June 11, 2009

MEMORANDUM TO: James J. Shea
Senior Reactor Engineer
Engineering Review Branch 2
Division of License Renewal

FROM: Mark A. Cunningham, Director **/RA/**
Division of Risk Assessment

SUBJECT: RESPONSE TO A NON-CONCURRENCE ON DRAFT REGULATORY GUIDE DG-1199, "ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS"

On April 14, 2009, you outlined reasons for non-concurring on the subject Draft Guide (DG-1199) that provides methods for modeling the radiological consequences of design basis accidents. I appreciate that you have taken time to provide your concerns and to document your views. It is my understanding that you have previously raised these concerns to your immediate supervisor and that there have been numerous meetings and discussions on these issues. Your experience, views, and efforts to share these views during the development of DG-1199 are important to the NRC's mission to protect the health and safety of the public.

You have chosen Management Directive 10.158 (MD 10.158) entitled, "NRC Non-Concurrence Process," to provide your views and concerns. MD 10.158 directs a formal response to your non-concurrence submittal. Attachment 1 to this letter contains background information on DG-1199. Attachment 2 is a summary of your concerns and a response to these concerns. Attachment 3 is your original non-concurrence submittal. This letter responding to your non-concurrence submittal will be included as part of the concurrence package for DG-1199.

It is the policy of the Nuclear Regulatory Commission (NRC) to maintain a working environment that encourages employees to make known their best professional judgments even though they may differ from the prevailing staff view, disagree with a management decision or policy position, or take issue with a proposed or established agency practice involving technical, legal or policy issues. The NRC management values each staff member's view and encourages staff to express those views.

This response to your non-concurrence submittal concludes efforts to address your concerns through the non-concurrence process. At this time, I have decided not to make any changes to DG-1199 based on your comments. However, I have instructed the Accident Dose Branch Chief to keep a record of your concerns and to reevaluate whether any changes are needed to DG-1199 once public comments are received. If you feel that your concerns were not resolved in an appropriate manner, the NRC's differing professional opinion process is available to pursue your concerns. The differing professional opinion process is documented in Management Directive 10.159, "The NRC Differing Professional Opinions Program."

Enclosures:
As stated

Background on DG-1199

Introduction

In the early 1970s, the Nuclear Regulatory Commission (NRC) staff issued Regulatory Guides (RGs) 1.3 and 1.4 for evaluating the radiological consequences of design basis accidents (DBAs). RGs 1.3 and 1.4 use the radiological source term described in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

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."

10 CFR 50.67

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).

10 CFR 50.67 provides requirements on the acceptable dose limits from the design basis analyses and the assumption that the fission product release, assumed for these calculations, be based upon a major accident that is historically taken to involve a substantial core melt. The regulatory approach of using design basis accidents and applying performance based regulatory requirements is consistent with the approach provided in other NRC regulations such as 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and 10 CFR 50.65 "Requirements for monitoring the effectiveness of maintenance at nuclear power plants."

When 10 CFR 50.67 was codified, the NRC intentionally did not include any reference to NUREG-1465. This is consistent with the NRC regulatory philosophy and the staff's desire to allow changes to the defined source term or the development of other technically sound source term estimates without requiring additional rulemaking. Instead of codifying NUREG-1465 in 10 CFR 50.67, the NRC staff used NUREG-1465 and other technical information to develop RG 1.183, Revision 0, as one methodology acceptable to the staff for complying with 10 CFR 50.67. This has provided the NRC and nuclear industry with the flexibility to consider and incorporate new research and technical advancements without having to conduct rulemaking.

Regulatory Guide 1.183

In July 2000, the NRC staff issued RG 1.183, Revision 0 which provides one method acceptable to the NRC staff for complying with the regulatory requirements contained in 10 CFR 50.67. As with all RGs, RG 1.183 is not a regulatory requirement and, therefore, licensees may propose, and the NRC may approve, alternative methodologies which have a sound technical and scientific basis to demonstrate that the regulatory requirements of 10 CFR 50.67 are satisfied. RG 1.183 simply provides one set of acceptable assumptions and parameters that licensees can use to calculate postulated radiological doses for light-water reactor (LWR) DBAs.

Since the initial issuance of RG 1.183, the NRC staff and the commercial nuclear industry both have gained substantial experience with the implementation of 10 CFR 50.67 and RG 1.183. Based on this experience and on specific feedback and comments from licensees, the anticipation of licensing advanced LWRs, and new research, the NRC is proposing to update RG 1.183. Draft Regulatory Guide 1199 (DG-1199) is the proposed Revision 1 to RG 1.183. In accordance with the NRC's regulatory processes, the staff is proceeding with soliciting internal and external stakeholder feedback through appropriate mechanisms such as Advisory Committee for Reactor Safeguards reviews and a public comment period.

Boiling Water Reactor (BWR) Main Steam Line Leakage Pathway

Research was initiated to determine whether updates to the RG 1.183 BWR main steam isolation valve leakage (MSIV) modeling methodologies were warranted. A discussion of the MSIV pathways follows.

BWRs operate by boiling water in direct contact with the reactor fuel rods and passing this steam directly through the power turbines by means of large main steam lines. Because steam lines could provide a potential direct release pathway from the core to the environment, two quick closing safety-related main steam line isolation valves (MSIVs) were included in the original design. The MSIVs on each steam line isolate the containment boundary from the environment in the event of a core damage accident. This isolation accomplishes a critical safety function of mitigating the release of fission products.

Because MSIVs are not leak tight, acceptable leakage limits were established and incorporated into the plant design and technical specifications in accordance with 10 CFR 50.36, "Technical specifications." Licensees periodically test MSIV leakage in accordance with established surveillance requirements to ensure the leakage remains below these limits. RG 1.183, Revision 0 provides a methodology acceptable to the NRC staff for establishing acceptable MSIV leakage limits. Specifically, it provides guidance on the radioactivity that is assumed released to the steam lines as well as methods for reducing these releases by crediting holdup and deposition in the steam line piping and in the main condenser.

In 1998, the staff performed an assessment to estimate the deposition in steam lines for the first AST application. The methods used by the NRC staff in calculation AEB 98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term," (ADAMS ML011230531) have been used by many licensees to model MSIV leakage. However, in 2006, the Office of Nuclear Regulatory Regulation (RES) informed the Office of Nuclear Reactor Regulation (NRR) that the AEB 98-03 report contained some technical errors.

In response, NRR prepared a User Need and coordinated with RES to undertake additional research focused on MSIV leakage modeling. The NRC contracted with Sandia National Laboratories (SNL) to perform a reassessment of the methods in AEB 98-03 using state-of-the-art computer codes and modeling. The results of this reassessment are contained in SNL report SAND2008-6601, "Analysis of Main Steam Isolation Valve Leakage in Design Basis Accidents Using MELCOR 1.8.6 and RADTRAD." The SNL report provides a state-of-the-art assessment indicating that a revision to the MSIV leakage assumptions in RG 1.183 should be considered.

Main Steam Line Leakage Methodology Update

DG-1199 provides improved methods based upon the SNL report to calculate DBA doses from the MSIV leakage pathway. DG-1199 proposes methods to calculate: 1) the concentration of radioactivity used as the source for the MSIV leakage and 2) deposition of radioactivity in the steam line piping and condenser.

RG 1.183, Revision 0, like its Regulatory Guide 1.3 and 1.4 predecessors, assumed that the concentration of radioactivity used as the source of the MSIV leakage is approximated by the concentration of radioactivity in the containment following a postulated core melt. The radioactivity from the core melt is assumed to mix instantaneously and homogeneously throughout the free air volume of the containment building and reactor coolant system piping. This idealized view presumes that the radioactivity is released from the fuel, transported out of the vessel, and instantaneously and homogenously distributed within the drywell volume. This equilibrated drywell atmosphere is assumed to be the source of the flow through the leaking MSIVs. In reality, the radioactivity in the vessel steam dome is not instantaneously equilibrated with the drywell or containment volume (See Figure 1).

In order to examine more realistically the behavior of radioactivity in the steam dome and containment, SNL used the MELCOR code to make best estimate predictions of the radioactivity released, the transport behavior in the vessel and containment, and the resulting leakage to the environment through leaking MSIVs. SNL used the MELCOR code because it is internationally recognized as the state-of-the-art for modeling nuclear power plant severe accidents.

The SNL report demonstrated that radioactivity transport is very different from the assumptions currently in RG 1.183. The current RG 1.183 methodology underestimates the MSIV leakage source term during the first two hours of the DBA. RG 1.183 assumes that the vessel and containment atmosphere are in equilibrium before reflood and that the containment atmosphere can be used as the source of the MSIV leakage. The technical validity of this assumption was explicitly evaluated in the SNL research. The SNL report documented findings from MELCOR code runs which showed that the concentration of radioactivity in the steam dome may be substantially higher than in containment until reflood occurs. After reflood, the vessel activity is swept into the containment and over time the vessel and containment atmosphere equilibrate. Therefore, consistent with the SNL research, DG-1199 proposes to use the concentration of radioactivity in the steam dome before reflood, and thereafter use the concentration of radioactivity in the containment.

A second proposed update modifies the amount of radioactivity deposited in the steam line. The current methodology used by the NRC is contained in AEB 98-03 discussed above.

The SNL reports showed that the AEB 98-03 methodology non-conservatively overestimated the amount of deposition of radioactivity in the steam line. One factor that contributed to the differences include more realistic steam dome, steamline and condenser modeling in the SNL report.

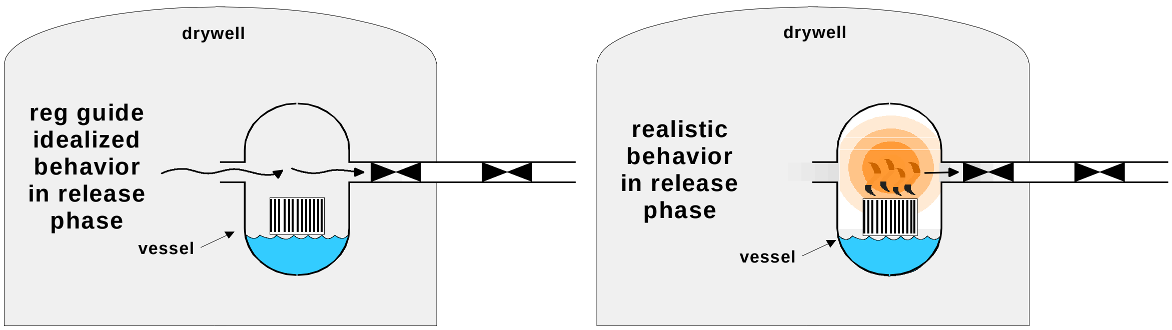

Figure 1   Idealized regulatory model of airborne fission products (left) compared to realistic prediction of airborne radioactivity (right) during release phase of a DBA with core damage. Note, in actuality the source of airborne activity emanates from vessel core (right)

Summary and Resolution of Non-concurrence Issues

Summary

Your non-concurrence submittal was reviewed and is summarized by the following issues:

Scrutinizing the proposed change using an "NRC Rule Making process" is required prior to issuing the draft regulatory guidance because it changes the accident "source term" characteristics and the application of the "source term" in design basis dose consequences analyses.

The assumptions used in the SNL analysis are not appropriate. The proposed change used a beyond design bases analysis by incorporating a MELCOR in-vessel "source term" that maximized radio-aerosol concentration coupled with deterministic assumptions to maximize dose consequence from boiling water reactors (BWR's) main steam lines (MSLs). You cited the specific issues given below:

- The proposed change used a beyond design bases analysis by incorporating a MELCOR in-vessel "source term" that maximized radio-aerosol concentration.

- Experience would show that for some or all of the first hour after a plant transient, flow may actually be into the vessel from residual steam from the turbine stop valves back to the steam dome rather than being biased out to maximize dose to the control room.

- The Sandia report clamped (normalized) the in-vessel concentration after the first hour of the accident instead of taking the analyzed best estimate data that showed the concentration in-vessel substantially declining after the first hour of the accident. This could substantially decrease the dose from the MSL pathway for the duration of the accident and may prove that the AST/TID containment "source term" would, in actuality, be more conservative over the total duration of a DBA LOCA for BWRs.

- In addition to these non-realistic biases used to develop the current guidance, the analysis did not model or assume that the BWR vessel separators and dryers would not reduce dose consequence by deposition. These assumptions were based on a PWR study that is not necessarily applicable to BWR designs.

Response to Non-Concurrence Issues

Non-Concurrence Issue

Scrutinizing the proposed change using an “NRC Rule Making process” is required prior to issuing the draft regulatory guidance because it changes the accident “source term” characteristics and the application of the “source term” in design basis dose consequences analyses.

Response

The proposed DG-1199 changes, related to the treatment of the MSIV leakage pathway source term, were scrutinized to determine if rulemaking would be required. The changes were determined to not conflict with the existing 10 CFR 50.67 rule or the 10 CFR 50.2 definition of source term.

As previously described, 10 CFR 50.67 is a performance-based regulation which allows licensees voluntarily to revise their licensing basis source term used to evaluate the consequences of applicable design basis accidents. 10 CFR 50.67 provides acceptance limits for design purposes and states:

The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of design analyses or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

10 CFR 50.2, “Definitions,” defines “source term” as:

Source term refers to the magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release.

The proposed change does not change the magnitude and mix of the radionuclides in or released from the fuel, expressed as fractions of the fission product inventory in the fuel, or their physical and chemical form or the timing of their release. Instead, the proposed change uses state-of-the-art research to model the realistic transport of fission products generated during a core melt accident instead of solely relying on the assumption that the MSIV leakage source term can be approximated by the concentration of radioactivity in the containment. The SNL MELCOR calculations were performed using methods similar to those used to create the NUREG-1465 source term.

In accordance with the principles of performance-based regulation, 10 CFR 50.2 and 10 CFR 50.67 do not specify any particular transportation pathway from the fuel to the reactor vessel steam dome or the drywell. Therefore, no regulatory conflict exists between the proposed changes in DG-1199 and the existing rule. Discussions with both the technical author and the Office of General Counsel reviewer for the 10 CFR 50.67 rule and Regulatory Guide 1.183 indicate agreement with this assessment.

DG-1199 will be formally scrutinized consistent with Office Instruction ADM-004, Revision 2, "Regulatory Guide Development, Revision, and Withdrawal Process," which is implemented by the Office of Nuclear Regulatory Research (RES). ADM-004 will guide the reviews and concurrences by 3 program offices (Office of Nuclear Reactor Regulation (NRR), Office of New Reactors (NRO), and RES). Additionally, NRR has specifically requested RES to request reviews of DG-1199 by the Committee to Review Generic Requirements (CRGR) and the Advisory Committee on Reactor Safeguards (ACRS) before publishing the DG-1199 for public comment.[1] Additionally, the Office of General Counsel's (OGCs) "No Legal Objection" determination will be required prior to issuing the DG for public comment. RES will transmit your reasons for the non-concurrence of DG-1199 and my response to these concerns to all three program offices, as part of the concurrence package for DG-1199.

Non-Concurrence Issue

The Sandia MELCOR analysis inappropriately combined some realistic analyses with deterministic assumptions that result in main steamline models that may not "realistically" or appropriately model the actual dose consequences from a BWR MSL in the event of a DBA LOCA.

Response

The SNL design basis analysis calculations for the main steam line models were structured to inform the staff's definition of an appropriately conservative set of assumptions to test the performance of the main steam line leakage pathway. The SNL researchers intentionally focused on those processes and phenomenon considered to most accurately predict and model this radionuclide transport pathway. For processes and phenomenon not explicitly modeled, the researchers selected assumptions and models that provide reasonable margin against unpredicted events in the course of an accident and to compensate for large uncertainties in facility parameters, accident progression and radioactive material transport.

It should be noted that the limits contained in 10 CFR 50.67 and RG 1.183 do not constitute acceptable limits for emergency doses to the public under accident conditions. Rather, these analyses calculate reference design doses used in the evaluation of proposed design basis changes to a nuclear power plant. They are not intended to model actual dose consequences and are meant to be intentionally conservative in order to address uncertainties in accident progression, fission product transport and atmospheric dispersion.

[1] Memo transmitting DG-1199 to the Office of Research from Mark Cunningham to Michael Case, entitled "Transmittal of Draft Regulatory Guide DG-1199," dated March 26, 2009 (ML090050330).

Non-Concurrence Issue

The proposed change used a beyond design bases analysis by incorporating a MELCOR in-vessel “source term” that maximized radio-aerosol concentration coupled with deterministic assumptions to maximize dose consequence from boiling water reactors (BWR's) main steam lines (MSLs).

Response

SNL did not maximize concentrations of radioactivity in the steam dome. SNL used state-of-the-art MELCOR models to calculate the realistic concentrations of radioactivity in the steam dome for several classes of radionuclides. The SNL staff abandoned the approach of using the highest concentration of all radionuclide classes because it significantly overestimated the concentration of radioactivity in the steam dome. Instead, the SNL analysis used the best estimate concentrations of iodine, cesium, and strontium to determine the ratio of radioactivity in the steam dome to the radioactivity in the containment. The state-of-the-art MELCOR code models a realistic sequence driven fission product release with consistent thermal-hydraulics aerosol mechanics and other physics-based models to predict best estimate source terms and transport behavior. These models, therefore, do not reflect a maximized in-vessel source term.

Non-Concurrence Issue

Experience would show that for some or all of the first hour after a plant transient, flow may actually be into the vessel from residual steam from the turbine stop valves back to the steam dome rather than being biased out to maximize dose to the control room.

Response

The MELCOR models actually predict and demonstrate for short periods of time that flow at the inboard MSIV would oscillate between downstream toward the condenser and upstream back into the reactor vessel due to the thermal-hydraulic conditions within the main steam lines. The predicted flow is dependent upon several parameters including the design of non-safety related main steam line piping, thermal contact with the containment, and the amount of insulation present. These parameters can vary between plants and were either not practical to model in the SNL analysis or were not modeled. The phenomenon is also dependent upon the accident scenario.

Given the many uncertainties of parameters that can impact the main steam line flow, flow outward was promoted in the calculations by reducing the heat transfer coefficients of the heat structures of the main steam line piping between the steam dome and the inboard MSIV for one hour. This modeling assumption did not stop the oscillations but used a conservative method to model the flow in light of uncertainties with the heat transfer coefficients, MSIV closure times and differences in BWR designs. This approach is consistent with other DBA models where it is either not possible or impractical to directly model each individual phenomenon or process.

Non-Concurrence Issue

The Sandia report clamped (normalized) the in-vessel concentration after the first hour of the accident instead of taking the analyzed best estimate data that showed the concentration in-vessel substantially declining after the first hour of the accident. This could substantially decrease the dose from the MSL pathway for the duration of the accident and may prove that the AST/TID containment “source term” would, in actuality, be more conservative over the total duration of a DBA LOCA for BWRs.

Response

A significant finding of the SNL work is that during the first two hours of a LOCA the concentration of radioactivity in the steam dome is significantly greater than that in the drywell. Since the steam lines are connected to the steam dome and not the containment, it is the concentration of radioactivity in the steam dome that should be used to determine the design bases for the MSIV leakage pathway.

Consistent with the historical modeling of the LOCA for 50.67, reflood is assumed to occur at two hours. Upon reflood, some of the activity in the vessel will be swept into the containment. Over time, the containment and steam dome are expected to come to an equilibrium concentration of radioactivity.

SNL analyzed MELCOR results to develop the MSIV leakage pathway methodology contained in DG-1199. The MSIV leakage pathway methodology includes a method for modeling the concentration of radioactivity available for release. SNL analyzed the concentrations of radioactivity in the steam dome and containment during the accident. Based upon this information, SNL determined that the concentration of radioactivity in the steam dome was comparable to that of the containment after the first hour and, therefore, it was appropriate to assume it is equal to the concentration in containment after the first hour of the accident. SNL used the following information to justify this assumption.

- The ratio of radioactivity in the steam dome to the radioactivity in the containment varies significantly over time. The steam dome-to-containment concentration ratios for Cesium and Iodine in the Mark-III recirculation line break case are somewhat greater than 1 during the period from 1 to 2 hours (the range is from 100 to 0.5). Despite the ratio of greater than one during the time period from 1 to 2 hours, the SNL methodology assumes the steam dome-to-containment concentration ratio is 1 for times greater than 1 hour. This underestimation of the concentration of radioactivity in the steam dome was done to compensate for times after reflood where the steam dome-to-containment concentration ratio might be less than one and to simplify the analyses licensees would have to perform in order to apply this research.

- Computationally, MELCOR cases that model reflood are very resource intensive and SNL could only one run one case. The one case that modeled core reflood modeled the scenario to 10 hours. Once computational and input issues were resolved, the time to run the case was greater than one week. Therefore, limited data exists for post reflood conditions. For the one SNL reflood case, the steam dome-to-containment concentration ratio varies after reflood, but within 2 hours approaches a value of one (approximately 0.4).

- The steam dome concentrations during the first hour of the accident contribute more to dose than the containment concentrations after reflood. Based upon Figure 2-44 of the SNL report, the impact of the pre-reflood concentration of radioactivity in the steam dome on dose is 200 times greater than the post-reflood concentration. The post-reflood drywell concentration contributes less than 0.5% to dose. Therefore, the most significant impact on dose is due to the concentrations before reflood.

Non-Concurrence Issue

In addition to these non-realistic biases used to develop the current guidance, the (SNL) analysis did not model or assumed that the BWR vessel separators and dryers would not reduce dose consequence by deposition.

These assumptions were based on a PWR study that is not necessarily applicable to BWR designs.

Response

The steam separator and dryer are modeled in the MELCOR deck. MELCOR predicted deposition on these surfaces and a corresponding reduction in the concentration of radioactivity available to be released from the steam dome.

Summary of Responses

In summary, the proposed changes have been thoroughly reviewed and are found to be consistent with existing NRC rules and practices. The NRC's internal processes, reviews and concurrences discussed above will formally confirm this assessment prior to issuing DG-1199 for public comments.

Jim Shea's Non-concurrence

The proposed Alternate Source Term (AST) Regulatory Guide (RG) 1.183 Revision 1 incorporates research that has not been appropriately processed by NRC "Rule Making" given the change to the accident "source term" characteristics and application in design basis accident (DBA) dose consequence analysis. In addition the proposed change used a beyond design bases analysis by incorporating a MELCOR in-vessel "source term" that maximized radio-aerosol concentration coupled with deterministic assumptions to maximize dose consequence from boiling water reactors (BWR's) main steam lines (MSLs). Finally, the MELCOR realistic / deterministic approach for DBA dose consequence in BWRs does not meet the NRC core values of Efficiency, Clarity and Reliability.

Accident Source Term Regulatory Basis

In Title 10 of the Code of Federal Regulations (10 CFR) Part 100 "Siting Criteria" The "source term" has been used as a bounding deterministic (non-realistic) release of radioactivity from the core to containment in order to design and test Engineered Safety Features (containment) for reactor plant siting requirements. For almost 50 years starting with The Technical Information Document (TID) 14844," Calculation of Distance Factors for Power and Test Reactror Sites;" a bounding deterministic containment "source term" has been applied in DBA dose consequence analysis for 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance." The TID states on page 12 that, "The objective of estimating the radioactive inventory within the outer containment barrier is to attain a starting point for calculating the potential radiological hazard in the surrounding environs. For people in the proximity of the reactor building, factors such as the physical nature of the material leaking from **the containment vessel**, release height, particle deposition with distance, wind direction, speed and variability, and air temperature gradients become important in determining the extent of these potential hazards. It is from this complexity of interwoven technical parameters that the values for the exclusion area, low population zone and population center distance must be determined."

Since the publication of TID-14844, significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases. In 1995, the NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." NUREG-1465 used updated research to provide more realistic estimates of the containment accident source term that were physically based and that could be applied to the design of future light-water power reactors. The NRC staff also determined that some current licensees may wish to use the NUREG-1465 source term referred to as the AST in analyses to support cost-beneficial licensing actions. The NRC staff, therefore, initiated several actions to provide a regulatory basis for operating reactors to use an AST in design basis analyses. These initiatives resulted in the development and issuance of 10 CFR 50.67, "Accident Source Term" and RG 1.183 (July, 2000)

In the NRC rule making for the AST the fundamental containment source term derived for the maximum credible accident was not changed from the TID concept. The change that provided licensees with some relief to the TID source term is principally due to the change in the timing

and chemical make-up of the containment “source term” which in the TID assumed that the release was instantaneous and that 50% of the radioiodine’s were available for release to the environs. The AST did not supplant these licensing bases fundamental accident source term assumptions but rather refined the bounding containment “source term” concept noting that the TID may be more conservative than realistically necessary for licensing purposes.

The licensing bases concepts of a deterministic bounding containment “source term” is imbedded in the foot-notes to 10 CRR 100.11 and 10 CFR 100.67 which states, “The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of design analyses or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible [maximum credible accident]. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

In the TID/AST deterministic approach the containment “source term” was conceived and applied in light water reactor licensing to encompass any possible accident by assuming a non-realistic radioactive release from the reactor core to the containment. This was intended to provide a bounding analysis for plant siting purposes.

<u>Purpose for Sandia Research</u>

The research done by Sandia originated from a NRR/DRA/AADB user need dated June 15, 2007 (No response returned as of the time of this writing) to confirm or determine if changes were needed to the NRC staffs radioactive aerosol settling and deposition methodologies used to asses the accident dose contribution from BWR MSL TS leakages. The NRC staff has been using a conservative application of AEB 98-03, “Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term” for determining radio-aerosol deposition on MSL’s for BWRs. In a staff briefing of the results of the Sandia findings based on the MELCOR modeling of the Peach Bottom (PB) plant, it was reported that the MELCOR analysis did confirm that the conservative application of AEB 98-03 was in line with findings from the MELCOR analysis when a containment “source term” is applied for MSL deposition. In addition the Sandia results showed that when the Condenser is credited for hold-up and deposition the dose contributor for the DBA LOCA is practically inconsequential.

<u>User Need Change in Focus</u>

At some point following the Research User Need request it was determined that a containment “source term” from TID/AST was not appropriate for BWR MSL analysis, this conclusion resulted in a new in-vessel “source term” that predicts for the first hour of a LOCA a significantly higher radio-nuclide concentration in-vessel than that found in the containment “source term.”

The application of the new Sandia MELCOR in vessel “source term” analysis combined with the already licensed AST/TID containment “source term” adds additional conservatisms to the licensing bases of BWRs that have not properly been scrutinized in an NRC Rule Making process.

Sandia MELCOR in-vessel "source term"

The MELCOR code could be a valuable tool for a PRA based safety and consequence analysis similar to what has been advocated by the International Atomic Energy Agency (IAEA) Safety Guide on Deterministic Safety Analysis for Nuclear Power Plants (DS 395). The IAEA draft Safety Guide describes the technological shift from a deterministic [AST/TID] safety analyses (past practices) which used rigorous conservative approaches to a more realistic approach (current preferred practice) together with an evaluation of uncertainties or a best estimate analysis.

In the RG update the MELCOR PB model used to develop the proposed dose consequence analysis procedure for BWR MSLs mixed the deterministic bounding core melt parameters with so called realistic core dynamics associated with a DBA LOCA. This modeling of the in–vessel "source term" generated by MELCOR was scaled up to meet the AST concentration values required for a full core melt in NUREG-1465. In addition the MSL flow was biased to ensure flow was always out of the MSL and not into the steam dome in the event of an actual DBA LOCA. Experience would show that for some or all of the first hour after a plant transient flow may actually be in-to the vessel from residual steam from the turbine stop valves back to the steam dome rather than being biased out to maximize dose to the control room. Also the Sandia report clamped (normalized) the in-vessel concentration after the first hour of the accident instead of taking the analyzed best estimate data that showed the concentration in-vessel substantially declining after the first hour of the accident. This could substantially decrease the dose from the MSL pathway for the duration of the accident and may prove that the AST/TID containment "source term" would, in actuality, be more conservative over the total duration of a DBA LOCA for BWRs. In addition to these non-realistic biases used to develop the current guidance, the analysis did not model or assumed that the BWR vessel separators and dryers would not reduce dose consequence by deposition. These assumptions were based on a PWR study that is not necessarily applicable to BWR designs.

It appears that the Sandia MELCOR analysis combined some realistic analysis with additional deterministic assumptions that result in a PB MSL model that may not "realistically" or appropriately model the actual dose consequence from a BWR MSL in the event of a DBA LOCA. During an inquiry of how this would affect the recently approved PB AST using the conservative application of AEB 98-03 for radio-aerosol deposition, I was told that the dose to the CR operator would increase a magnitude of approximately 6 times the currently approved value of close to the allowed limit of 5 rem TEDE.

NRC core values of Efficiency, Clarity and Reliability

The proposed changes to RG 1.183 as presented in revision 1 Appendix A-5 and the associated MELCOR model described in Reference A-10 creates a in-vessel accident "source term" that has no bases in current regulations. In our staff's review of the IAEA draft guidance discussed above AADB concluded that "Best-estimate safety analyses, as described and endorsed by the IAEA draft Safety Guide utilizing up to date computer modeling [such as MELCOR] and PRA techniques to show compliance with reactor siting and control room habitability regulations have not been done by US licensees. Moving from a conservative deterministic analysis in these areas may require revision of the current NRC regulations and regulatory guidance."

In addition to the lack of clear regulatory precedence for this new proposed in-vessel source term, the model presented in RG 1.183 rev 1Appendix A-5 for BWR MSLs used the best estimate code MELCOR as a tool to add un-realistic assumptions and requirements on the leakage path to maximize dose from a BWR MSL that is beyond design bases. In addition the research seemed to bias certain potential realistic parameters that may have provided better insight into the actual predicted dose consequence from this pathway in the event of a DBA LOCA.

The research also confirmed what we had originally asked in the AADB user need. It was shown that the NRC staff practice of a conservative application of the AEB deposition model was appropriate when used with the deterministic AST/TID containment “source term.” Additionally the research showed that when crediting the main condenser the dose consequence from BWR MSL’s is inconsequential.

Finally the result of the inclusion of a in-vessel “source term” for BWR’s would force a licensee who wanted to perform an AST analysis in accordance with 10 CFR 50.67, into a research type evaluation of the dose consequences for the MSL contribution to the DBA LOCA analysis. This analysis was already difficult under AEB 98-03, now a more expensive and laborious process has been created by this RG revision and by definition of its overly conservative beyond current licensing basis requirements. When applied to the recently approve PB AST the results show that most if not all currently approved BWR AST’s would fail to meet the CR dose acceptance criteria of 5 Rem TEDE. The proposed RG draft incorporating the Sandia MELCOR in-vessel accident “source term” for BWRs clearly does not meet the agencies goal of efficient, clear and reliable regulation.

Members of the DRA/AADB staff had presented management with an alternative to wholesale incorporation of the new BWR “in-vessel” source term developed by Sandia that was more appropriate to the current licensing structure. Questions were also raised as to why the division of NEW Reactors is not incorporating this research or using this Regulatory Guide for new-reactors as was originally planned. These questions as well as many questions that were sought concerning the details of the Sandia research including the effect on currently approved AST approvals, the affect on the few remaining non-AST BWR’s, and Back-fit concerns have not been addressed adequately to justify rushing this regulatory guide through the approval process.

James J. Shea
Senior Reactor Engineer on Detail in DLR

NRC FORM 757
NRC MD 10.158
(3-2009)

U.S. NUCLEAR REGULATORY COMMISSION

NON-CONCURRENCE PROCESS

SECTION A - TO BE COMPLETED BY NON-CONCURRING INDIVIDUAL

TITLE OF DOCUMENT	ADAMS ACCESSION NO.
DRAFT REGULATORY GUIDE DG-1199 (Proposed Revision 1 of RG 1.183)	**ML090570555**
DOCUMENT SPONSOR	SPONSOR PHONE NO.
Mark ~~Blumberg~~ Cunningham	**301-415-~~1083~~** 2884
NAME OF NON-CONCURRING INDIVIDUAL	PHONE NO.
James Shea	**301-415-1388**

☑ DOCUMENT AUTHOR ☑ DOCUMENT CONTRIBUTOR ☐ DOCUMENT REVIEWER ☐ ON CONCURRENCE

TITLE	ORGANIZATION
Senior Reactor Engineer	**NRR/DRA/AADB currently on detail in DLR**

REASONS FOR NON-CONCURRENCE

See Attached Word Document

☐ CONTINUED IN SECTION D

SIGNATURE

DATE 4/14/09

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NON CONCURRENCE PROCESS

The proposed Alternate Source Term (AST) Regulatory Guide (RG) 1.183 Revision 1 incorporates research that has not been appropriately processed by NRC "Rule Making" given the change to the accident "source term" characteristics and application in design basis accident (DBA) dose consequence analysis. In addition the proposed change used a beyond design bases analysis by incorporating a MELCOR in-vessel "source term" that maximized radio-aerosol concentration coupled with deterministic assumptions to maximize dose consequence from boiling water reactors (BWR's) main steam lines (MSLs). Finally, the MELCOR realistic / deterministic approach for DBA dose consequence in BWRs does not meet the NRC core values of Efficiency, Clarity and Reliability.

Accident Source Term Regulatory Basis

In Title 10 of the Code of Federal Regulations (10 CFR) Part 100 "Siting Criteria" The "source term" has been used as a bounding deterministic (non-realistic) release of radioactivity from the core to containment in order to design and test Engineered Safety Features (containment) for reactor plant siting requirements. For almost 50 years starting with The Technical Information Document (TID) 14844," Calculation of Distance Factors for Power and Test Reactror Sites;" a bounding deterministic containment "source term" has been applied in DBA dose consequence analysis for 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance." The TID states on page 12 that, "The objective of estimating the radioactive inventory within the outer containment barrier is to attain a starting point for calculating the potential radiological hazard in the surrounding environs. For people in the proximity of the reactor building, factors such as the physical nature of the material leaking from **the containment vessel**, release height, particle deposition with distance, wind direction, speed and variability, and air temperature gradients become important in determining the extent of these potential hazards. It is from this complexity of interwoven technical parameters that the values for the exclusion area, low population zone and population center distance must be determined."

Since the publication of TID-14844, significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases. In 1995, the NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." NUREG-1465 used updated research to provide more realistic estimates of the containment accident source term that were physically based and that could be applied to the design of future light-water power reactors. The NRC staff also determined that some current licensees may wish to use the NUREG-1465 source term referred to as the AST in analyses to support cost-beneficial licensing actions. The NRC staff, therefore, initiated several actions to provide a regulatory basis for operating reactors to use an AST in design basis analyses. These initiatives resulted in the development and issuance of 10 CFR 50.67, "Accident Source Term" and RG 1.183 (July, 2000)

In the NRC rule making for the AST the fundamental containment source term derived for the maximum credible accident was not changed from the TID concept. The change that provided licensees with some relief to the TID source term is principally due to the change in the timing and chemical make-up of the containment "source term" which in the TID assumed that the release was instantaneous and that 50% of the radioiodine's were available for release to the environs. The AST did not supplant these licensing bases fundamental accident source term assumptions but rather refined the bounding containment "source term" concept noting that the TID may be more conservative than realistically necessary for licensing purposes.

The licensing bases concepts of a deterministic bounding containment "source term" is imbedded in the foot-notes to 10 CRR 100.11 and 10 CFR 100.67 which states, "The fission product release assumed for these calculations should be based upon a major accident, hypothesized for

purposes of design analyses or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible [maximum credible accident]. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

In the TID/AST deterministic approach the containment "source term" was conceived and applied in light water reactor licensing to encompass any possible accident by assuming a non-realistic radioactive release from the reactor core to the containment. This was intended to provide a bounding analysis for plant siting purposes.

Purpose for Sandia Research

The research done by Sandia originated from a NRR/DRA/AADB user need dated June 15, 2007 (No response returned as of the time of this writing) to confirm or determine if changes were needed to the NRC staffs radioactive aerosol settling and deposition methodologies used to asses the accident dose contribution from BWR MSL TS leakages. The NRC staff has been using a conservative application of AEB 98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term" for determining radio-aerosol deposition on MSL's for BWRs. In a staff briefing of the results of the Sandia findings based on the MELCOR modeling of the Peach Bottom (PB) plant, it was reported that the MELCOR analysis did confirm that the conservative application of AEB 98-03 was in line with findings from the MELCOR analysis when a containment "source term" is applied for MSL deposition. In addition the Sandia results showed that when the Condenser is credited for hold-up and deposition the dose contributor for the DBA LOCA is practically inconsequential.

User Need Change in Focus

At some point following the Research User Need request it was determined that a containment "source term" from TID/AST was not appropriate for BWR MSL analysis, this conclustion resulted in a new in-vessel "source term" that predicts for the first hour of a LOCA a significantly higher radio-nuclide concentration in-vessel than that found in the containment "source term."

The application of the new Sandia MELCOR in vessel "source term" analysis combined with the already licensed AST/TID containment "source term" adds additional conservatisms to the licensing bases of BWRs that have not properly been scrutinized in an NRC Rule Making process.

Sandia MELCOR in-vessel "source term"

The MELCOR code could be a valuable tool for a PRA based safety and consequence analysis similar to what has been advocated by the International Atomic Energy Agency (IAEA) Safety Guide on Deterministic Safety Analysis for Nuclear Power Plants (DS 395). The IAEA draft Safety Guide describes the technological shift from a deterministic [AST/TID] safety analyses (past practices) which used rigorous conservative approaches to a more realistic approach (current preferred practice) together with an evaluation of uncertainties or a best estimate analysis.

In the RG update the MELCOR PB model used to develop the proposed dose consequence analysis procedure for BWR MSLs mixed the deterministic bounding core melt parameters with so called realistic core dynamics associated with a DBA LOCA. This modeling of the in–vessel "source term" generated by MELCOR was scaled up to meet the AST concentration values required for a full core melt in NUREG-1465. In addition the MSL flow was biased to ensure flow was always out of the MSL and not into the steam dome in the event of an actual DBA LOCA.

Experience would show that for some or all of the first hour after a plant transient flow may actually be in-to the vessel from residual steam from the turbine stop valves back to the steam dome rather than being biased out to maximize dose to the control room. Also the Sandia report clamped (normalized) the in-vessel concentration after the first hour of the accident instead of taking the analyzed best estimate data that showed the concentration in-vessel substantially declining after the first hour of the accident. This could substantially decrease the dose from the MSL pathway for the duration of the accident and may prove that the AST/TID containment "source term" would, in actuality, be more conservative over the total duration of a DBA LOCA for BWRs. In addition to these non-realistic biases used to develop the current guidance, the analysis did not model or assumed that the BWR vessel separators and dryers would not reduce dose consequence by deposition. These assumptions were based on a PWR study that is not necessarily applicable to BWR designs.

It appears that the Sandia MELCOR analysis combined some realistic analysis with additional deterministic assumptions that result in a PB MSL model that may not "realistically" or appropriately model the actual dose consequence from a BWR MSL in the event of a DBA LOCA. During an inquiry of how this would affect the recently approved PB AST using the conservative application of AEB 98-03 for radio-aerosol deposition, I was told that the dose to the CR operator would increase a magnitude of approximately 6 times the currently approved value of close to the allowed limit of 5 rem TEDE.

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"source term" for BWRs clearly does not meet the agencies goal of efficient, clear and reliable regulation.

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James J. Shea
Senior Reactor Engineer on Detail in DLR

NRC FORM 757
NRC MD 10.158
(3-2009)

U.S. NUCLEAR REGULATORY COMMISSION

NON-CONCURRENCE PROCESS

TITLE OF DOCUMENT
Draft Regulatory Guide DG-1199 (Proposed Revision 1 of RG 1.183)

ADAMS ACCESSION NO.
ML090570555

SECTION B - TO BE COMPLETED BY NON-CONCURRING INDIVIDUAL'S SUPERVISOR
(THIS SECTION SHOULD ONLY BE COMPLETED IF SUPERVISOR IS DIFFERENT THAN DOCUMENT SPONSOR.)

NAME
Robert Taylor

TITLE
Branch Chief

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ORGANIZATION
NRR/DRA/AADB

COMMENTS FOR THE DOCUMENT SPONSOR TO CONSIDER

- [ ] I HAVE NO COMMENTS
- [x] I HAVE THE FOLLOWING COMMENTS

As Mr. Shea's supervisor and the document sponsor, I greatly appreciate the contributions made by Mr. Shea during the development of DG-1199 and encourage him to continue to voice his concerns so that they may be considered. The concerns identified in his non-concurrence were raised during the development of the DG-1199 and numerous meetings were held to explore them. I solicited input on his technical and regulatory concerns from the subject matter experts in the Offices of Nuclear Regulatory Research, Nuclear Reactor Regulation, and General Counsel. In addition, his technical concerns were provided to Sandia National Laboratories to ensure that they had been appropriately explored and assessed during the development of the research. Mr. Shea was provided responses to these concerns during two distinct internal formal comment periods on the research and draft regulatory guide or provided an explanation during the meetings held on the research and draft guide.

After considering Mr. Shea's concerns, I determined that the appropriate next step was to follow NRC's regulatory processes related to the development of regulatory guides and seek additional stakeholder feedback. As we progress in the development and revision of DG-1199, input will be explicitly sought from the Committee to Review Generic Requirements, the Advisory Committee for Reactor Safeguards, and external stakeholders such as the public and nuclear industry. After receiving this additional stakeholder input, Mr. Shea's concerns will be reevaluated to determine what changes are appropriate to DG-1199.

Mr. Shea's non-concurrence was received after the Director of the Division of Risk Assessment signed out DG-1199. As such, it was not available for review by myself, the document signer and others on concurrence. Upon receipt of his non-concurrence, it was provided to my supervisor, the Director of the Division of Risk Assessment, for consideration. The attached document provides his review of Mr. Shea's non-concurrence.

- [ ] CONTINUED IN SECTION D

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6/11/09

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NON-CONCURRENCE PROCESS

TITLE OF DOCUMENT	ADAMS ACCESSION NO.
Draft Regulatory Guide DG-1199 (Proposed Revision 1 of RG 1.183)	**ML090570555**

SECTION C - TO BE COMPLETED BY DOCUMENT SPONSOR

NAME	
Mark Cunningham	
TITLE	PHONE NO.
Division Director	**301-415-2884**
ORGANIZATION	
NRR/DRA	

ACTIONS TAKEN TO ADDRESS NON-CONCURRENCE (This section should be revised, as necessary, to reflect the final outcome of the non-concurrence process, including a complete discussion of how individual concerns were addressed.)

Please see the attached response.

☐ CONTINUED IN SECTION D

SIGNATURE - DOCUMENT SPONSOR	DATE	SIGNATURE - DOCUMENT SIGNER	DATE
	6/11/09		6/11/0

NON-CONCURRING INDIVIDUAL *(To be completed by document sponsor* when process is complete, i.e., after document is signed):

☐ CONCURS
☒ NON-CONCURS
☐ WITHDRAWS NON-CONCURRENCE (i.e., discontinues process)

☒ WANTS NCP FORM PUBLIC
☐ WANTS NCP FORM NON-PUBLIC