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Date: 10/31/03 4:22PM
Subject: NEI comments on NUREG CR/6595

Dan:

Our comments are attached for your information. There were sent by hard copy to the rules and directives branch.

If you have any questions, please let me know.

Biff Bradley

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NUCLEAR ENERGY INSTITUTE

Anthony R. Pietrangelo
SENIOR DIRECTOR, RISK REGULATION

October 31, 2003

Chief
Rules and Directives Branch
Office of Administration
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Comments on Draft NUREG/CR-6595 Rev. 1, *An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events* (68 Federal Register 52064, August 29, 2003)

The Nuclear Energy Institute[1] offers the following comments on the subject *Federal Register* notice, which solicited public comments on the proposed revisions to NUREG/CR-6595. As indicated in the attachment, we believe significant revisions are necessary to this proposed NUREG.

NUREG/CR-6595 provides information that would be used to support risk-informed decisionmaking by both industry and NRC staff. Proper consideration of containment sequences and bypass events is important, as these events have significant impacts on risk metrics and conceivable large impacts on regulatory decisions. As recently discussed by the Chairman of NRC, realistic conservatism should be an important consideration for NRC decisionmaking. We believe the current draft of NUREG/CR-6595 does not live up to the concept of realistic conservatism, as it contains excessive conservatism, and simplifications that could lead to unrealistic results.

Our detailed comments are attached. Please contact me if you would like to discuss these comments further, or desire additional information.

Sincerely,

Anthony R. Pietrangelo

Anthony R. Pietrangelo

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1 NEI is the organization responsible for establishing unified nuclear industry policy on matters affecting the nuclear energy industry, including regulatory aspects of generic operational and technical issues. NEI members include all utilities licensed to operate commercial nuclear power plants in the United States, nuclear plant designers, major architect/engineering firms, fuel fabrication facilities, materials licensees, and other organizations and individuals involved in the nuclear energy industry.

NEI COMMENTS SUBMITTED TO U.S. NRC
REGARDING NUREG/CR-6595 REV. 1 (Draft)

Summary Comments

In order to be an effective tool in risk-informed decision-making, the following should be considered for inclusion in NUREG/CR-6595:

- The definitions (e.g., LERF) should be made consistent with R.G. 1.174 and ACRS guidance.
- The technical basis for the quantified node values should be provided so that users can know how to adjust those values when the technical inputs associated with the quantification are different on a plant-specific basis.
- The acceptable methods or approaches to make these CET inputs plant-specific should be identified.

The following methods described in the document are inadequate for effective implementation:

- The shutdown CETs appear to be too simplified. In addition, the shutdown CETs are provided without a technical basis for the nodes (mitigation capabilities) that are included and for those nodes that are dismissed as ineffective.
- The BWR methodology appears to be based on outdated BWROG EOPs. This compromises the degree of confidence that can be had in the report.
- Mark III CETs are not considered to be appropriate because of the extensive plant-specific capabilities that are not incorporated.
- The split fractions for early containment failure of Ice Condenser plants under conditions without the igniters are grossly conservative and based on an inappropriate reference.
- Late containment failure is not part of the risk metrics cited in RG 1.174 and should be removed. In addition, as written, the incorporation of "late containment failure" seems to confuse late failures with large failures. The description of what is being evaluated is inadequate and the basis for the approach is incomplete. Finally, the late containment failure CET and methods appear to be added as an afterthought. The associated CET is too primitive to be useful.

General Technical Comments

1. The main report does not define Large Early Release, yet this does not preclude assignment of CET end states to Large Early Release without a definition. Appendix A provides three separate definitions of LERF. However, it indicates all three definitions "have been utilized" (P. A-3). This implies that the most limiting definition is used. This, of course, is inconsistent with the intent of a risk-informed tool and is contrary to R.G. 1.174.

2. Furthermore, the main report does not define Late Containment Failure. The various uses of the term invoke terms and issues which are confusing and contradictory. The terms "late containment failure" and "early health effects" (p. 1-3) are not defined. This can only lead to confusion in the interpretation of the analysis and its usefulness. In the late containment failure event trees, the endstates are labeled "Large Release". The assessment does not address large vs. small, only whether containment failure is likely. There are many factors that could impact whether a release is "large" or not including containment failure mode, fission product scrubbing in the release pathway, timing of the release, etc. This is judged to result in significant conservatism in the assignment of "large" releases.

3. The reference to IPE results as the technical basis for the individual point estimate conditional probabilities is a useful reference, but it should be made clear that those values are neither recommended nor suggested values for use in the CET quantifications. The IPEs were developed at a time when severe accident phenomena had not been fully assessed. IPE analyses were developed with varying degrees of conservatism, where the purpose was only to demonstrate that no vulnerability existed. As such, the use of conservative assumptions was considered appropriate as long as it did not influence the identification of "vulnerabilities."

   There have been substantial additional evaluations using later versions of deterministic codes, such as MAAP 4. These evaluations plus the changes in the Level 1 CDF risk profile and Emergency Operating Procedures make the IPE-developed conditional probabilities out-dated and no longer applicable to the as-built, as-operated plants of today.

   Risk-informed decision-making has stringent requirements in that the input information should be realistic and not biased in the conservative or non-conservative direction. The use of conservative, 10-year-old IPE assumptions to define the conditional probability of uncertain phenomena would appear to be a misapplication of this information to a realistic risk-informed process.

   Because the IPE results use varying levels of conservatism in the assessment of phenomena, it is considered inappropriate to use such subjective information as the technical basis for a document to be used in the future for risk-informed decision making. There does not appear to be any tie of the CET probabilities cited in Sections 2 and 3 to specific recognized, realistic technical analysis.

There is a general lack of sufficient descriptive material to allow clear interpretation of the modeling approach adopted in the subject report. Examples of needed documentation to clarify the level of conservatism in these models include the following:

- The definition of each node in the event tree. This should include a top logic fault tree to describe the "acceptable" inputs and a technical basis for the success criteria.
- A definition of the criteria that would be applied to change the assumptions in NUREG/CR-6595 where they are inappropriate on a plant specific basis.

4. The following statement is provided to qualify the use of the conservative "bounding" point estimates used in the simplified CETs:

   > "... as the split fractions are intended to encompass the likelihood of containment failure for most of the plants within a particular containment type they are somewhat bounding in nature. Consequently, an alternative split fraction (less bounding) could be used for a particular plant provided sufficient justification is given."

   However, by not providing a technical basis for the proposed point estimate or what constitutes its "bounding" properties makes adjustment to this value on a plant-specific basis very difficult.

5. The fundamental flaw in the CETs of NUREG/CR-6595 is that the Level 1 accident sequence dependencies are not captured in the calculation of model failure probabilities in the CET. This is of particular concern for BWRs where there is significant interplay between the Level 1 PRA systems and the containment performance. To disconnect the consideration of these important interactions creates the potential to miss key dependencies. To overcome this flaw, it appears that excess conservatism is being added to the Level 2 CET node quantification. This would appear to place extreme penalties on plants that have properly accounted for these dependencies.

6. NUREG/CR-6595 Rev. 1 CETs, as noted in the report are simplified and will yield conservative estimates of the LERF. Examples of the very conservative nature of the modeling are the following:

   Mark II Plants:

   Containment failure at vessel breach is 0.3 for the RPV at high pressure. There is no basis to support a conditional probability this high based on the phenomena cited, e.g., steam explosions, vessel blowdown, and direct containment heating.

   Ice Condenser Plants:

   The split fractions identified for the cases when igniters are unavailable are taken from a bounding assessment that was not intended to provide realistic input for risk-informed decision-making.

7. It is not clear how the LERF CETs and Late Containment Failure CETs are to be used to quantify various containment failure states. For example, how are the LERF CET endstates that answer the final question affirmatively (No Potential For Early Fatalities) accounted for? There appears to be a general lack of coordination between the endstate assignments in the LERF CET and the Late Containment Failure CET. The two CETs do not provide comprehensive coverage and accounting of the endstates. (See Comment #3.22 in Section 3.) There appears to be a failure to fully account for the various containment endstates (i.e., early/late, large/not large) in the proposed framework. This is even more true in the shutdown CETs.

Section 1 - Introduction

1.1 Section 1 of the report (and the Abstract) should be clear on whether this methodology is for calculating quick bounding estimates (e.g., "...provides a simplified approach to calculate a bounding LERF estimate ...") or realistic LERF values. The report seems to take both positions. In some sections, it states that the method is to obtain initial bounding LERF estimates, and in other sections it mentions "reasonably accurate estimate of LERF". For example, does the phrase "...to initially estimate LERF" on p. x equate to "...calculate a bounding LERF estimate"?

1.2 The subject report states that the CETs "should only be used as a first step scoping study" (p. 1-3); however, the degree of conservatism in the CETs significantly reduces their usefulness as a scoping tool. (See Comment 1.7.)

1.3 In the fourth paragraph on page 1-2, the report refers to draft DG-1061. This should be changed to Reg. Guide 1.174.

1.4 P. 1-2 and the BWR sections refer to ISLOCA as the only contributor to the Containment Bypass containment failure category. Breaks Outside Containment (BOC), which are LOCAs outside containment in high pressure rated lines, should also be included in this containment failure mode category.

1.5 In the definitions on page 1-2, the following definitions are inconsistent with the R.G. 1.174 definition of LERF:

- Early structural failure – references "early" to the time of core damage rather than to the time of effective evacuation or public protective actions.
- Late structural failure – references "late" to reactor vessel failure rather than to the time of effective evacuation or public protective actions.

P. 1-2 defines late structural failure as "several hours after reactor vessel failure". This definition should be further clarified, because the current definition can result in defining late structural failures for scenarios that result in LERF releases. For example, assuming "several hours" is 3 hours, in the case of a LLOCA without injection containment failure can occur in the 4-6 hour time frame – which would result in "early" releases as defined in many industry PRAs.

1.6 The last line of page 1-2 refers to five simplified CETs. There are now many more than five CETs.

1.7 P. 1-3 and other sections of the report mention that if the estimated LERF using the NUREG/CR-6595 approach is near the DG-1061 acceptance guidelines (again, should be referencing RG 1.174 here), near being defined as an order of magnitude, that further analysis would be required. Does this guideline of an order of magnitude apply to total LERF or delta LERF? If total LERF, the RG 1.174 acceptance guideline is 1E-5/yr, then "near" would be 1E-6/yr – which would likely mean that the NUREG/CR-6595 initial LERF estimating approach would rarely be successful as a quick useful estimate.

1.8 References 3 and 4 appear to be identical (This is also true in the Executive Summary)

Section 2 – Simplified Event Trees for PWRs

2.1 Under Question 2, there is a reference indicating that "*leakage rates greater than 100 percent containment volume per day have been found to be risk significant in past studies.*" No references are provided for these studies.

2.2 The discussion on pages 2-2 and 2-3 regarding SGTRs caused by SG pressure differentials provide a lot of considerations but no references or specific approaches to quantifying the likelihood. Additional guidance is necessary to appropriately quantify this node.

2.3 At the end of the discussion on SGTRs on page 2-3, there is a parenthetical reference to the potential for SGTR in ATWS events. However, the next bullet specifically addresses ATWS events. It is not clear if additional considerations are necessary or if this is redundant.

2.4 Expand the bullet on page 2-3 that discusses loss of containment heat removal (CHR). In PWRs with large volume, these events take very long times to occur. As such, the ensuing core damage would generally not qualify as "early" releases. This consideration should be addressed.

2.5 The discussion regarding Internal Fire on page 2-3 describes the treatment of containment isolation as "normally quantified" as part of the Level 1 fire PRA. Consistent with internal events PRAs, it is not standard practice to include containment isolation in Level 1 fire PRAs.

2.6 For ice condenser containments, the bases for quantification of Question 7 (No Containment Failure at or Before Vessel Breach) is based on NUREG/CR-6427. NUREG/CR-6427 analysis are very conservative with respect to the quantities of hydrogen generated during core melt and combustion mechanisms present without igniters. The hydrogen quantities used in NUREG/CR-6427 are equivalent to the 95th percentile values of the distribution provided in NUREG/CR-6338 and the fraction of clad oxidized was biased much higher than best estimate values. In the area of hydrogen combustion mechanisms, the analysis ignores the potential for early ignition of hydrogen accumulation (i.e., before the pressure rise could threaten containment) and ignores the fact that the lower compartment may be steam inerted for low pressure sequences

2.7 The approach to PWR Late Containment Failures is overly simplistic and will lead to inappropriate results. Inadequate definitions are provided to understand what is meant by "late" and "large". The variety of containment configurations, plant damage states and accident management strategies are not adequately reflected in this simplified model.

2.8 The last sentence of the first paragraph of Section 2.3 appears to be redundant to other statements from previous sections and does not appear necessary.

2.9 Under question 1, the discussion appears to be focused on whether the cavity is flooded at the time of vessel failure. Two additional issues need to be addressed here. First, whether the cavity will be flooded following vessel breach (i.e., due to low pressure systems injecting when vessel pressure drops). The second issue

is whether the cavity remains flooded long term (i.e., whether makeup to the cavity will continue). An initially flooded cavity is not sufficient. Without replenishment, the cavity will dry out and the core concrete interactions will initiate.

2.10 Figure 2.3 is labeled "Late Containment Failure for PWRs", but the endstates are labeled "Large Release". This implies that the containment failures labeled with "Yes" are "large". This is simply untrue. Many other factors impact the source term for these scenarios, including containment failure mode, failure location, accident management strategies employed, etc.

2.11 Sequences 4 and 6 of Figure 2.3, where containment failure is due to ineffective CHR could be due to the buildup of non-condensibles in the containment. Depending upon the containment, such a pressurization could take a day and a half or longer. This is outside the timeframe generally considered in PRAs and one that could allow significant offsite resources to be brought to bear in addressing the cause of containment challenge. In addition, in steel lined concrete containments, the dominant containment failure mode for such scenarios is often a leak-type failure mode that would not qualify as a "Large" release.

2.12 The potential for late hydrogen burns does not appear to be addressed at all in Figure 2.3. In cases where sprays (or fan coolers) are recovered and initiated, the normal steam inerted condition would be lost. This scenario has been found to be a contributor in some Level 2 PRAs to late containment failure. In fact, it is one of the few late containment failure modes that would be expected to cause a containment rupture leading to a large fission product release.

2.13 The description of Question 2 (Is Core Debris Coolable) is inadequate to guide quantification. Additional references or bases must be provided in order to support assignment of the nodal probabilities.

2.14 In addition, under Question 2, it is important to track where the core debris goes when the vessel fails. For HPME events, significant core debris may accumulate outside of cavity. This may have both positive and adverse impacts. It spreads the debris over a larger area, but may also require that water be supplied to multiple areas. These considerations should be referenced.

Section 3 – Simplified Event Trees for BWRs

3.1 Section 3.1 appropriately states that the structure of the BWR CETs assumes that all early releases that are scrubbed by the suppression pool are not LERF. However, the CETs do not appear to explicitly discuss the suppression pool. Consider providing more discussion on this topic including specific nodes to address the scrubbing capability.

3.2 P. 3-1, the discussion for Question 1 (Core Damage) is sparse and would not be sufficient direction for a significant fraction of readers. A reader would expect to see directions on process, different approaches, tips, etc.

3.3 The 1.0 containment failure probability (early failure mode) for Mark I vessel breach without water scenarios would not apply to Mark I's with concrete containments (e.g., Brunswick). This should be acknowledged in the guidance. Brunswick Units 1 and 2, for example, have a concrete torus and drywell. There is not the same impact of debris attack on the drywell shell that is seen in other Mark I plants. In a similar vein, Oyster Creek has a curb in the drywell that offers some protection against the debris attack of the steel shell. These differences make it improper to use a single value and imply its applicability to all Mark I plants.

3.4 The Mark I and II scenarios for core melt arrested in-vessel are modeled directly to No Large Early Release. This approach must be dismissing in-vessel steam explosions and other related phenomena that can result in large containment failure even with the core initially arrested in-vessel. This should be acknowledged in the guidance.

3.5 P. 3-3, the discussion for Question 2 (Containment Isolated) that addresses "failure to isolate" should provide additional guidance. Information on pre-existing containment large leakage can be found in references such as NUREG/CR-4220 (and others).

3.6 P. 3-3, the discussion for Question 2 (Containment Isolated) that addresses ISLOCA states that it is necessary to determine whether or not the release path is submerged. Further guidance should be provided to the reader regarding how an ISLOCA release can be submerged and when this condition could lead to a non-LERF end state.

3.7 P. 3-3, the discussion for Question 3 (RCS Depressurized) appropriately discusses the possibility of the RCS boundary failing. However, the reader should be provided with information regarding the failure modes and their probabilities.

3.8 P. 3-4, the discussion for Question 4: Core Damage Arrested In-Vessel for depressurization by operator states that the plant "may wish" to take credit for low pressure injection for high pressure core damage scenarios that are depressurized in Question 3. The text should be revised to state that the plant should credit low pressure injection in such scenarios.

3.9 Under Question 6, the conditional probability assigned to the high pressure core melt scenarios for Mark I containments does not use the technical information developed in NUREG/CR-5423 (Theofanous). This publication demonstrated that immediately following RPV blowdown when RPV injection is restored, the core debris will be quenched and shell failure will be prevented with a high success probability.

3.10 The revised NUREG/CR 6595 Rev. 1 CETs for BWRs has added a node to address questions regarding RPV venting. However, this is an example of the use of 10-year-old IPE information in today's evaluation. While the IPEs did point out that the EPGs in place at the time (1992) could result in RPV vent and a consequential high, early release, the BWROG has since implemented a change to the containment flood procedure to preclude this RPV vent release mode. Therefore, the IPE insights no longer apply.

3.11 In addition, the NRC has added the potential for drywell venting to result in early high releases, however, calculations of the containment flood process and the use of drywell sprays has identified this release mode as <u>not</u> a large release. The NRC should review these insights and any NRC calculations before implementing the identified Question No. 7 for BWRs.

3.12 For Mark III BWRs, the node for RPV venting/DW venting does not apply because neither the RPV vent EOP direction nor the drywell vent description are appropriate for current Mark III operations. The vent process is from the containment wetwell air space, and the radionuclide release would be scrubbed through the suppression pool.

3.13 The Mark III containment is significantly different from the Mark I and II containments. However, these substantial differences do not appear to be reflected in the CETs presented in NUREG/CR-6595.

3.14 The Mark III CET description has a caution at the beginning that cites the significant difference in the Mark III design compared with the Mark I or II designs and provides qualitative guidance. However, the translation of this qualitative guidance into the quantified CET appears to be lacking. There is no clear link between the CET and the significant plant design differences.

3.15 Mark III Question 2 (P. 3-10) (Containment Isolated or Not Bypassed?). The question does not adequately address the possible different containment isolation and bypass methods.

a. directly from the RPV to the environment

b. directly from the RPV to the wetwell air space

c. containment leakage with the drywell bypassed

d. These failures can lead to bypass of the suppression pool and therefore the discussion of containment leakage does not apply. Each of the three identified classes of events would require evaluations not covered by the description provided.

3.16 The NUREG/CR-4551 analysis used older EPGs and HCOG procedures. The current procedures specifically restrict igniter operation if they are not initiated prior to RPV water level decreasing below TAF. These procedures are considered to significantly alter the assessments in NUREG/CR-4551. These in turn, make the NUREG/CR-6595 observations based on the out dated NUREG/CR-4551 analysis inappropriate for this document.

3.17 The purpose of developing a "late" containment failure needs to be clarified. If it is only to assess seismic sequences, this should be clearly stated and the CET renamed.

3.18 The Mark III CET does not question one of the most important determining factors in the BWR Mark III for the assessment of LERF determination, i.e., the containment sprays. Without determining the containment spray status, which protects the containment outer shell and provides fission product scrubbing, it would appear to be very difficult to arrive at a realistic LERF estimate.

3.19 The "Late Containment Failure for BWRs" CET (Figure 3.4) does not apply to the Mark III containment. Specifically, the Mark III containment can be effective in limiting the radionuclide release to less than "large" if either containment venting is implemented or containment fails in the wetwell airspace because of the effectiveness of the suppression pool scrubbing.

This is part of a larger question regarding each of these nodes in that the success criteria for the node is not clearly stated. For example, does the CHR node include RHR, sprays, and venting (or a combination of each) as a success for the node. The wording certainly <u>implies</u>, but does not state, that the active RHR system is what is meant as CHR. This needs to be stated if that is the NRC assumption.

3.20 The "late containment failure" evaluation (Figure 3.4) does not appear to be consistent with thermal-hydraulic analysis and no references could be found to support the assignment of "large" releases. In fact, the discussion (definition) on P. 1-2 appears to be inconsistent with its usage in the discussion of Figure 3.4.

3.21 The following statement does not appear to be adequate to describe the "late containment event tree" nodal approach, nor is there guidance in the document regarding the development of answers to these questions.

> *most of the questions will be determined from information provided in the Level 1 PRA supplemented by additional analysis and information. The CETs include split fractions only for questions dealing with the likelihood of containment failure.*

The Level 1 PRA does not calculate the nodal probabilities included in the "late" containment failure CET.

3.22 Given that the "late containment failure" assessment is retained in NUREG/CR-6595, the following direction on use of Late Containment Failure Figure 3.4 does not appear appropriate:

> *The fraction of these accident sequences that do not result in containment failure (i.e., positive response to Question 6 and 7 in Figure 3.1, Figure 3.2, and Figure 3.3) are processed through the CET in this section.*

This direction does not encompass all of the sequences that could be affected, i.e., the proposed directions are non-conservative. For example, Mark I sequences that should be added to this list include: Sequences 1, 3, 5, 8, 10, 12, 14, 16, 19, 21, 23, 24.

3.23 One of the significant mitigation measures in the BWR Mark III is the use of sprays. The "late containment event tree" does not properly treat the effectiveness of sprays in the mitigation of accident sequences.

3.24 The Figure 3.5 "Late Containment Failure for BWRs" CET Sequence Path 2 does not always result in a "Large" Release. Either the containment vent or a drywell head failure in a Mark I or II with DW sprays operating would result in a low radionuclide release. This has been shown by extensive industry severe accident calculations, however, this success path is not acknowledged by the CET.

Section 4 – Simplified Event Trees During Shutdown

4.1 The shutdown event trees are extremely simplistic. Implementation of the concepts contained here will be extremely complex. The primary complication is that these simple event trees would potentially need to be applied to many, many configurations within a POS, in order to address the timing and dependency considerations.

4.2 The simplified event trees imply that core damage accidents that occur in POS 3 and the late occurrences of POS 2 and 1 can not lead to releases that cause early fatalities. However, in some cases, this may not be valid, For example, in a 13 day outage at a PWR involving a late mid-loop condition and the containment hatch open, could potentially lead to an "early" and "large" release, depending on a number of factors.

4.3 The impact of external events do not appear to be addressed in Section 4.

4.4 The shutdown event trees developed do not have the "core damage arrested in-vessel" node that is included in the at-power simplified CET. The ability to restore adequate core cooling during shutdown is judged to be much more likely under POS 1 and POS 2 than at-power. To neglect this node appears to introduce an unacceptable conservatism. (See also comment B.4)

4.5 There is no thermal hydraulic analysis provided or referenced to support the timing of early release in the CET. It would appear that the CET is premature until there is some thermal hydraulic analysis to support the decisions in the CET.

4.6 Some of the information that is cited to support the shutdown CETs in NUREG/CR-6595 Rev. 1 are References 3 and 4, draft documents. These draft documents should be part of the NUREG/CR-6595 Rev. 1 review and comment process.

4.7 For BWRs, the treatment of suppression pool scrubbing appears to require the DW head to be in place (p. 4-4). The ability to arrest core damage in-vessel and to have the release path scrubbed in the suppression pool would result in a reduced release magnitude to below the "large" definition. This does not require the drywell head to be in place.

4.8 No credit for the Reactor Building and the SGTS is included in the analysis. Without analysis to support such a conservative assumption, it appears imprudent to not include them in the CET.

4.9 This "so-called" Simplified Event trees During Shutdown are not event trees as are normally developed as part of a PRA. They do not provide a basis for quantification of nodal events and the development of accident sequence probabilities. They are merely a series of questions. The series of questions, then, has a probabilistic estimate appended to it without any technical basis or discussion.

4.10 Core damage is assumed to lead to LERF directly (e.g., POS 2). There is no credit for the mitigative actions that can be taken for:

- In-vessel recovery to terminate core melt progression
- Operation of SGTS
- Purge and vent
- Scrubbing (Mark III)

4.11 The second, fifth, and eighth sequences in Figure 4.5 relate to an indeterminate state (i.e., Large Late or no Failure). This is inconsistent with the objective of identifying late containment failures. If these event trees are to be retained, then these trees should be expanded or the guidance revised to address how to determine the endstate.

4.12 On Page 4-9, Question 4, implies that, if open. the containment must be reclosed and capable of holding design pressure. Holding design pressure may not be necessary to avoid an early release. Some licensees utilize temporary hatches that can withstand a lower pressure, but are capable of delaying release and withstanding pressures expected following loss of core cooling events.

4.13 The last sentence under Question 4 on Page 4-9 states that "If the containment is not closed". This should be modified to say that "If the containment is not *able to be* closed" in order to address the period where the containment is open and able to be reclosed prior to core damage.

4.14 Question 4 on Page 4-9 should also address the considerations necessary to accomplish containment reclosure. This includes the human reliability considerations, dependencies, and plant conditions.

4.15 Various abbreviations are used without definitions:

- CP is Figure 4.3
- CD on Page 4.2

Appendix A – Definition and Potential Specification of LERF

A.1 The definition of Large Early Release Frequency (LERF) is adequately presented in R.G. 1.174 as follows:

> *LERF is being used as a surrogate for the early fatality QHO. It is defined as the frequency of those accidents leading to significant, unmitigated releases from containment in a time frame prior to effective evacuation of the close-in population such that there is a potential for early health effects. Such accidents generally include unscrubbed releases associated with early containment failure at or shortly after vessel breach, containment bypass events, and loss of containment isolation. This definition is consistent with accident analyses used in the safety goal screening criteria discussed in the Commission's regulatory analysis guidelines.*

The ACRS has adequately supplemented this definition with some quantitative parameters that provide sufficient guidance to define LERF in a quantitative fashion. [1, 2]

The additional alternative definitions proposed in Appendix A of NUREG/CR-6595 Rev. 1 are not supported by reference calculations and are judged to be in conflict with the R.G. 1.174 and ACRS definition of LERF. The draft document used the magnitude of release of greater than 0.03 Iodine as the definition of "large release" for one of the alternatives. This is in conflict with the definition proposed by the ACRS. [1, 2] Specifically the magnitude definition chosen for the alternative case studies is conservative by more than 300%. Therefore, it is recommended that the ACRS endorsed definition of LERF and the one used in R.G. 1.174 be retained and the alternative definitions of LERF in NUREG/CR-6595 Rev. 1 be deleted.

The presentation of the three different definitions of LERF does not contribute to stability and consistency in the search for risk-informed inputs. The definition provided in RG 1.174 and quantified by ACRS [1, 2] represents a consistent methodology that has been widely adopted by individual utilities and should be adopted by the NRC. The introduction of an alternative methodology results in destabilization of the progress toward a risk-informed methodology.

A.2 The term "Early" is often used in the subject document; however, "Early" is not defined in the appendix describing the technical derivation of the LERF estimator.

A.3 P. A-1 correctly states that most plants have undertaken a Level 2 PRA. As such, what is the audience for NUREG/CR-6595? Why would plants with Level 2 PRAs (which is most if not all US plants) use the NUREG/CR-6595 approach instead of using their Level 2 PRA?

A.4 Appendix A refers to a study called the "LaSalle Independent Risk Assessment", but does not provide a reference. Is the LaSalle RMIEP or PRUREP study what is meant by this reference?

Appendix B – Case Studies

B.1 Table B-1, LERF Methods 2 and 3 as defined in Appendix B are based solely on magnitude of release and not timing (i.e., releases with Cs>0.1 occurring "late" would appear to be categorized as LERF by Method 3). This of course is completely inconsistent with the IPE development of a LERF definition.

B.2 Table B-1, the Method 3 value for Peach Bottom (2.7E-8) appears to be incorrect. The PB IPE value for High/Early releases alone is 5.68E-8.

B.3 Table B-1, the Method 3 value for Limerick (2.6E-8) appears to be incorrect. The Limerick IPE value for High/Early releases alone is 2.6E-8.

B.4 One of the key insights derived by BNL in the reassessment of NUREG/CR-6595 is stated in Appendix B.4:

Due to reduced decay heat, the probability of arresting the core damage before vessel breach could be significantly higher than that used for accident scenarios occurring during full power operation. Specific guidance on the recovery probability during shutdown therefore should be provided.

However, this insight is not included in the shutdown CETs developed as part of NUREG/CR-6595 Rev. 1.

B.5 A Lesson's Learned Recommendation in Section B.5 appears to have been ignored without justification.

The guidance in DRAFT DG-1061 was prepared for 5 containment types. Most of the top events in the event trees are similar but the guidance associated with the same top event in different event trees can vary from one tree to another. It is also confusing if the guidance provided for a top event refers to the guidance of the same top event for a different containment type (as was done in a few case in Draft DG-1061). In other cases, the same guidance was duplicated with the minor changes to account for the difference in containment types. For example, the guidance provided for the "reactor coolant system (RCS) depressurized" top event for the two PWR event trees are of different lengths. It is not obvious if the longer discussion provided for a dry containment is also applicable to an ice condenser containment. It is important to ensure that the guidance for each containment type has enough detail in itself without the need to have such cross references.

As noted in the Recommendations, this should be corrected in NUREG/CR-6595 Rev. 1.

REFERENCES

[1] Sherry, R., “Considerations for Plant-Specific Site-Specific Application of Safety Goals and Definition of Subsidiary Criteria,” Memorandum to ACRS Members, June 27, 1997.

[2] Kaiser, G. D., “The Implications of Reduced Source Terms for Ex-Plant Consequence Modeling,” ANS Executive Conference on the Ramifications of the Source Term, March 12, 1985, Charleston, SC.