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 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 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 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 to be 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]
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> 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:
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> 1. 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.
>
> 2. 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.
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> 3. 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.

<u>DISCUSSION:</u>

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 <u>containment integrity</u> 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:

> "<u>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</u>. 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:
>
>> (a) Serious injury to individuals off-site should be avoided if an unlikely, but still credible, accident should occur;
>>
>> (b) Even if a more serious accident (not normally considered credible) should occur, the number of people killed should not be catastrophic;

> (c) 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

<u>The NRC is withdrawing RG 1.3 because it is outdated</u>. 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, <u>significant advances</u> 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 <u>design bases</u> 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 <u>performance-based</u> 10 CFR 50.67, regulates all U.S. commercial nuclear power plants. 10 CFR 50.67 is an alternative <u>voluntary regulation</u> 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 <u>events used to</u> <u>test the facility</u> 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 <u>containment failure</u> mode and hydrogen behavior. The SOARCA analyses of Peach Bottom, a significantly larger reactor compared with the Fukushima reactors, <u>consistently predicted drywell liner [CONTAINMENT] failure</u> 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., <u>drywell liner [CONTAINMENT] failure versus drywell head flange [CONTAINMENT FAILURE] leakage</u> 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, <u>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 genera</u>l.

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 <u>containment failure</u>.

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. <u>The STSBO has a more prompt radiological release and a slightly larger release compared to LTSBO over the same interval of time</u>.

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.