January 31, 2011

MEMORANDUM TO: William F. Burton, Chief
Rulemaking and Guidance Development Branch
Division of New Reactor Licensing
Office of New Reactors

FROM: Earl R. Libby, Project Manager ***/RA/***
Rulemaking and Guidance Development Branch
Division of New Reactor Licensing
Office of New Reactors

SUBJECT: SUMMARY OF THE DECEMBER 2, 2010, PUBLIC MEETING TO DISCUSS EVALUATING PLANT CHANGES OR MODIFICATIONS TO DESIGN FEATURES RELIED UPON TO MITIGATE EX-VESSEL SEVERE ACCIDENTS

On December 2, 2010, the U.S. Nuclear Regulatory Commission (NRC) staff held a Category 3 public meeting with industry and Nuclear Energy Institute representatives. The purpose of the meeting was to discuss NRC staff proposals for evaluating plant changes or modifications to design features relied upon to mitigate ex-vessel severe accidents (EVSA) for reactors licensed under Part 52. The associated meeting notice is available at NRC's Agencywide Documents Access and Management System (ADAMS) accession number ML103130315. The following provides a brief summary of the meeting.

<u>Summary</u>

Mr. Earl R. Libby, Rulemaking and Guidance Development Branch (NRGA), Division of New Reactor Licensing (DNRL), Office of New Reactors (NRO), opened the meeting; Dr. Don Dube, Division of Safety Systems and Risk Assessment, and Mr. CJ Fong, Division of Construction Projects, lead the discussions throughout the day.

The participants discussed the need for a sub-section in the revised Nuclear Energy Institute, (NEI) 96-07 Appendix C guidance on the definition of common terms used including *previously reviewed* and *ex-vessel severe accidents*. For example, the participants discussed the scope of "ex-vessel" severe accidents and whether design features to address *induced steam generator tube rupture* (ISGTR) and *interfacing systems loss-of-coolant accident* (ISLOCA) are in-scope. Likewise, workshop participants questioned whether all combustible controls design features and some aspects of emergency operating procedures/severe accident management guidelines were in-scope. NRC staff will consider its positions on the issues.

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The participants discussed potential definitions of what might constitute a "substantial increase" in probability and public consequences of ex-vessel severe accidents. Qualitative and quantitative definitions were considered. For "substantial increase" in probability, there was general consensus on adopting a definition along the lines of the rule language for the Advanced Boiler Water Reactor (ABWR) design certification in Part 52, i.e.:

> *"A particular ex-vessel severe accident previously reviewed and determined to be not credible could become credible."*

The design control documents (DCDs) or supporting topical reports on severe accidents provide the source of evaluations on which design features have rendered potential severe accident containment challenges as *not credible,* although the staff emphasized that these discussions in the DCDs were not centrally located, and that often the terms *practically eliminated, not physically feasible,* and *not relevant* may have been used in lieu of *not credible*. It would be advantageous if each design center working group created a comprehensive list/table of these ex-vessel design features and the technical basis (quantitative and/or qualitative) for concluding why certain containment challenges from severe accidents are deemed not credible. This would assist COL holders in future screening of changes against the VIII.B.5.c criteria.

For increases in public consequences, there appeared to be consensus on the part of workshop participants to use surrogate measures for "public consequences" such as containment failure probability/frequency and fission product release fractions as opposed to offsite dose calculations. Dose calculations require a level-3 Probabilistic Risk Assessment (PRA), which is not required by the regulations for new reactors. A combination of qualitative and quantitative criteria for what constituted a "substantial increase" in public consequences seemed to be advantageous, e.g.: Remove or significantly degrade an ex-vessel severe accident mitigation design feature, or a combination of relative increase (e.g., order of magnitude) and contribution of the change to total containment failure probability/release fraction might suffice.

Workshop participants found value in the conceptual framework figure of slide 11, although it was recommended that the upper right block on containment bypass sequences be re-considered as discussed above, and that the EVSA functions include containment pressure control as well as fission product control.

NRC staff presented eight examples of what may or may not constitute "substantial increase" in probability and public consequences. Participants provided useful feedback on the examples, and staff committed to revising some of the examples for possible inclusion in the guidance document. Industry will propose other examples for consideration.

Workshop participants briefly discussed Problem Statement #3 related to the incorporation of risk-informed guidance in NEI 96-07 Appendix C. NRC staff expressed the desirability of having a comprehensive process diagram that encompassed all new reactor change processes. Public meetings on this task are planned to commence in January 2011.

NRC staff presented on Slide 31 a comprehensive list of the major programs and processes to be reviewed to determine whether any gap exists that might allow a "significant decrease in the enhanced level of safety" as discussed in SECY-10-0121. The highest priority should be given to 50.59, RG 1.174, risk-informed technical specifications, and risk-informed inservice inspection. There was consensus among meeting participants that existing processes do not appear to address changes to some non-ex-vessel severe accident mitigation design features.

An early spring 2011 time frame for a draft NEI 96-07 Appendix C addressing ex-vessel severe accident change processes was preliminarily proposed.

POST MEETING PRESENTATION CORRECTION NOTICE

Clarification
The following statement on slides 5 and 37 of the December 2, 2010 public meeting, derived from the SOC for the ABWR design certification rulemaking, should be deleted:

> **"It does not apply to those design features intended to meet design basis requirements and resolve severe accidents."**

The following SOC statement from the Federal Register Notice Vol. 62, No. 91 page 25806, dated May 12, 1997, is appropriate:

> **"In addition, the Commission is cognizant of certain design features that have intended functions to meet 'design basis' requirements and to resolve 'severe accidents.' These design features will be reviewed under either VIII.B.5.b or VIII.B.5.c depending upon the design function being changed."**

Enclosure:
As stated

cc w/encl: See next page

level of safety" as discussed in SECY-10-0121. Highest priority should be given to 50.59, RG 1.174, risk-informed technical specifications, and risk-informed inservice inspection. There was consensus amongst meeting participants that existing processes do not appear to address changes to some non-ex-vessel severe accident mitigation design features.

A late winter/early spring time frame for a draft NEI 96-07 Appendix C addressing ex-vessel severe accident change processes was preliminarily proposed.

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