May 31, 2020

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001
Attention: Rulemakings and Adjudications Staff

Subject: Petition to Amend 10 CFR 50.67, *Accident Source Term*, to Include Methodologies and Release Fractions

Pursuant to 10 CFR 2.802, please consider this petition for rulemaking.

The proposed rule would revise 10 CFR 50.67 Accident Source Term to codify the source term methodologies recommended in Sandia National Laboratories Report SAND2008-6601. Additionally, it would codify a modified version of Draft Regulatory Guide DG-1199 (Proposed Revision 1 of Regulatory Guide 1.183, dated July 2000) *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*—that would include the source term methodologies recommended in SAND2008-6601 and the corresponding Release Fractions. The codified version of DG-1199 would also account for the uncertainties that high burnup fuel pellets could fragment, relocate axially and possibly disperse outside of the fuel rod during postulated design-basis accidents.

Problem Background:

NUREG-1465: In 1962, the Atomic Energy Commission issued Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactors." In this document, a release of fission products from the core of a light-water reactor (LWR) into the <u>containment atmosphere</u> ("source term") was postulated for the purpose of calculating off-site doses in accordance with 10 CFR Part 100, "Reactor Site Criteria." The source term postulated an accident that resulted in substantial meltdown of the core, <u>and the fission products assumed released into the containment were based on an understanding at that time of fission product behavior</u>. In addition to site suitability, <u>the regulatory applications of this source term (in conjunction with the dose calculation methodology) affect the design of a wide range of plant systems</u>. [emphasis added]

<u>The use of postulated accidental releases of radioactive materials is deeply embedded in the regulatory policy and practices of the U.S. Nuclear Regulatory Commission (NRC)</u>. For over 30 years, the NRC's reactor site criteria in 10 CFR Part 100 (Ref. 1) have required, for licensing purposes, that an accidental fission product release resulting from "substantial meltdown" of the core into the containment be postulated to occur and that its potential radiological consequences be evaluated assuming that the containment remains intact but leaks at its maximum allowable leak rate. Radioactive material escaping from the <u>containment</u> is often referred to as the "radiological release to the environment." [emphasis added]

RG 1.183: *For the purpose of this analysis, the activity available for release via MSIV leakage should be assumed to be that activity determined to be in the drywell for evaluating containment leakage*. [emphasis added]

Problem Description:

Much of the past and present source term methodologies (including release fractions) used by nuclear power plants to perform accident dose calculations are inaccurate and nonconservative.

The current NRC guidance (Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000) used to calculate radiological release doses is "conceptually inaccurate" and "nonconservative." This is explained in Sandia National Laboratories report SAND2008-6601 (October 2008), which was conducted on behalf of the NRC. SAND2008-6601 also explains the conceptual errors in NUREG-1465.

Prompted by SAND2008-6601, the NRC drafted Revision 1 to RG 1.183 (DG-1199) in 2009, that increased release fractions, most of which are greater than NUREG-1465 levels. As of, June 25, 2018, the NRC still "concludes that a revision to RG 1.183 is warranted" (NRC Memorandum: The Results of Periodic Review of Regulatory Guide 1.183).

Despite these acknowledgements and the safety significance of accident source terms, the NRC has not yet approved DG-1199. Consequently, accident doses have been undercalculated for over twenty-five years.

DG-1199 states: *"The DBA* [design basis accident] *source term used for dose consequence analyses is a fundamental assumption upon which a significant portion of the facility design is based."* Given this understanding and the historical misconceptions of accident source terms, it appears nuclear power plants (facilities) were/are inadequately designed. They are currently operated with inaccurate dose consequence analyses.

Proposed Solution:

Amend 10 CFR 50.67, *Accident Source Term,* to codify the source term methodologies recommended in Sandia National Laboratories Report SAND2008-6601.

Amend 10 CFR 50.67 *Accident Source Term* to codify a version of Draft Regulatory Guide DG-1199 (Proposed Revision 1 of Regulatory Guide 1.183, dated July 2000) *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*—that is corrected to include the source term methodologies recommended in SAND2008-6601 and the corresponding release fractions. It would also account for the uncertainties that high burnup fuel pellets could be reduce to a powder form and disperse outside of the fuel rod during clad failure accidents (with or without fuel melt), by using the RASCAL calculation described in NUREG-1940:

$$I_{ACTUAL} = I_{38,585} \times (BURNUP_{ACTUAL} \div 38{,}585\ \text{MWd/MTU})$$

The specific source term methodologies recommended by SAND2008-6601 include:

> *6.1 Recommendations for the Application of Steam Dome-to-Drywell Concentration Ratios to RADTRAD Calculations*
>
> *MELCOR best estimate analyses of two DBA initiated accidents where significant core melting and fission product release has occurred have been explored to evaluate MSIV leakage behavior for two widely deployed BWR containment designs, Mk-I and Mk-III. These analyses have shown that, during the first two hours of such an accident, the airborne fission product aerosol concentrations in the reactor vessel significantly exceed those in the drywell. <u>Since the atmosphere in the reactor vessel supplies the effluent that ultimately leaks through the MSIV's, not the atmosphere in the drywell volume, these findings conclude that the current regulatory guidelines permitting the use of the fission product concentration in the drywell atmosphere during the first two hours prior to assumed vessel reflood is non-conservative for the purposes of evaluating the dose resulting from MSIV leakage, in addition to being conceptually inaccurate</u>. This study has investigated means of adapting the current regulatory containment source term for application to MSIV leakage analysis by means of scaling factors, accounting for differences in vessel fission product concentration and containment concentrations, and for differences in the NUREG-1465 derived containment concentrations compared to current best estimate derived containment concentrations. The developed scaling methodology preserves the simplified approach currently described in the regulatory guide by maintaining use of the AST; alternatively, detailed physics-based computer codes such as MELCOR, could be used to analyze source term release and transport. <u>Based on this work, it is recommended that the NUREG-1465 drywell fission product concentrations be scaled based on the time-phased scaling factors presented in Table 5-1 and Table 5-2 for application to Mk-I and Mk-III containments when determining the fission product concentration that is the source for MSIV leakage</u>. These tables correspond to the Powers 10% aerosol removal by natural processes model for BWR drywells. The scaling ratios are approximately 20% higher when using the Powers 50% model. The differences in scaling factors for Mk-I and Mk-III are predominantly a result of the differences in the drywell volumes alone. Therefore, extension of these recommendations to Mk-II or containments with different volumes can be justified by scaling the steam dome to drywell fission product concentration ratios by drywell volume (if the drywell volume is larger, then a larger scaling factor is required to obtain correct vessel concentration). This is a generic recommendation concerning the appropriate airborne concentration for evaluating MSIV leakage. A methodology for accomplishing this recommendation using RADTRAD has been demonstrated, using RADTRAD techniques such as superposition to accommodate the time-phased behavior of the releases; however, other codebased methods could be used to accomplish the same objective.* [emphasis added]
>
> *6.2 Recommendations for Sprays in RADTRAD Calculations*

The MELCOR Full Reactor model results show that the activation of drywell sprays reduces the fission product release to the environment by reducing the drywell pressure. This pressure reduction decreases the MSIV leakage flow rates thus reducing fission product release. It also increases the flow rate between the steam dome which moves additional fission products out of the steam dome into the drywell. This increase did not, however, appreciably reduce the fission product concentration observed in the steam dome for approximately 1 hour in the cases that were studied. While drywell sprays can clearly reduce containment leakage, and indirectly reduce MSIV leakage by lowering drywell pressure, <u>it is recommended that sprays not be credited for any reduction of the airborne concentration in the vessel supplying MSIV leakage during the first two hours, since, as discussed earlier, it is the vessel, not the drywell, that is the source of fission product concentration available for MSIV leakage.</u> Following a presumed recovery by vessel reflooding at two hours, drywell sprays can be credited for reducing the concentration of containment aerosols flowing back into the vessel and to the MSIV regions. Additionally, it would seem reasonable to allow credit for the pressure reduction resulting from the use of containment sprays, thereby reducing the MSIV pressure difference and the flow rate through the valves, provided adequate engineering analysis of containment response to sprays and valve leakage as a function of pressure drop is performed. [emphasis added]

6.3 Recommendations for Removal Coefficients in RADTRAD Calculations

Removal coefficients were calculated for the in-board segments of the MSLs. It is recommended, however, that no credit be taken for aerosol deposition in this portion of the MSLs. The basis for this recommendation is that at times in the simulation the temperature of portions of the in-board MSL piping are predicted to be high enough to vaporize fission products that had been previously deposited. This secondary source cannot easily be incorporated into a RADTRAD model, and if the initial deposition (via a non-zero removal coefficient) is credited, the omission of this secondary source would result in an under-prediction of fission products released downstream, and ultimately to the environment. There have also been questions raised regarding decay heat from fission products deposited in the in-board piping that could cause natural convection-driven bi-directional flow between the steam dome and the in-board MSLs. While the well-mixed nature of the MSL control volumes does, in some fashion, capture the enhanced mixing that such flow would cause, it does not address the potential for enhanced bulk transport of fission products from the steam dome into the in-board MSLs. Note that this issue has been previously cited as a basis for not taking credit for aerosol deposition in the in-board MSLs – see page 4-15 of reference [3]. Recommended removal coefficients based on the MSL-only model uncertainty results are given in the following tables, repeated for convenience from Section 4.

6.4 Recommendations for Post-Reflood Conditions in RADTRAD Calculations

The results from the Mk-III RLB case in which core sprays were activated 10 minutes before lower head failure indicate that the re-introduction of water into the core generates sufficient steam such that the vast majority of fission products in the steam dome and, to a lesser extent, in the drywell are swept into the wetwell, which effectively prevents them from being available for release to the environment via MSIV leakage. This same behavior is also seen in the Mk-I results at the time of lower core support plate failure. Based on this observation, we recommend that the scaling factors be set to unity following the first two hour period where scaling factors are used to adjust the AST-based containment concentrations to reflect vessel concentrations. This conservatively assumes that the drywell and steam dome environments are well mixed after vessel reflood takes place.

Deposition properties after reflood should be based on the characteristics of the containment aerosol (i.e size effects). In the single analysis that explored post-reflood conditions it was observed that the intermediate pipe deposition lambdas decreased over time following reflood owing to the change in size distribution brought about as the larger particles fall out. These depletion trends were not significantly different from the 5% results reported in Table 6-1 (~1 hr1). More analyses may be required to summarize the physical effect of decreasing lambda with decreasing particle mean diameter; however, the trends reported in this report serve as a minimal basis for recommending smaller lambdas in the hours and days following reflood.

6.5 Recommendation for the Influence of Flow Rates on MSL and Condenser Removal Coefficients

The results from the MELCOR MSL RLB flow uncertainty case show that there is some relationship between flow rates in the MSLs and the removal coefficient. However, this relationship is also highly dependent on the point in time of the accident progression. This is due to the influence of the aerosol particle size distribution on the removal coefficient. The current results do not support the development of a quantitative relationship between MSL flow and MSL or condenser removal coefficients.

6.6 Recommendation Regarding the Use of Effective Filter Efficiencies for RADTRAD MSL Modeling

In the RADTRAD code the user has the option of specifying either removal coefficients to volumes or filter efficiencies to the flowpaths that connect the volumes. While these two treatments have been deemed equivalent [NRC 1998], this is only true in the specific case where the nuclide storage term is zero (i.e., steady-state conditions). The two options in RADTRAD for treating deposition are both empirical, and each represents very different fundamental views of physics. While a particular filter efficiency can be selected that in the end reflects the degree of holdup and removal of aerosols in a flow path, it

cannot mechanistically represent the operative physics. To explore this further a small test problem, with geometry and conditions representative of MSIV leakage flow through steamlines, was investigated as described in the following. A small MELCOR test problem was developed to evaluate the error in using filter efficiencies rather than removal coefficients under transient conditions. As shown in Figure 7-1, the problem consisted of four volumes: (1) a constant-pressure volume which was set to drive a constant 108 SCFH through down-stream flowpaths and volumes; (2) a volume [high-pressure] in which a single RN aerosol class was introduced at a constant continuous rate (2.05e-7 kg/s), this volume also contains a floor heat structure which provides a deposition location for gravitationally settled aerosol particles; (3) a volume with floor heat structure; and (4) and a constant-pressure volume [environment] which was set to drive a constant 108 SCFH through up-stream flowpaths and volumes.

The test problem was run out to the time at which a steady-state concentration was reached (i.e., the slope of the curve of integrated RN mass in the environment is constant). Removal coefficients (lambda) and effective filter efficiencies (Feff) were calculated for the problem per the equations in the figure. An identical test problem model was then built for RADTRAD. That model was then run for two cases: (1) using the removal coefficients for the volumes; and (2) using the effective filter efficiencies for the flowpaths. The results were then compared to the MELCOR-predicted RN release to the environment. As shown in Figure 7-2, the case using the filter efficiency under-predicted the RN release (compare the blue curve with the black curve), while the case using the removal coefficients predicted an identical release to the MELCOR test problem result (compare the red dashed curve with the black curve). Based on these results, it is recommended that deposition in MSL pipes only be modeled in RADTRAD using removal coefficients and that the optional modeling method of using effective filter efficiencies applied to flowpaths not be used unless an actual filtering process is being represented.

6.7 Recommendation Regarding Calculation of MSIV Leakage Flow

It is recommended that analysis of valve leakage flow be generally based on the flow predicted using nozzle-flow theory as described in many fundamental text books such as Bird, Stewart and Lightfoot [11]. <u>The flow equations described earlier in section 5.4, accommodate critical or subsonic flow and serve as a defensible means of analyzing both test leakage behavior and the scaling of these measured flows to the flows that would be expected under accident conditions. The temperature and pressure expected upstream of each MSIV should be used in the determination of flow through that valve</u>. The determination of bounding (conservative) values for the pressures and temperatures upstream of the MSIVs were not part of this study. Therefore, the temperatures and pressures used in the example are solely for the purposes of demonstrating the methodology and should not be taken as recommended values for MSIV leakage or as representative of the conditions upstream of the MSIVs. [emphasis added]

BASIS for RULEMAKING

The current source term guidance is inadequate and ineffective. *"Regulatory guides* [and NUREGs] *are not substitutes for regulations and compliance with them is not required."* As such, there are no definitive source term requirements. Nuclear power plants currently use different regulatory guidance (e.g., TID-14844, NUREG-1465, or RG 1.183)—that use different source term methodologies/release fractions—to satisfy the same regulations. Even the NRC uses different guidance (NUREG-1940). Consequently, the guidance used by the NRC to calculate *accident doses* is different from the guidance used by nuclear plants to comply with *regulations*. The purposed rulemaking will eliminate these inconsistencies and provide the requisite means to ensure compliance with the underlying regulations.

As stated in RG 1.183,

> *There are several regulatory requirements for which compliance is demonstrated, in part, by the evaluation of the radiological consequences of design basis accidents. These requirements include, but are not limited to, the following.*
>
> - *Environmental Qualification of Equipment (10 CFR 50.49)*
> - *Control Room Habitability (GDC-19 of Appendix A to 10 CFR Part 50)*
> - *Emergency Response Facility Habitability (Paragraph IV.E.8 of Appendix E to 10 CFR Part 50)*
> - *Environmental Reports (10 CFR Part 51)*
> - *Facility Siting (10 CFR 100.11)*

Furthermore,

> *There may be additional applications of the accident source term identified in the technical specification bases and in various licensee commitments. These include, but are not limited to, the following from Reference 2, NUREG-0737:*
>
> - *Post-Accident Access Shielding (NUREG-0737, II.B.2)*
> - *Post-Accident Sampling Capability (NUREG-0737, II.B.3)*
> - *Accident Monitoring Instrumentation (NUREG-0737, II.F.1)*
> - *Leakage Control (NUREG-0737, III.D.1.1)*
> - *Emergency Response Facilities (NUREG-0737, III.A.1.2)*
> - *Control Room Habitability (NUREG-0737, III.D.3.4)*

Because there is no accurate source term guidance, it is highly likely that many nuclear power plants are NOT in compliance with some, or all of these, regulations and requirements—that ultimately protect people and the environment. The purposed rulemaking would require nuclear power plants to perform complete recalculations of all facility radiological analyses. These recalculations would then be used to verify compliance, or otherwise eliminate the undue risks of radiological releases that currently exists.

PETITIONER:

Brian D. Magnuson

magnuson28@msn.com

1020 Station Blvd. #212

Aurora, IL 60504

Lead Emergency Management Specialist—Exelon Corporation (Warrenville, IL)

Former NRC Licensed Senior Reactor Operator/Operations Shift Manager

—Acting expressly as a member of the public.

REFERENCES:

1. SECY-98-154, RESULTS OF THE REVISED (NUREG-1465) SOURCE TERM REBASELINING FOR OPERATING REACTORS (June 30, 1998)

   The original source term, which was based on releases from a severely damaged core, was published in 1962 by the U.S. Atomic Energy Commission in Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactors." Since that time there have been significant advances in our understanding of the timing, magnitude and chemical forms of the fission product release from severe reactor accidents. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," was published in February 1995, and reflects that extensive research and experience culminating in the development of a new or revised source term.

   The impetus for operating reactors to adopt the revised source term is that through its more realistic characterization of the source term, plants may modify existing restrictive plant features, (e.g., component actuation times, leakage control systems). [emphasis added]

   In an SRM dated February 12, 1997, the Commission approved the staff's plan outlined in SECY-96-242 and directed the staff to commence rulemaking upon completion of the source term rebaselining and concurrent with the pilot plant evaluations. The Commission further stipulated that implementation of the NUREG-1465 source term at operating reactors should include the revised Part 20 dose methodology (total effective dose equivalent or TEDE criterion) and should include consideration of the dose over the worst two hour interval after the accident. [emphasis added]

   It has been recognized since the development of the revised source term that changes in the prescription of the source term from that originally described in TID-14844 would influence the major areas of dose analysis and could prompt plant, technical specification and procedure modifications. [emphasis added]

The most significant changes in the source term are the treatment of the fission product release as a time dependent process and the release of radioiodine primarily as an aerosol. In the revised source term the fission product release is assumed to occur over roughly two hours as opposed to the TID source term which assumed the release of the entire source term occurs instantaneously at time zero. In addition, in the revised source term, 95% of the radioiodine is assumed to be released as an aerosol, CsI, with the remaining 5% as a combination of inorganic and organic vapors. This is in contrast to the original TID source term which prescribed the opposite ratio, 95% of the iodine as vapor and 5% as aerosol. Therefore, plant systems originally provided to mitigate an instantaneous source term by very rapid actuation would not be required to perform under such stringent requirements with the revised source term. Likewise, systems needed to remove iodine vapors are less important under conditions where iodine is an aerosol. [emphasis added]

As part of rebaselining these issues and a number of other more subtle differences between the source terms, as well as the impact of improved modeling of fission product processes, were explored to position the staff for review of pilot plants and rulemaking.

The overall impact of implementing the revised source term in the majority of cases is to produce lower calculated doses, ranging from a slight reduction up to an order of magnitude decrease, for an individual, whether for the EAB, LPZ or control room. In addition, in assessing the impact of implementing the revised source term and comparing new calculated doses against earlier (and occasionally much older) analyses, it was confirmed that changes in the calculated dose may also occur for reasons not directly related to the source term itself. For example, in older calculations, the dose to an individual was calculated using dose conversion factors taken from International Commission on Radiation Protection (ICRP) Publication 2. Current analyses including those implementing the revised source term would use updated dose conversion factors taken from EPA Federal Guidance Reports (FGR) 11 and 12. The use of updated dose conversion factors alone will produce reductions in the dose by up to 40%.

The extent of further reduction in calculated doses is also influenced by several factors, two of which are connected to differences between the TID and revised source term. As noted previously, the revised source term treats the release of fission products as a time dependent release, thus reduction of doses will be strongly influenced by safety features which are timing sensitive.

Treatment of iodine primarily as an aerosol in the revised source term resulted in a substantial reduction of the dose.

The TID source term is an instantaneous release of noble gases and iodine, which is assumed to be predominantly in elemental vapor form. For equipment qualification, additional nuclides of lower volatile species, in aerosol form, are assumed. The existing regulatory framework for operating reactors prescribes the calculation of offsite doses at the exclusion area boundary (EAB) for the first two hours after the accident and at the low population zone (LPZ) boundary for the course of the accident, which for dose analysis is assumed to continue for 30 days. Control room doses are also calculated for a 30 day period. Although the existing requirements involve the calculation of whole body and thyroid doses, in practice the thyroid dose, resulting from the iodine, has generally been the limiting dose for plants. [emphasis added]

The revised source term is markedly different in that there is a more mechanistic and realistic treatment of the rate of release of fission products and their chemical and physical form. The revised source term is a time dependent release of fission products with nuclides, other than noble gases, primarily in aerosol form. [emphasis added]

In conjunction with the implementation of the revised source term, the staff is considering new accompanying regulatory criteria addressing dose acceptance limits. In place of dose acceptance limits of 300 rem thyroid and 25 rem whole body, the staff has proposed to use the total effective dose equivalent (TEDE) methodology in 10 CFR Part 20 with a limit of 25 rem TEDE for offsite doses and 5 rem TEDE for control room doses. Implementation of the revised source term will also be accompanied by the related requirement that doses for the EAB be calculated over the worst two hour period. [emphasis added]

Equipment Qualification Dose Conclusions: For airborne gamma and beta doses, the doses with the TID source term were about the same as those with the NUREG-1465 source term, because only noble gases and iodine were assumed to be airborne and the magnitudes of the noble gas and iodine releases of the source terms are similar. For the sump, the gamma doses are higher at later times for the NUREG-1465 source term, because of the large amount of cesium in the NUREG-1465 source term. The TID source term included 1% of the core inventory of cesium, while the NUREG-1465 source term includes 30% of the cesium.

2. REGULATORY GUIDE 1.183 (Draft was issued as DG-1081) ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS (July 2000)

   Since the publication of TID-14844 (Ref. 1), significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents. In 1995, the NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (Ref. 5). NUREG-1465 used this research to provide estimates of the accident source term that were more physically based and that could be applied to the design of future light-water power reactors. NUREG-1465 presents a representative accident source term for a boiling-water reactor (BWR) and for a pressurized-water reactor (PWR). These source terms are characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the *release to the containment. The NRC staff considered the applicability of the revised source terms to operating reactors and determined that the current analytical approach based on the TID-14844 source term would continue to be adequate to protect public health and safety. Operating reactors licensed under that approach would not be required to re-analyze accidents using the revised source terms. The NRC staff also determined that some licensees might wish to use an AST in analyses to support cost-beneficial licensing actions.* [emphasis added]

The prior practice of deterministically assuming that a 50% plateout of iodine is released from the fuel is no longer acceptable to the NRC staff as it is inconsistent with the characteristics of the revised source terms.

For the purpose of this analysis, the activity available for release via MSIV leakage should be assumed to be that activity determined to be in the drywell for evaluating containment leakage (see Regulatory Position 3). [emphasis added]

3. SANDIA REPORT SAND2008-6601, ANALYSIS OF MAIN STEAM ISOLATION VALVE LEAKAGE IN DESIGN BASIS ACCIDENTS USING MELCOR 1.8.6 AND RADTRAD (RANDELL O. GAUNTT, TRACY RADEL, MICHAEL A. SALAY, AND DONALD A. KALINICH) (OCTOBER 2008)

   Analyses were performed using MELCOR and RADTRAD to investigate main steam isolation valve (MSIV) leakage behavior under design basis accident (DBA) loss-of-coolant (LOCA) conditions that are presumed to have led to a significant core melt accident. Dose to the control room, site boundary and LPZ are examined using both approaches described in current regulatory guidelines as well as analyses based on best estimate source term and system response. At issue is the current practice of using containment airborne aerosol concentrations as a surrogate for the in-vessel aerosol concentration that exists in the near vicinity of the MSIVs. This study finds current practice using the AST-based containment aerosol concentrations for assessing MSIV leakage is non-conservative and conceptually in error. [emphasis added]

   This misconception or oversimplification in viewing fission product transport from overheated fuel has led to subsequent important conceptual errors in analysis such as proposed use of drywell sprays to reduce airborne radioactivity (as illustrated in Figure 1-4) or equilibrating drywell and wetwell airspace volumes to achieve the same effect, when neither of these processes can directly affect the airborne concentration within the reactor vessel where a continuous source of fission products issues from the overheated fuel. In short, what is needed to evaluate MSIV leakage is a source term to the reactor vessel steam dome (which feeds the steam lines) and not a source term to the containment. [emphasis added]

4. DRAFT REGULATORY GUIDE DG-1199 (PROPOSED REVISION 1 OF REGULATORY GUIDE 1.183) ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS (OCTOBER 2009)

   As required by Title 10 of the Code of Federal Regulations, Section 50.34, "Contents of Applications; Technical Information" (10 CFR 50.34), each applicant for a construction permit or operating license must 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. [emphasis added]

   The facility final safety analysis report (FSAR) documents the analyses and evaluations required by 10 CFR 50.34 and 10 CFR Part 52. Fundamental assumptions that are design inputs, including

the source term, are to be included in the FSAR and become part of the facility design basis. [emphasis added]

The NRC issues regulatory guides to describe to the public methods that the staff considers acceptable for use in implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific problems or postulated accidents, and to provide guidance to applicants. Regulatory guides are not substitutes for regulations and compliance with them is not required. [emphasis added]

For BWRs, the main steam isolation valves (MSIVs) have design leakage that may result in a radioactivity release. The licensee should analyze and combine the radiological consequences from postulated MSIV leakage with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA.

The source of the MSIV leakage is assumed to be the activity concentration in the reactor vessel steam dome. [emphasis added]

All the MSIVs should be assumed to leak at the maximum leak rate above which the technical specifications would require declaring the MSIVs inoperable.

For fuel handling accidents postulated to occur within the fuel building, the following assumptions are acceptable to the NRC staff: The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period. [emphasis added]

5. TREATMENT OF REACTOR SYSTEMS WITHIN DRAFT REGULATORY GUIDE 1.183 DG-1199, N.C. ANDREWS AND R.O. GAUNTT, SANDIA NATIONAL LABORATORIES, ACCESSION NUMBER: ML19094A305 (APRIL 2019)

6. DRAFT REGULATORY GUIDE, DG-1199 - BWR OWNERS' GROUP REQUEST FOR SUPPORTING DOCUMENTATION AND COMMENT PERIOD EXTENSION (DOCKET ID NRC2009-0453, JANUARY 6, 2010)

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

7. NUCLEAR ENERGY INSTITUTE COMMENTS ON U.S. NUCLEAR REGULATORY COMMISSION DRAFT REGULATORY GUIDE DG-1199 (JANUARY 20, 2010):

There would likely be considerable more cost associated with complying with DG-1 199 than stated. Specifically, the proposed MSIV model changes in Section A-5 will significantly increase the doses from the MSIV release pathway. [emphasis added]

It is unlikely that BWRs would commit to using it due to extreme penalties with regard to MSIV leakages. [emphasis added]

By taking the approach proposed in the DG, BWRs with AST will be forced to tighten up the MSIV leakage pathway to reduce dose consequences or expend significant effort/cost to defend a seismically rugged condenser system or to re-activate leak collections systems that have high maintenance costs." [emphasis added]

Some early BWRs excluded MSIV leakage from LA [La referenced in App. J to Part 50], with NRC eventually requiring an application for a 1OCFR50 Appendix J exception, and MSIV specific LOCA dose contribution assessment.

Currently, the non LOCA gap fractions (Table 3) for Kr-85 (10%), I-131 (8%), and the alkali metals (12%) are set to different values from the post LOCA gap phase release fractions for these isotopes (5%). These differences are made larger with the revisions to Table 3 and the new Table 4 proposed in DG-1199. Higher baseline non LOCA gap fractions are proposed for Kr-85 (35%) and the alkali metals (46%), significantly increasing the differences between them and the corresponding post LOCA gap release fractions.

An important point to make is that the Reg Guide should be written in a way as to NOT preclude the possibility for fuel rod average exposures beyond 62 GWd/MTU in the future. A path and/or acceptable method for calculating approved source terms at higher burnup should be included in the Reg Guide. (i.e. the AST Reg Guide should not dictate the industry's maximum allowable fuel exposure) [emphasis added]

Provided below is a comparison of NUREG/CR 5009, RG 1.183 Rev 0, and DG-1199 gap fractions.

Group	*NUREG/CR 5009 (TID)* [1988]	*RG 1.183 Rev. 0* [2000]	*DG-1199* [2009]
I-131	*0.12*	*0.08*	*0.08*
Kr-85	*0.3*	*0.1*	*0.35*
I-132	*0.1*	*0.05*	*0.23*
Other Hal	*0.1*	*0.05*	*0.05*
Other NG	*0.1*	*0.05*	*0.04*
Alkali	*0.17*	*0.12*	*0.46*

8. RESPONSE TO THE BOILING WATER REACTORS OWNER'S GROUP REQUEST TO EXTEND THE COMMENT PERIOD FOR DRAFT REGULATORY GUIDE – 1199 (MARCH 22, 2010)

By letter dated January 6, 2010, the Boiling Water Reactor Owner's Group (BWROG) requested an extension of the public comment period for Draft Regulatory Guide – 1199 (DG-1199), "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Agencywide Documents Access and Management Systems (ADAMS) Accession No. ML090960464, open from October 14, 2009, to January 13, 2010. The extension request stated that, in order to gain an understanding of the implications and potential consequences of the proposed revision, the BWROG will need to perform a detailed review of the Staff's research supporting the proposed changes to modeling of the main steamline isolation valve (MSIV) leakage. The January 6, 2010, letter also included a request for the MELCOR input decks supporting the boiling water reactor (BWR) MSIV leakage analyses.

The Nuclear Regulatory Commission (NRC) staff has reviewed the stated basis for the request to extend the public comment period. Based upon this review, the staff has determined it will not extend the public comment period for the reasons discussed below.

On October 9, 2010, the staff released the technical basis for the proposed DG-1199 MSIV modeling changes to the public in a Sandia National Laboratories Report, SAND2008-6601, "Analysis of Main Steam Isolation Valve Leakage in Design Basis Accidents Using MELCOR 1.8.6 and RADTRAD," ADAMS Accession No. ML083180196. On November 16, 2010, the staff held a full day public workshop that included a presentation on the proposed MSIV modeling changes, including an extensive discussion of the role of the supporting MELCOR work. Based on its review of the request by the BWROG, <u>the staff has determined that no substantive issues with the staff's research were identified as the basis for extending the public comment period</u>. [emphasis added]

9. TECHNICAL BASIS FOR REVISED REGULATORY GUIDE 1.183 (DG-1199) FISSION PRODUCT FUEL-TO-CLADDING GAP INVENTORY (JULY 26, 2011 NRC MEMORANDUM):

Table 3 Non-LOCA Fraction of Fission Product Inventory in Gap

Group	*Fraction*
I-131	*0.08*
I-132	*0.09*
Kr-85	*0.38*
Other Noble Gases	*0.08*
Other Halogens	*0.05*
Alkali Metals	*0.50*

10. NRC INFORMATION NOTICE NO. 90-08: KR-85 HAZARDS FROM DECAYED FUEL (FEBRUARY 1990):

Analysis of hypothetical accidents involving decayed spent fuel has focused attention on potential difficulties that could be associated with the exposure of onsite personnel to an accidental release of Kr-85. Kr-85 is a noble gas fission product that is present in the gaps between the fuel pellets and the cladding. It has a 10.76-year half-life, and, as a result of the considerably shorter half-lives of virtually all other gaseous fission products (I-129 being the exception, but in low abundance), Kr-85 becomes increasingly the dominant nuclide in the accident source term for gap releases as decay times increase. After 2 weeks of decay, Kr-85 is a significant nuclide in the source term, and after 190 days of decay, it is the predominant gaseous nuclide for a gap release. The unusual decay characteristics of Kr-85 give cause for focusing attention on the onsite consequences of a gap release from decayed fuel.

Kr-85 emits beta radiation with a maximum energy of 0.67 MeV for 99.6 percent of the decays and 0.51 MeV gamma radiation for 0.4 percent of the decays. Consequently, direct exposure to this gas would result in a dose to the skin approximately 100 times the whole-body dose.

Accordingly, it is important to be able to properly survey and monitor for Kr-85, and to assess the skin dose to workers who could be exposed to Kr-85 in the event of an accident with decayed spent fuel.

Licensees may wish to reevaluate whether Emergency Action Levels specified in the emergency plan and procedures governing decayed fuel-handling activities appropriately focus on concern for onsite workers and Kr-85 releases in areas where decayed spent fuel accidents could occur, for example, the spent fuel pool working floor. Furthermore, licensees may wish to determine if emergency plans and corresponding implementing procedures address the means for limiting radiological exposures of onsite personnel who are in other areas of the plant.

11. NUREG/CR-6042 (REV. 2) /SAND 93-0971 PERSPECTIVES ON REACTOR SAFETY (MARCH 2002)

<u>*Nuclear power is by its very nature potentially dangerous, and ... one must continually question whether the safeguards already in place are sufficient to prevent major accidents*</u>*.* [emphasis added]

Table 5.1-1 shows the principal components of the 5 billion or so curies of radioactive materials in the core of a 3300 MWt light water reactor 30 min after shutdown according to their relative volatilities. Of the groups listed, radionuclides of the noble gases krypton (Kr) and xenon (Xe) are the most volatile and, consequently, the most likely to be released from the plant to the environment during an accident. Up to 100% of the noble gases could be released in severe accidents involving containment failure or bypass. Radioactive iodine and cesium, which rank second in volatility, could also be released in substantial quantities. Radioiodine can concentrate in the thyroid. As a result, small quantities of radioiodine can cause damage to the thyroid gland.

Long-lived radioactive cesium is a potential source of long-term offsite dose (e.g. from Chernobyl).

12. NUREG-1465 ACCIDENT SOURCE TERMS FOR LIGHT-WATER NUCLEAR POWER PLANTS (FEBRUARY 1995):

In 1962, the Atomic Energy Commission issued Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactors." In this document, a release of fission products from the core of a light-water reactor (LWR) into the containment atmosphere ("source term") was postulated for the purpose of calculating off-site doses in accordance with 10 CFR Part 100, "Reactor Site Criteria." The source term postulated an accident that resulted in substantial meltdown of the core, and the fission products assumed released into the containment were based on an understanding at that time of fission product behavior. In addition to site suitability, <u>the regulatory applications of this source term (in conjunction with the dose calculation methodology) affect the design of a wide range of plant systems</u>. [emphasis added]

<u>The use of postulated accidental releases of radioactive materials is deeply embedded in the regulatory policy and practices of the U.S. Nuclear Regulatory Commission (NRC)</u>. For over 30 years, the NRC's reactor site criteria in 10 CFR Part 100 (Ref. 1) have required, for licensing purposes, that an accidental fission product release resulting from "substantial meltdown" of the core into the containment be postulated to occur and that its potential radiological consequences be evaluated assuming that the containment remains intact but leaks at its maximum allowable leak rate. Radioactive material escaping from the containment is often referred to as the "radiological release to the environment." [emphasis added]

Recent information has indicated that high burnup fuel, that is, fuel irradiated at levels in excess of about 40 GWD/MTU, may be more prone to failure during design basis reactivity insertion accidents (RIA) than previously thought. <u>Preliminary indications are that high burnup fuel also may be in a highly fragmented or powdered form, so that failure of the cladding could result in a significant fraction of the fuel itself being released</u>. In contrast, the source term contained in this report is based upon fuel behavior results obtained at lower burnup levels where the fuel pellet remains intact upon cladding failure, resulting in a release only of those fission product gases residing in the gap between the fuel pellet and the cladding. <u>Because of this recent information regarding high burnup fuels, the NRC staff cautions that, until further information indicates otherwise, the source term in this report (particularly gap activity) may not be applicable for fuel irradiated to high burnup levels (in excess of about 40 GWD/MTU)</u>. [emphasis added]

13. UPDATE ON GENERIC ISSUE 92: FUEL CRUMBLING DURING LOCA: LETTER FROM RALPH MEYER (SENIOR TECHNICAL ADVISOR IRA/ SAFETY MARGINS AND SYSTEMS ANALYSIS BRANCH DIVISION

OF SYSTEMS ANALYSIS AND REGULATORY EFFECTIVENESS OFFICE OF NUCLEAR REGULATORY) TO JOHN FLACK (ASSISTANT BRANCH CHIEF REGULATORY EFFECTIVENESS AND HUMAN FACTORS BRANCH DIVISION OF SYSTEMS ANALYSIS AND REGULATORY EFFECTIVENESS OFFICE OF NUCLEAR REGULATORY RESEARCH) (FEBRUARY 8, 2001)

14. GENERIC SAFETY ISSUES: ISSUE 170: FUEL DAMAGE CRITERIA FOR HIGH BURNUP FUEL (REV. 2) (NUREG-0933, MAIN REPORT WITH SUPPLEMENTS 1–34)

    The lower threshold for release of gap activity plus the dispersal of particulate fuel at high burnup would increase plant activity and public dose, and the dispersal of fuel would alter the character of these Chapter 15 transients.

    Resolution of this issue could be accomplished by updating the existing burnup-independent criteria to include the effects of burnup, or to develop substitute criteria, as appropriate. Updated criteria could be incorporated in revisions to 10 CFR 50.46.

    The provisions in 10 CFR 50.46 that are in question were controversial when originally established and changes to this regulation will be avoided unless absolutely necessary; it is possible that the existing criteria can continue to be used as long as careful attention is given to initial oxidation and method of analysis for high burnup fuel.

    <u>Although the specified acceptable fuel design limits are probably not providing the protection intended</u>, it is believed that licensees are employing more stringent measures that are not derived from the licensing safety analysis, e.g., power maneuvering restrictions and barrier fuel designs are being used to reduce fuel failures, which the 1% strain limit would not prevent. [emphasis added]

15. MEMORANDUM FOR W. TRAVERS FROM A. THADANI, "CLOSURE OF GENERIC ISSUE 170, REACTIVITY TRANSIENTS AND FUEL DAMAGE CRITERIA FOR HIGH BURNUP FUEL," MAY 4, 2001.

16. SECY-15-0148 EVALUATION OF FUEL FRAGMENTATION, RELOCATION AND DISPERSAL UNDER LOSS-OF-COOLANT ACCIDENT (LOCA) CONDITIONS RELATIVE TO THE DRAFT FINAL RULE ON EMERGENCY CORE COOLING SYSTEM PERFORMANCE DURING A LOCA (50.46C) (NOVEMBER 30, 2015)

    In SECY-12-0034, the staff also provided information related to an <u>emerging research</u> finding that high burnup fuel pellets could fragment, relocate axially and possibly disperse outside of the fuel rod during postulated design-basis accidents including, but not limited to, LOCA. <u>In March 2012, the staff did not have a sufficient technical basis for concluding whether and in what manner these phenomena should be addressed</u>. [emphasis added]

The current state-of-knowledge indicates that FFRD is not necessarily limited to LOCA and could occur during non-LOCA design-basis events in which fuel rod ruptures are predicted to occur. Any future regulatory action, if needed [?]*, should be developed in a holistic manner to address both LOCA and non-LOCA scenarios.* [emphasis added]

17. NUREG/CR-7012 ORNLITM-2010/41 UNCERTAINTIES IN PREDICTED ISOTOPIC COMPOSITIONS FOR HIGH BURNUP PWR SPENT NUCLEAR FUEL (JANUARY 2011)

18. RESEARCH INFORMATION LETTER (RIL) 0801, "TECHNICAL BASIS FOR REVISION OF EMBRITTLEMENT CRITERIA IN 10 CFR 50.46," (ADAMS ACCESSION NUMBER, ML081350225)

19. NUREG-1940, RASCAL 4: DESCRIPTION OF MODELS AND METHODS (SEPTEMBER 2012):

RASCAL 4 is a significant advancement in the U.S. Nuclear Regulatory Commission's emergency response consequence assessment tools. RASCAL 4 includes improvements in the models and methods related to source term calculations, atmospheric dispersion and deposition, and dose calculations.

The methods that the RASCAL 4 source term calculations use for nuclear power plant accidents are based largely on the methods described in NUREG-1228, "Source Term Estimation during Incident Response to Severe Nuclear Power Plant Accidents," (McKenna and Giitter, 1988). Various aspects of the source term estimation methodology, including release timing, have been modified to account for the accident source term insights in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants, Final Report," (Soffer et al., 1995).

RASCAL 4 adjusts the inventory of radionuclides that have half-lives that exceed 1 year to account for burnup. Equation 1-1 is used to calculate the inventory for the core-average burnup, I_{ACTUAL}, for nuclides with half-lives of more than 1 year. Inventories of nuclides with half-lives of less than 1 year are not adjusted for burnup because the activities for these nuclides are more closely related to reactor power than they are to burnup.

$I_{ACTUAL} = I_{38,585} \times (BURNUP_{ACTUAL} \div 38{,}585\ MWd/MTU)$ [emphasis added]

The RASCAL 4 database contains two fuel burnup numbers. The first is the core average burnup (megawatt days per metric ton of uranium) for each reactor. A default value of 30,000 MWd/MTU is used in the database for all reactors. This value represents the average burnup of a core that is roughly two-thirds of the way to the end of core life, assuming typical current fuel management practices. The value changes with time and with the mix of old and new fuel in the core. The user can change the value if more information is available, but usually such a change will not significantly change the calculated projected doses. This burnup number is used to adjust the available inventory of radionuclides with halflives greater than 1 year (Section 1.1.1).

The second burnup number is the peak rod burnup. The peak-rod burnup is used as the burnup of fuel to be sent to the spent-fuel pool for long-term storage. A value of 50,000 MWd/MTU is used in the database. Again, the user may change the value if a better number is available. The spent

fuel burnup is used to generate source terms for spent fuel accidents using the method in Section 1.1.1.

20. NUREG-1940 SUPPLEMENT 1, RASCAL 4.3: DESCRIPTION OF MODELS AND METHODS (May 2015)

    RASCAL 4.3 contains a number of new features and revision of several old features in response to the lessons learned by the U.S. Nuclear Regulatory Commission staff during its response to the events at the Fukushima Daiichi nuclear power plants following the March 11, 2011 earthquake off the coast of Japan and the tsunami that it triggered. This document is a supplement to NUREG-1940, "RASCAL 4: Description of Models and Methods," which is the technical basis for the models and methods used in the RASCAL computer code versions 4.0, 4.1 and 4.2. This supplement contains the technical basis for changes and additions to RASCAL implemented in versions RASCAL 4.3 and 4.3.1. Additionally, Appendix A provides errata information for NUREG-1940.

    RASCAL 4.3 includes new features to acquire meteorological data from the Internet, to calculate custom radionuclide inventories for reactor cores and spent fuel, to display a radionuclide activity balance within the power plant components and atmosphere, and to sort and display the radionuclides released to the atmosphere by importance to dose pathways. RASCAL 4.3 also adds the capability to import, merge, and export source terms. <u>Improvements include revision of the pressure-hole size method of calculating the leak rate from containment and revision of the calculation of spent fuel source terms</u>. [emphasis added]

    2.9 Spent Fuel and Fuel Cycle Facility Accidents

    > *There are two components to the RASCAL 4.3 spent fuel accident source term calculation. The first is the calculation of the at-risk radionuclide inventory, and the second is the calculation of the portion of the at-risk radionuclide inventory that is released to the environment. The changes to the spent fuel accident source term calculation in RASCAL 4.3 are in the determination of the at-risk radionuclide inventory in the spent fuel. There have not been any significant changes in the calculation of the spent fuel or fuel cycle facility source terms related to release paths.*
    >
    > *It is important for the user to understand that, the start time from when the fuel is uncovered to the time that runaway cladding oxidation (a.k.a., a zirconium fire) (if one occurs) is highly dependent on the age of the fuel, its storage configuration relative to other 'hot' fuel, the size and location of any leakage in the SFP, etc.*

    2.9.2 Damaged Fuel

    > *The at-risk radionuclide inventory in a damaged spent fuel accident is limited to the radionuclide inventories in the damaged fuel assemblies. RASCAL 4.3 calculates the at-risk radionuclide inventory in the damaged fuel assemblies based on the burnup of the damaged fuel assemblies and the time since the irradiation of the fuel. Similar to the SFP uncovered event, RASCAL 4.3 has two methods for calculating the at-risk inventories for the damaged spent fuel accidents.*

2.2 Loss of Coolant Accident (LOCA) Core Release Fractions

NUREG-1465 defines reactor release fractions for BWR and PWR LOCAs for 4 release phases: gap activity release, early in-vessel release, ex-vessel release, and late in-vessel release. RASCAL 4.3 and earlier versions of the code use the release fractions for the gap release, the early in-vessel release, and the ex-vessel release phases to define the release of core activity into the containment. These previous versions of the RASCAL computer code do not include the late in-vessel release phase. However, RASCAL 4.3.1 includes the activity released from the late in-vessel phase in LOCA source terms. The durations of each phase including the late in-vessel phase are defined in NUREG-1465 Table 3.6. The late in-vessel release phase commences simultaneously with the occurrence of the ex-vessel release phase and extends several hours after the end of the ex-vessel phase release. Table 3.8, of NUREG-1465, classifies radionuclides into eight radionuclide groups based on expected release characteristics. Core release fractions to containment for each group by release phase for BWRs and PWRs are listed in Tables 3.12 and 3.13 of NUREG-1465.

The only radionuclide groups that have releases during the late in-vessel release phase are the halogens (iodine and bromine group), the alkali metals (cesium and rubidium group), and the tellurium group (tellurium, antimony, and selenium). In RASCAL 4.3.1 the core release fractions for these groups for the ex-vessel phase have been increased to include the portion of the late in-vessel release phase that occurs during the time of the ex-vessel release phase, and the late in-vessel release fractions have been decreased to account for the activity released during the ex-vessel release phase. <u>Tables 2-6 and 2-7 present the core release fractions used by RASCAL 4.3.1 for the eight NUREG-1465 groups for PWRs and BWRs respectively. These tables include the revised release fractions for the ex-vessel and late in-vessel phases.</u> [emphasis added]

6.0 ACTIVITY BALANCE

The RASCAL 4.3 source term calculations track radionuclide activity in the fuel, as it passes through the plant, and as it enters the environment. The details of these calculations have been available in files used to track program execution and troubleshoot problems. However, these files have not been available to the RASCAL users in previous versions of the code. RASCAL 4.3 makes this information available to users by creating an activity balance file and displaying the contents through the user interface.

[Note: Neither NUREG-1940 nor NUREG 1940 Supplement 1 reference RG 1.183 or DG-1199.]

21. LETTER TO MR. VICTOR M. MCCREE (NRC) FROM RALPH O. MEYER, PH.D. (AUGUST 7, 2016)

22. NRC-2020-0055 / PRM-50-121, RE-SUBMITTAL - 10 CFR 2.802 PETITION FOR RULEMAKING ACCIDENT DOSE CRITERIA, PARILLO, JOHN (NOVEMBER 23, 2019)

    The new TEDE criterion [10 CFR 50.67] *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, the 25 rem TEDE criterion allows for the associated thyroid dose to substantially exceed the previously controlling 300 rem thyroid limitation. Therefore, new reactors are being sited with a less restrictive dose criterion than the earliest reactors.*

    [Note: Conspicuously missing from this well referenced petition, by a knowledgeable member of the NRC staff, is any reference to SAND2008-6601 or DG-1199.]

23. INFORMATION NOTICE NO. 82-23: MAIN STEAM ISOLATION VALVE (MSIV) LEAKAGE

    IE has completed a survey of MSIV performance at BWRs for the years 1979 through 1981. IE found that 19 of 25 operating BWRs had MSIVs which failed to meet, during one or more surveillance tests, the limiting condition for operation (LCO) which specifies the maximum permissible leak rate. The number of MSIV test failures exceeded 151 and occurred with MSIVs supplied by all three MSIV vendors, i.e., Atwood & Morrill, Crane, and Rockwell.

    Measured leak rates which exceeded the LCO ranged from greater than 11.5 standard cubic feet per hour (scfh) to 3427 scfh. Twelve stations had 57 MSIV tests with results greater than 11.5 scfh and less than 100 scfh, and five stations (nine units) had 66 MSIV tests with results between 100 and 3500 scfh. Four other licensees had more than 24 test failures but did not measure, estimate, or report the magnitudes of the leak rates. These results are summarized in Attachment (1) and are shown in detail in Attachment (2).

    This information indicates that some MSIVs may not adequately limit release of radioactivity to the environment if called upon to do so. NRC is considering the need for improved MSIV maintenance, more frequent MSIV testing or installation of leakage control systems.

    This information notice is provided as notification of events that may have safety significance. It is expected that recipients will review the information for applicability to their facilities; however, <u>*no specific action or response is required at this time*</u>. [emphasis added]

24. NUREG-1285, NRC STAFF EVALUATION OF THE GENERAL ELECTRIC COMPANY NUCLEAR REACTOR STUDY ("REED REPORT") (JULY 1987)

    4.3 Main Steam Isolation Valve Leak Tightness Issue

    The issue of leak tightness of main steam isolation valves (MSIVs) was identified in the Reed Report in the section on Mechanical Systems and Equipment, but was not discussed in the GE status report provided in 1978. <u>*Main steam isolation valves (MSIVs) have been notorious for*</u>

<u>leaking at high rates</u> when they are tested during the 18-month leak tightness testing that is generally required by the technical specifications. Most plants have a technical specification leak rate limit of 11.5 standard cubic feet per hour (scfh) per valve. At some plants the as-found leak rate has been as high as 4500 scfh. With such high leak rates, the MSIV-leakage control system (MSIV-LCS) probably would not be capable of performing its safety-related function of removing the leakage from between the closed MSIVs following a design-basis LOCA.

Safety Significance

In its evaluation of the safety features of nuclear power plants, the past practice of the staff has been to give no credit for any structure, system, or component that was not safety related (sometimes referred to as safety grade). Given this past practice, following a design-basis LOCA with no credit for non-safety-related components, and assuming the single failure of one MSIV to close, the design-basis maximum allowable leakage through the MSIVs, for most plants, is 11.5 scfh. This limit on MSIV leakage is to maintain the offsite radiological consequences to within a small fraction of regulatory limits in the event of an accident. Thus, <u>if the MSIVs were to leak at a rate greater than 11.5 scfh</u>, and particularly at a rate that caused the MSIV-LCS to fail, <u>the offsite consequences could exceed the regulatory limits in the event of a severe accident</u>. [emphasis added]

Status

In recognition of this continuing problem of MSIV leakage, and the potential consequences in terms of offsite doses, the NRC staff early initiated Generic Issue C-8, "MSIV Leakage and Leakage Control Systems Failures." This generic issue considered the actual natural phenomena associated with the behavior and the characteristics of radioactive materials and the historical capability of "nonsafety-related" components to survive seismic events. In assessing the consequences of MSIV leakages, credit was given for fission product decay, plate-out on cold surfaces, and gravitational settling, and for a realistic evaluation of the actual materials that would be transported along the main steam line. Because it is assumed in design-basis accient analyses that offsite power will be lost following a LOCA (as a result of the tripping of the turbine generator and failure of offsite power), no credit was given for any equipment that was not powered from the emergency diesel generator busses. The analysis performed under Generic Issue C-8 indicates that the leak rate through MSIVs could be as high as 1500 scfh without using the MSIV-LCS, and the offsite doses would be less than those specified in the regulations. The study identified a method of calculating this leakage rate, but the actual leak rate would have to be determined on a plant-by-plant basis. This information was documented in NUREG-1169, published in August 1986.

MSIV leak tightness was a concern in 1975, and it is still a concern that has not been fully resolved. The BWR Owners Group (BWROG) formed a committee to evaluate this same issue independently, with GE giving technical support to the BWROG committee. <u>This committee generally found that the high leakage rates were attributable to valve maintenance practices</u>. For those plants that have adopted the BWROG recommendations resulting from their evaluation, the as-found MSIV leak rates have generally been within the plant-specific technical

specification limit, or within a factor of 2 or 3 of that limit. For example, Peach Bottom 3, had typical as-found leak rates of over 3000 scfh for each of the MSIVs. After following the BWROG recommendations, the next as-found leak rates were found to be less than 11.5 scfh for seven of the eight MSIVs and approximately 14.7 scfh for the eighth MSIV. This demonstrates that the MSIVs can be maintained within their respective technical specification leakage limits, and that the use of the leakage control system is not necessarily the optimum method for handling the leakage through the MSIVs in the event of a LOCA. [emphasis added]

The technical specification limit of MSIV leakage is conservatively set to ensure that offsite dose consequences of a main steam line break are a small fraction of the regulatory limits in 10 CFR Part 100. Although MSIV leakage is an issue of continuing concern, the current state of the art and conservative limits justify continued operation of BWR plants as the MSIV leakage issue is pursued.