywell floor, such as the LTSBO, is thermal failure (melting) of the drywell liner following contact with molten core debris (i.e., drywell liner melt-through). Containment failure by this mechanism occurs after debris is released from the reactor vessel lower head and flows out of the reactor pedestal onto the main drywell floor. If a sufficiently large quantity of debris accumulates in the pedestal, it can flow out of the pedestal through a large doorway in the concrete pedestal wall.

If the debris temperatures remain sufficiently high as it spreads across the drywell floor and contacts the drywell liner, the liner would melt and fail. The precise conditions under which core debris would flow out of the pedestal and across the drywell floor are uncertain. These uncertainties are adequately captured by assuming debris mobility and the potential for liner failure are represented by two key parameters: debris mass (i.e., static head) necessary for lateral flow and debris temperature (which characterizes debris rheological properties and internal energy available to challenge the liner).
If debris flows out of the reactor pedestal and spreads across the drywell floor, as described above, and contacts the outer wall of the drywell, the steel liner [CONTAINMENT] will fail. This failure opens a release pathway to the lower reactor building. Heat transfer between the steel liner and molten core debris is not explicitly calculated in the MELCOR model, due to limitations of the CAV Package, which addresses ex-vessel model debris behavior. The model assumes an opening in the drywell liner [CONTAINMENT FAILURE] occurs 15 minutes after debris first contacts the drywell wall. This time delay represents an average of estimates for failure time discussed in NUREG/CR-5423 [27] for situations in which the drywell floor is not covered with water.

An ignition source for hydrogen combustion in the reactor building is unclear during a SBO. Since there are no electrically energized components in the reactor building during a SBO, the most likely ignition source will be a hot surface. Default ignition parameters were used in the SOARCA calculations for NUREG/CR-7110 Volume I. However, the accumulation of hydrogen due to an absence of an electrical ignition source is credible. The ignition of hydrogen from a hot surface is caused by local heating of the hydrogen-oxygen mixture to a point where there is a sufficiently large volume of the mixture reaching the auto ignition.

The importance of zircaloy melt breakout temperature (SC1131-2) is explained by the effect this parameter has on oxidation. Larger breakout temperatures lead to greater oxidation. Greater oxidation leads to greater heat generation and earlier MSL rupture. Earlier MSL rupture allows more gaseous iodine to enter the drywell instead of being vented to the wetwell (through the stuck-open SRV) where it would be efficiently scrubbed in the wetwell pool. Once in the drywell,

the gaseous iodine is readily available to escape containment through the drywell head flange or a drywell liner melt-through.

When a MSL rupture occurs, containment over pressurizes and leaks past the drywell head flange. This results in an early release.

Whether a surge of water from the wetwell up onto the drywell floor occurs relates to amounts of cesium that deposit in the wetwell pool but fail to be confined there. In a large number of the realizations, a surge of water from the wetwell up onto the drywell floor occurs when the containment depressurizes in response to a breach developing in the drywell liner due to core debris contacting the liner and melting through it. The wetwell pool is saturated at the time and susceptible to flashing given a depressurization. The vacuum breakers between the wetwell and the drywell are overwhelmed and contaminated water from the wetwell surges up onto the drywell floor. Most of the water moves out the liner breach but some of it pools above the core debris on the drywell floor. The pool subsequently evaporates introducing its inventory of fission products to the atmosphere and structures of the drywell where they are available for release to the environment. (Note that the flow path representing the liner breach in the MELCOR model is a 6-cm high horizontal slot with its lowest point 0.41 m off the drywell floor.)

There is a correlation between the uncertainty in the drywell liner breach size and whether a surge of water from the wetwell occurs as evidenced in Figure 6.1-14. Larger sizes cause stronger containment depressurizations and hence larger potentials for water to surge from the wetwell.

From: Meighan, Sean
Sent: Tuesday, March 22, 2022 7:45 AM
To: Brian Magnuson
Cc: Wang, Weidong
Subject: RE: RE: RE: ACRS Subcommittee Meeting - Regulatory Guide 1.183 (DG-1389) "Alternative Radiological Source Terms for Evaluating Design Basis Accident at Nuclear Power Reactors"

Good morning Mr. Magnuson:

Thank you very much for your feedback. I am truly happy that you left the meeting with those impressions. The process for the presentation and the transcripts is to assemble a package of the presentation, transcripts, and public comments and place them together in ADAMS. We (obviously) have the presentation in hand, and expect to have the full package prepared and placed in ADMAS within a two week time frame.

As soon as I get word they are in ADAMS, I will send you an e-mail.

Very Respectfully

Sean Meighan
NRR/DRA/ARCB
Reactor Scientist.

From: Brian Magnuson <magnuson28@msn.com>
Sent: Monday, March 21, 2022 9:26 PM
To: Meighan, Sean <Sean.Meighan@nrc.gov>

Cc: Wang, Weidong <Weidong.Wang@nrc.gov>
Subject: [External_Sender] RE: RE: ACRS Subcommittee Meeting - Regulatory Guide 1.183 (DG-1389) "Alternative Radiological Source Terms for Evaluating Design Basis Accident at Nuclear Power Reactors"

Sean:

Wednesday's ACRS Subcommittee Meeting was informative and insightful.

I appreciate opportunity to provide verbal comments. Please let me know when the transcript is available.

Also, would you send me the meeting presentation.

I will email my post-meeting public comments by March 29.

Thank you,
Brian

From: Meighan, Sean
Sent: Monday, March 14, 2022 4:44 PM
To: 'magnuson28@msn.com'
Subject: FW: RE: ACRS Subcommittee Meeting - Regulatory Guide 1.183 (DG-1389) "Alternative Radiological Source Terms for Evaluating Design Basis Accident at Nuclear Power Reactors"

Good afternoon Mr. Magnuson:

I am working on compiling answers to your below questions. In order to get you some further answers before the ACRS meeting, would you be amenable to a call tomorrow? I have not been able to find your phone number in any of your correspondence. Or conversely feel free to call me at 301-287-9094.

There are preliminary answers below which will support your participation in the ACRS Subcommittee meeting Wednesday.

Very Respectfully

Sean Meighan
NRR/DRA/ARCB
Reactor Scientist.

From: Brian Magnuson <magnuson28@msn.com>
Sent: Friday, March 11, 2022 4:01 PM
To: Meighan, Sean <Sean.Meighan@nrc.gov>
Cc: Blumberg, Mark <Mark.Blumberg@nrc.gov>; Smith, Micheal <Micheal.Smith@nrc.gov>; Wang, Weidong <Weidong.Wang@nrc.gov>; Petti, David <David.Petti@nrc.gov>; Dickson, Elijah <Elijah.Dickson@nrc.gov>; Vasavada, Shilp <Shilp.Vasavada@nrc.gov>; Jones, Steve <Steve.Jones@nrc.gov>; Parillo, John <John.Parillo@nrc.gov>; Clifford, Paul <Paul.Clifford@nrc.gov>
Subject: [External_Sender] RE: ACRS Subcommittee Meeting - Regulatory Guide

1.183 (DG-1389) "Alternative Radiological Source Terms for Evaluating Design Basis Accident at Nuclear Power Reactors"

March 11, 2022

Sean/All:

Your invitation to attend another RG 1.183 public meeting is appreciated. -Thank you.

After reviewing the information you provided and documents from the prior meetings, I have a few questions and requests in preparation for this meeting.

I intend to make comments on March 16; however, because the time allotted for all public comments is limited to 15 minutes, I again intend to email my public comments before and after the meeting.

Is this still acceptable? Yes, please send your comments in an email to weidong.wang@nrc.gov

Please send me the ML# when my March 16, 2022 public meeting comments are posted in ADAMS. Yes we can.

DG-1389 was not provided for this meeting, and it is not listed on the NRC Regulatory Guide webpage with the other RG 1.183 draft guides. I eventually found a copy on the internet. It is designated as "*ACRS Public Version for November 19, 2021 Meeting*" with an issue date of "*Month 20xx*."

Is this the current revision of DG-1389? If not, please provide the current revision. Yes, this is the current version of the DG

Please provide information about the November 19, 2021 meeting. I assume it was not the November 19, 2020 public meeting. The November 19, 2021 ACRS meeting was cancelled.

I also discovered DG-1389 referenced in "*Guidance for Implementation of 10 CFR 50.59, "Changes, Tests and Experiments," at Non-power Production or Utilization Facilities (NRC-2021-0194)"* which is clearly unrelated to DG-1389. Nevertheless, this document states: "*The staff is also issuing for public comment a draft regulatory analysis (ADAMS Accession No. ML21243A104) for DG-1389."* This same statement was also posted in the Federal Register (Vol. 86, No. 223 / Tuesday, November 23, 2021 (66465)), which is odd because DG-2007 and 10 CFR 50.59 are clearly unrelated to DG-1389.

Please explain why these unrelated documents reference DG-1389.

Was DG-1389 actually issued for public comments as stated in the Federal Register? DG-1389, revision

to RG 1.183 has not been issued as of yet for public comment

Will the NRC afford me the opportunity to provide public comments on DG-1389? Yes.

While I appreciate the invitation and opportunity to attend another public meeting, I do not believe the NRC has been responsive to my prior RG 1.183/DG-1199 public comments. The NRC posted my public comments (and questions) in ADAMS, but has not provided documented responses. In effect, this negates my efforts and the intent of NRC public meetings.

Will the NRC provide documented responses my prior public comments (refer ADAMS and email inserted below)?

Will the NRC post your responses in ADAMS?

Thanks again.

Regards,
Brian

[Email inserted on March 11, 2022]

From: Brian Magnuson
Sent: Tuesday, December 22, 2020 9:33 AM
Subject: RE: Regulatory Guide 1.183 Revision Public Meeting - Magnuson Comments

Micheal:

-Thank you. Please keep me in the loop.

Happy holidays to you and your team.

Brian

From: Smith, Micheal
Sent: Monday, December 21, 2020 10:48 AM
To: Brian Magnuson
Cc: Blumberg, Mark; Meighan, Sean
Subject: RE: Regulatory Guide 1.183 Revision Public Meeting - Magnuson Comments

Brian,

The ML# for the November 19th public meeting summary is ML20342A198. The meeting summary refers to your emails (ML20343A064 (your first email) & ML20351A321 (your updated email)) in the public feedback section of the summary.

We look forward to your attendance during our next public meeting.

Happy Holidays!

Micheal Smith

Health Physicist and Assistant Radiation Safety Officer
Radiation Protection and Consequence Branch
Division of Risk Assessment
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
301-415-3763

From: Brian Magnuson <magnuson28@msn.com>
Sent: Tuesday, December 08, 2020 10:10 PM
To: Smith, Micheal <Micheal.Smith@nrc.gov>
Cc: Blumberg, Mark <Mark.Blumberg@nrc.gov>; Meighan, Sean <Sean.Meighan@nrc.gov>
Subject: [External_Sender] Regulatory Guide 1.183 Revision Public Meeting - Magnuson Comments

December 8, 2020

Micheal:

My RG 1.183 Public Meeting comments are inserted (**bold font**) in the NRC's presentation below.

Please send the ML# when they are placed in ADAMS.

Thank you,
Brian

RG 1.183 Public Meeting November 19, 2020 – Brian Magnuson Comments

The NRC staff has restarted efforts to revise RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

> **DG-1199 (Draft RG 1.183 Revision 1) was the first effort to revise RG 1.183. It was prompted by SAND2008-6601 and published by the NRC in 2009; however, it was never implemented. After eleven years, what prompted this effort?**

incorporate relevant operating experience as well as recent post-Fukushima seismic risk insights and walkdowns;

> **As important, are the accident source terms insights from Fukushima that were incorporated into RASCAL 4 (NUREG-1430, September 2012) source terms and methodologies. Will these insights be incorporated into RG 1.183-Revision 1?**
>
> **Why is the revision to RG 1.183 lagging behind revisions to RASCAL?**

ensure sufficient guidance is in place for licensing advanced light-water reactors (LWRs), accident tolerant fuel (ATF), high-burnup, and increased enrichment fuel; and,

> **NUREG-1465 (1995) "Accident Source Terms for Light-Water Nuclear Power Plants":**

"Recent information has indicated that high burnup fuel, that is, fuel irradiated at levels in excess of about 40 GWD/MTU, may be more prone to failure during design basis reactivity insertion accidents (RIA) than previously thought. Preliminary indications are that high burnup fuel also may be in a highly fragmented or powdered form, so that failure of the cladding could result in a significant fraction of the fuel itself being released."

The underlying concern identified here, is a cladding failure source term release could exceed that of a fuel melt source term release. What should be considered in RG 1.183-Revision 1, is the radiological consequences of a lessor and more likely accident may be worse than the "maximum credible accident" assumed in licensees' current licensing bases.

Reports and studies (e.g., Resolution of Generic Safety Issues: Issue 170: Fuel Damage Criteria for High Burnup Fuel (Rev. 2)) have evaluated high-burnup fuel and approved higher burn-up levels, but they have neither disputed the fuel disintegration caused by high-burnup nor evaluated the consequences of a powdered fuel source term. Until this NUREG-1465 concern has been eliminated, any revision to RG 1.183 should include a powdered fuel source term.

Limited range of applicability on Non-LOCA release fractions

Notably, DG-1199 significantly increased Non-LOCA nobel gas release fractions (above RG 1.183 Revision 0) and returned them to NUREG-1465 levels.

Excessive MISV leakage rates and realizations from the TMI accident prompted control room habitability studies and modifications to install Control Room Emergency Ventilation/Filter Systems. Subsequently, RG 1.183-Revision 0 required Control Room Operator doses to be evaluated for specific accidents, including the Non-LOCA fuel handling accident (FHA); however, missing from RG 1.183-Revision 0 is a requirement to evaluate doses to those workers/fuel handlers that would be in close proximity to this accident. Given the concerns identified in NRC Information Notice No. 90-08: "*KR-85 Hazards From Decayed Fuel*" and estimations based on FHA doses to control room operators, workers near spent fuel pools during would undoubtedly be overexposed (> 5 Rem TEDE).

Because no amount of water in spent fuel pools will not prevent the release of nobel gas (Kr-85, a pure beta emitter) in a FHA, revisions to RG 1.183 should require the calculation of spent fuel pool doses to ensure workers are aware of the hazards. This calculation could also be used to ensure the viability of FLEX actions to intended to mitigate an extended loss of spent fuel pool cooling.

DG-1199
In October 2009, the NRC issued for public comment DG-1199 as a proposed Rev. 1 of RG 1.183.
Staff received 150 public comments

The reasons for revision of RG 1.183 in DG-1199 were:

Providing additional guidance for modeling BWR MSIV leakage,

SAND2008-6601 determined RG 1.183 BWR MSIV leakage source terms and methodologies are "non-conservative and

conceptually in error." These conceptual errors (and others) should be corrected in any revision to RG 1.183.

2019 License Amendment Requests

In 2019, NRC received several AST LARs requesting increased MSIV leakage As a result, work on DG-1199 was postponed to allow NRC staff to incorporate lessons learned, from evaluation of the LARs, into the revised RG 1.183:

James A. FitzPatrick Amendment No. 338 for AST, July 21, 2020 (ML20140A070)
Quad Cities Nuclear Power Station, Units 1 & 2 – Amendment Nos. 281 and 277 to increase allowable MSIV leakage, June 26, 2020 (ML20150A328)
Nine Mile Point Nuclear Station, Unit 2 – Amendment No. 182 to change allowable MSIV leak rates, October 20, 2020 (ML20241A190)
Dresden Nuclear Power Station, Units 2 & 3 – Amendments Nos. 272 and 265 to increase allowable MSIV leakage, October 23, 2020 (ML20265A240)

Does the NRC mean say LARs from last year (2019) cause a 11-year delay? DG-1199 (RG 1.183 Revision 1 Draft) was published by the NRC in 2009. In consideration of "The NRC Approach to Open Government," please explain the 11-year delay.

SAND2008-6601 clearly explains/illustrates that RG 1.183 MSIV Leakage source terms and metrologies are "non-conservative and conceptually in error." Given this, why did the NRC approve the use of non-conservative and conceptually inaccurate guidance to increase MSIV leakage?

The intent of the NRC staff is for RG 1.183 Rev. 0 and Rev. 1 to co- exist

With known, fundamental errors in RG 1.183-Revision 0, why would the NRC allow it to co-exist?

The NRC's "RESULTS OF PERIODIC REVIEW OF REGULATORY GUIDE 1.183," dated June 25, 2018, states:

> ***"The known technical and regulatory issues are addressed in a draft revision to RG 1.183 issued for public comment (Draft Guide (DG)-1199, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors," published October 2009 (ADAMS Accession No. ML090960464)). The main technical issues are addressed in Regulatory Position (RP) 3.2, "Release Fractions," RP 5.3, "Meteorology Assumptions," and RP A-5, "Main Steam Isolation Value Leakage in Boiling Water Reactors.""***

DG-1199 was prompted by SAND2008-6601, which determined RG 1.183-Revision 0 source terms and methodologies are conceptually inaccurate. The intent of DG-1199 was to correct the fundamental errors in RG 1.183-Revision 0. Is this still the intent of RG 1.183-Revision 1?

RG 1.183 states:

> ***"The design basis accident source term is a fundamental assumption upon which a significant portion of the facility design is based."***

Considering the significance of the accident source term, why would the NRC continue to allow licensees to use RG 1.183-Revision 0? Is not negligent to allow licensees to base nuclear

power safety (systems) on conceptually inaccurate and non-conservative accident source terms?

Revised Fuel Handling Accident

Revisited the original studies forming the technical basis for the FHA and incorporate updated information.

Model improvements established from the current understanding of reactor fuel pin physics and iodine chemistry under the environmental conditions in which fuel handling operations are taking place.

Concluded that considerable margin exists regarding the scrubbing effects of iodine in the spent fuel or reactor pool and that the current staff DBA FHA fission product transport model can be refined while still maintaining conservatism.

Reference: Memo from RES to NRR, "Closeout to Research Assistance Request for Independent Review of Regulatory and Technical Basis for Revising the Design-basis Accident Fuel Handling Accident," November 23, 2019 (ML19270E335)

Prior to the accident at Three Mile Island (1979) and years afterward, control room operators were not protected by emergency air filtration systems. Operator doses from a DBA FHA (and other DBAs) were not publicly communicated because they exceeded General Design Criterion 19 limits (< 5 Rem whole body). After RG 1.183 was approved, the NRC required control room emergency filtration systems to be installed, and when their dose reduction factors were applied, operator doses were restored to within the new limits of 10 CFR 50.67 (< 5 Rem TEDE). Even still, today control room operator doses are often the most limiting regulatory dose.

While there may be margin regarding the iodine doses to control room operators, there is no margin regarding the Kr-85 doses in a DBA FHA. No amount of water in spent fuel pools will mitigate or prevent the release of Kr-85 in a FHA, and nobel gasses cannot be filtered. Consideration of "*KR-85 Hazards From Decayed Fuel*" (Information Notice No. 90-08) is conspicuously missing from RG 1.183-Revision 0. Any revision RG 1.183 should address IN 90-08 concerns and require that doses to fuel handlers/workers in the area of a FHA be calculated.

Over the last 10 years no applicant or licensee has adopted the methodology from SAND2008-6601, "Analysis of Main Steam Isolation Valve Leakage in Design Basis Accident Using MELCOR 1.8.6 and RADTRAD."

There have been no communications that applicants or licensees intend to adopt the SAND2008-6601 methodology.

SAND2008-6601 is a scientific study performed by Sandia National Laboratories on behalf of the NRC that clearly explains/illustrates that RG 1.183 BWR MSIV source terms and metrologies are "non-conservative and conceptually in error." It is the technical basis for the "*proposed DG-1199 MSIV modeling changes.*" Nuclear power plant owners (licensees) have not adopted SAND2008-6601 (and have resisted DG-1199) because it is unlikely that they can comply with 10 CFR 50.67 if accurate MSIV leakage models and source terms are used. Please refer to the following January 2010 letters.

January 6, 2010, Draft Regulatory Guide, DG-1199 - BWR Owners' Group Request for Supporting Documentation and Comment Period Extension (Docket ID NRC-2009-0453):

> ***We note from our review that substantive changes are being proposed to the modeling of MSIV leakage. Leakage through the steam line pathway currently represents a significant fraction of the postulated LOCA doses in the existing DBA analysis for BWRs, including plants that credit the alternate leakage pathway via the condenser. The proposed changes in DG-1199 would have the effect of increasing the source term concentration entering the steam line by up to 20 times that of the current Regulatory Guide 1.183 methodology and assumptions. In turn, this will significantly impact the LOCA dose analysis.***

January 20, 2010, Nuclear Energy Institute Comments on U.S. Nuclear Regulatory Commission Draft Regulatory Guide DG-1199, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" *(Federal Register* of October 14, 2009, 74 FR 52822).

"*It is unlikely that BWRs would commit to using it due to extreme penalties with regard to MSIV leakages* (Item 83)."

As stated in NRC's, March 22, 2010, "RESPONSE TO THE BOILING WATER REACTORS OWNER'S GROUP REQUEST TO EXTEND THE COMMENT PERIOD FOR DRAFT REGULATORY GUIDE – 1199":

> ***"By letter dated January 6, 2010, the Boiling Water Reactor Owner's Group (BWROG) requested an extension of the public comment period for Draft Regulatory Guide – 1199 (DG-1199), "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Agencywide Documents Access and Management Systems (ADAMS) Accession No. ML090960464, open from October 14, 2009, to January 13, 2010. The extension request stated that, in order to gain an understanding of the implications and potential consequences of the proposed revision, the BWROG will need to perform a detailed review of the Staff's research supporting the proposed changes to modeling of the main steam line isolation valve (MSIV) leakage."***
>
> ***"The Nuclear Regulatory Commission (NRC) staff has reviewed the stated basis for the request to extend the public comment period. Based upon this review, the staff has determined it will not extend the public comment period for the reasons discussed below."***
>
> ***"On October 9, 2010* [sic]*, the staff released the technical basis for the proposed DG-1199 MSIV modeling changes to the public in a Sandia National Laboratories Report, SAND2008-6601, "Analysis of Main Steam Isolation Valve Leakage in Design Basis Accidents Using MELCOR 1.8.6 and RADTRAD," ADAMS Accession No. ML083180196. On November 16, 2010* [sic]*, the staff held a full day public workshop that included a presentation on the proposed MSIV modeling changes, including an extensive discussion of the role of the supporting***

MELCOR work. Based on its review of the request by the BWROG, <u>the staff has determined that no substantive issues with the staff's research were identified</u> as the basis for extending the public comment period. Additionally, the staff believes that an extended period of time has been provided to provide comments on the proposed guidance."

Has the NRC disavowed SAND2008-6601?

If not, why has the NRC allowed licensees to use non-conservative and conceptually inaccurate MSIV leakage models and source terms for the past ten years?

If not, why would the NRC allow RG 1.183-Revision 0 to co-exist with RG 1.183-Revision 1?

The design basis accident source term is a fundamental assumption upon which a significant portion of every nuclear power plant design is based; therefore, RG 1.183-Revision 0 is, essentially, a generic safety issue.

The NRC's failure to act on this fundamental safety issue prompted PRM-50-122—10 CFR Part 2.802 request for rulemaking.

Additional Considerations
Consider revising footnote 7 which provides an incorrect method to convert thyroid dose to TEDE
Implies a back-of-the-envelope calculation appropriately converts between ICRP 2 and ICRP 26/30 dosimetry methodologies.
There is no simple methodology to convert between these two systems of dosimetry.
To correctly calculate the radiological dose consequences for design basis accidents the appropriate dose methodology (and DCFs) must be applied.

During the RG 1.183 public meeting on November 19, 2020, an industry member commented that the incorrect methods, described in RG 1.183, to calculate the radiological dose consequences, were used to assess Operability of structures, systems and components required by plant Technical Specifications.

Again, why would the NRC allow RG 1.183-Revision 0 to co-exist with RG 1.183-Revision 1?

From: Brian Magnuson
Sent: Saturday, December 5, 2020 3:58 PM
To: Smith, Micheal
Subject: RE: RE: Regulatory Guide 1.183 Revision Public Meeting Notice

Micheal:

I apologized for the late response.

The public meeting was informative. -Thank you.

Unfortunately, my attempts to make comments during the meeting failed for some reason. Because of this, I will revise my comments based on what I learned and resubmit them for ADAMS.

Regards,

Brian

From: Smith, Micheal
Sent: Friday, December 4, 2020 10:29 AM
To: Brian Magnuson
Subject: RE: RE: Regulatory Guide 1.183 Revision Public Meeting Notice

Brian,

I have not heard back from you so I did want to make you aware that I intend on placing your email below into ADAMS before the end of next week. I appreciate you taking the time to participate in our public meeting.

Enjoy your weekend!

Micheal Smith

Health Physicist and Assistant Radiation Safety Officer
Radiation Protection and Consequence Branch
Division of Risk Assessment
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
301-415-3763

From: Smith, Micheal
Sent: Thursday, November 19, 2020 4:21 PM
To: Brian Magnuson <magnuson28@msn.com>
Cc: Blumberg, Mark <Mark.Blumberg@nrc.gov>; Meighan, Sean <Sean.Meighan@nrc.gov>
Subject: RE: RE: Regulatory Guide 1.183 Revision Public Meeting Notice

Brian,

Thank you for taking the time to provide us with your questions and comments. As long as you are alright with it I plan on putting your email into ADAMS so that we can make sure we consider your questions and comments as we develop our draft guide. I will provide you the ML# once I have it.

If you have any additional questions just let us know.

Thanks,

Micheal Smith

Health Physicist and Assistant Radiation Safety Officer
Radiation Protection and Consequence Branch
Division of Risk Assessment
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
301-415-3763

From: Brian Magnuson <magnuson28@msn.com>
Sent: Thursday, November 19, 2020 3:25 PM
To: Smith, Micheal <Micheal.Smith@nrc.gov>
Cc: Blumberg, Mark <Mark.Blumberg@nrc.gov>; Meighan, Sean <Sean.Meighan@nrc.gov>
Subject: [External_Sender] RE: Regulatory Guide 1.183 Revision Public Meeting Notice

Micheal:

I have comments and questions.

From: Brian Magnuson
Sent: Thursday, November 19, 2020 11:55 AM
Subject: RE: Regulatory Guide 1.183 Revision Public Meeting Notice

Micheal:

I'm not sure how much time will be available today for comments; therefore, I have included some observations and questions regarding the presentation below.

Please review accordingly and let me know if you have any questions.

Thank you,
Brian

The NRC staff has restarted efforts to revise RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

> **DG-1199 (Draft RG 1.183 Revision 1) was approved (but not issued) by the NRC in 2010. After ten years, what prompted this effort?**

incorporate relevant operating experience as well as recent post-Fukushima seismic risk insights and walkdowns;

> **Insights from Fukushima were previously incorporated into RASCAL (NUREG-1430) source terms and methodologies. Will these same insights be incorporated into RG 1.183 Revision 1? Why is the revision to RG 1.183 lagging behind revisions to RASCAL? Also, please explain why RASCAL does not use RG 1.183 source terms and methodologies.**

ensure sufficient guidance is in place for licensing advanced light-water reactors (LWRs), accident tolerant fuel (ATF), high-burnup, and increased enrichment fuel; and,

NUREG-1465 (1995) "Accident Source Terms for Light-Water Nuclear Power Plants":

> ***"Recent information has indicated that high burnup fuel, that is, fuel irradiated at levels in excess of about 40 GWD/MTU, may be more prone to failure during design basis reactivity insertion accidents (RIA) than previously thought. Preliminary indications are that high burnup fuel also may be in a highly fragmented or powdered form, so that failure of the cladding could result in a significant fraction of the fuel itself being released."***

> **The underlying concern identified here is a cladding failure source term release could exceed that of a fuel melt source term release. What should be considered is, the radiological consequences of a lessor and more likely accident may be the new "maximum credible accident."**

Reports and studies (e.g., Resolution of Generic Safety Issues: Issue 170: Fuel Damage Criteria for High Burnup Fuel (Rev. 2)) have evaluated high-burnup fuel and approved higher burn-up levels, but they have neither disputed the fuel disintegration caused by high-burnup nor evaluated the consequences of a powdered fuel source term. Until this NUREG-1465 concern has been openly eliminated, any revision to RG 1.183 should include a powdered fuel source term.

Limited range of applicability on Non-LOCA release fractions

Notably, DG-1199 significantly increased Non-LOCA nobel gas release fractions (above RG 1.183 Revision 0) and returned them to NUREG-1465 levels.

Excessive MISV leakage rates and the TMI accident prompted control room habitability studies, regulation and modifications to install Control Room Emergency Ventilation/Filter Systems. Subsequently, RG 1.183 Revision 0 required Control Room Operator) doses to be evaluated for specific accidents, including the Non-LOCA fuel handling accident (FHA); however, missing from RG 1.183 is a requirement to evaluate doses to those fuel handlers/workers that would be in close proximity to this accident. Given the concerns identified the NRC identified in Information Notice No. 90-08: "*KR-85 Hazards From Decayed Fuel*" and the doses to control room the doses these ground zero workers could exceed federal limits and threaten their health and safety.

Because the water in spent fuel pools will not prevent the release of nobel gas (Kr-85, a pure beta emitter) in a FHA (mechanical damage or overheating), revisions to RG 1.183 should require the analysis of local doses to ensure the safety of workers in the area at the time of the accident. Additionally, the Non-LOCA FHA source term and methodologies should be used to ensure the viability of FLEX actions to intended to mitigate an extended loss of spent fuel pool cooling.

DG-1199
In October 2009, the NRC issued for public comment DG-1199 as a proposed Rev. 1 of RG 1.183.
Staff received 150 public comments

The reasons for revision of RG 1.183 in DG-1199 were:

Providing additional guidance for modeling BWR MSIV leakage,

SAND2008-6601 determined RG 1.183 BWR MSIV leakage source terms and methodologies are "non-conservative and conceptually in error." These conceptual errors (and others) should be corrected in any revision to RG 1.183.

2019 License Amendment Requests

In 2019, NRC received several AST LARs requesting increased MSIV leakage As a result, work on DG-1199 was postponed to allow NRC staff to incorporate lessons learned, from evaluation of the LARs, into the revised RG 1.183:

James A. FitzPatrick Amendment No. 338 for AST, July 21, 2020 (ML20140A070)
Quad Cities Nuclear Power Station, Units 1 & 2 – Amendment Nos. 281 and 277 to increase allowable MSIV leakage, June 26, 2020 (ML20150A328)
Nine Mile Point Nuclear Station, Unit 2 – Amendment No. 182 to change allowable MSIV leak rates, October 20, 2020 (ML20241A190)
Dresden Nuclear Power Station, Units 2 & 3 – Amendments Nos. 272 and 265 to increase allowable MSIV leakage, October 23, 2020 (ML20265A240)

Does the NRC mean say LARs from last year (2019) cause a 10-year delay? DG-1199 was approved (but not issued) by the NRC in 2010. In consideration of "The NRC Approach to Open Government," please explain the 10-year delay.

Because SAND2008-6601 clearly explains/illustrates that RG 1.183 MSIV Leakage source terms and metrologies are "non-conservative and conceptually in error," it does not seem that LARs to increase MSIV leakage are in the best interest of public health and safety.

The intent of the NRC staff is for RG 1.183 Rev. 0 and Rev. 1 to co- exist

According to RG 1.183, "T*he design basis accident source term is a fundamental assumption upon which a significant portion of the facility design is based."* Given this and SAND2008-6601, how does the existence (coexistence) and continued use of the non-conservative and conceptual errors in RG 1.183 benefit the health and safety of the public?

Revised Fuel Handling Accident

Revisited the original studies forming the technical basis for the FHA and incorporate updated information.

Model improvements established from the current understanding of reactor fuel pin physics and iodine chemistry under the environmental conditions in which fuel handling operations are taking place.

Concluded that considerable margin exists regarding the scrubbing effects of iodine in the spent fuel or reactor pool and that the current staff DBA FHA fission product transport model can be refined while still maintaining conservatism.

Reference: Memo from RES to NRR, "Closeout to Research Assistance Request for Independent Review of Regulatory and Technical Basis for Revising the Design-basis Accident Fuel Handling Accident," November 23, 2019 (ML19270E335)

While there may be margin regarding the scrubbing effects of iodine, there is no margin regarding the release of Kr-85 in a DBA FHA. Please consider DBA FHA doses to control room operators and extrapolate local area doses. No amount of water in spent fuel pools or the reactor pools, will shield or prevent the release of a nobel gas (Kr-85) in a DBA FHA (or other accidents that cause mechanical or overheating damage in these pools).

Consideration of "*KR-85 Hazards From Decayed Fuel*" (Information Notice No. 90-08) is conspicuously missing from RG 1.183 Revision 0. It should be included in any revision.

Over the last 10 years no applicant or licensee has adopted the methodology from SAND2008-6601, "Analysis of Main Steam Isolation Valve Leakage in Design Basis Accident Using MELCOR 1.8.6 and RADTRAD."

There have been no communications that applicants or licensees intend to adopt the SAND2008-6601 methodology.

SAND2008-6601 clearly explains/illustrates that RG 1.183 BWR MSIV source terms and metrologies are "non-conservative and conceptually in error." It identifies a safety concern (with a complex array of regulatory implications); however, this concern was not enough to motivate nuclear power plant owners/operators to adopt SAND2008-6601 or otherwise correct the non-conservative errors in RG 1.183—that adversely affect

the health and safety of the public. This is the crux of the matter and the reason for PRM-50-122.

From: Brian Magnuson
Sent: Wednesday, November 4, 2020 10:31 PM
To: Smith, Micheal
Cc: Blumberg, Mark; Meighan, Sean
Subject: Re: Regulatory Guide 1.183 Revision Public Meeting Notice

Micheal/Mark:

I appreciate the notification and plan to attend.

Thank you,
Brian

On Nov 4, 2020, at 10:33, Smith, Micheal <Micheal.Smith@nrc.gov> wrote:

Hello,

My name is Micheal Smith and I am currently the project lead for the revision of Regulatory Guide 1.183. Mark Blumberg (project technical lead) informed me that you might be interested in the revision of RG 1.183 so I am reaching out to inform you that we have a public meeting scheduled for November 19th from 1pm -4pm EST. The link to the public meeting notice is below.

https://www.nrc.gov/pmns/mtg?do=details&Code=20201297

Enjoy the rest of your week!

<image001.jpg>

Micheal Smith

Health Physicist and Assistant Radiation Safety Officer
Radiation Protection and Consequence Branch
Division of Risk Assessment
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
301-415-3763

[End of inserted email]

From: Meighan, Sean <Sean.Meighan@nrc.gov>
Sent: Friday, March 4, 2022 1:14:34 PM
To: Brian Magnuson <magnuson28@msn.com>
Subject: ACRS Subcommittee Meeting - Regulatory Guide 1.183 (DG-1389) "Alternative Radiological Source Terms for Evaluating Design Basis Accident at Nuclear Power Reactors"

Mr. Magnuson,

I hope you are doing well. Recognizing your continued interest on the revision of RG 1.183 I am forwarding you the public meeting announcement, and agenda (links below) for our next public meeting. The public meeting will be held March 16, 2022 from 0830 - 1200 EST. There is a Microsoft Teams link in the agenda. We look forward to your participation in the meeting. If you have any questions regarding the meeting logistics feel free to let me know.

Public Meeting Announcement:
https://www.nrc.gov/pmns/mtg?do=details&Code=20211604

Agenda
https://www.nrc.gov/docs/ML2205/ML22056A239.pdf

Very Respectfully

Sean Meighan

Radiation Protection and Consequence Branch
Division of Risk Assessment
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
301-287-9094