June 22, 2015

MEMORANDUM TO: James W. Andersen, Deputy Director
Division of Preparedness and Response
Office of Nuclear Security and Incident Response

FROM: Joseph D. Anderson, Chief ***/RA/***
Operating Reactor Licensing and Outreach Branch
Division of Preparedness and Response
Office of Nuclear Security and Incident Response

SUBJECT: SUMMARY OF PUBLIC MEETING BETWEEN THE U.S. NUCLEAR REGULATORY COMMISSION, THE NUCLEAR ENERGY INSTITUTE, AND INDUSTRY ON EMERGENCY PREPAREDNESS FREQUENTLY ASKED QUESTIONS HELD ON MAY 28, 2015

The purpose of this Category 2 public meeting with industry representatives, held on May 28, 2015, was to discuss emergency preparedness frequently asked questions (EPFAQs) related to emergency action level scheme development guidance changes based upon potential issues and concerns related to proposed FLEX mitigation strategies. The Nuclear Energy Institute (NEI) discussed potential EPFAQs for possible submission at a later date.

The attached enclosure of proposed FAQs is a handout provided by NEI to the U.S. Nuclear Regulatory Commission (NRC) via email dated May 21, 2015 (Agencywide Document Access and Management System No. ML15142A426) to facilitate the discussion. The questions and proposed solutions were provided by NEI and do not necessarily reflect the NRC staff's position. These are drafts that have not been formally submitted for processing.

The staff reiterated the process that industry generated EPFAQs should be reviewed, compiled, screened, and submitted to the NRC by the NEI. However, this does not preclude anyone from submitting EPFAQs directly to the NRC via the emergency preparedness section of the NRC public website (http://www.nrc.gov/about-nrc/emerg-preparedness/faq/faq-contactus.html).

Members of the public were provided an opportunity to provide comments.

CONTACT: Oscar L. Aragon, NSIR/DPR
(301) 287-3790

Enclosures:
Meeting Attendees
Proposed EPFAQs

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Meeting Attendees

PUBLIC MEETING TO DISCUSS CHANGES TO THE EMERGENCY PLANNING FREQUENTLY ASKED QUESTIONS PROCESS

Tuesday, May 28, 2015 (8:30 a.m. – 12:30 p.m.)

Name	Affiliation (if any)
Joseph Anderson	U.S. NRC (NSIR/DPR)
Oscar Aragon	U.S. NRC (NSIR/DPR)
Patricia Borchmann	N/A
Raymond Hoffman	U.S. NRC (NSIR/DPR)
Don Johnson	U.S. NRC (NSIR/DPR)
Robert Kahler	U.S. NRC (NSIR/DPR)
Richard Kinard	U.S. NRC (NSIR/DPR)
Steve LaVie	U.S. NRC (NSIR/DPR)
Martin Vonk	EP Consulting, LLC
Michael Wasem	U.S. NRC (NSIR/DPR)
David Young	Nuclear Energy Institute
Charles Murray	U.S. NRC (NSIR/DPR)
Randolph Sullivan	U.S. NRC (NSIR/DPR)
Eric Bowman	U.S. NRC (NRR/JLD)
Jana Bergman	Curtiss-Wright / Scientech
Michael Lewis	PPL Susquehanna
Kenneth Klass	PPL Susquehanna
Eric Schrader	U.S. NRC (NSIR/DPR)
Larry Baker	Exelon
Mike Daus	Ciel Consultants
Ed Bates	E. Kimball Bates Services
Dave Stobaugh	EP Consulting, LLC
John (Marty) Crain	STPEGS
Gregory Richardson	EPM, Inc.
Kelly Walker	OSSI

Individuals representing the industry called into the meeting

Enclosure

NEI Issues and Proposed Resolutions – Draft

Question – The most recent update of the Boiling Water Reactor Owners Group (BWROG) Emergency Procedure Guidelines (R3) allows for limiting reactor pressure vessel (RPV) depressurization by reclosing the Safety Relief Valves (SRVs). How should this change be addressed vis-à-vis the NEI 99-01, Boiling Water Reactor (BWR) Fission Product Barrier (FPB) Table, RCS Barrier Loss threshold, #3 Reactor Coolant System (RCS) Leak Rate?

Answer – There is no impact to the threshold intent. The relationship between the operationally significant action and the RCS barrier status is unchanged, i.e., performing an emergency RPV depressurization per site-specific emergency operating procedure (EOPs) is indicative of a loss of the RCS barrier. Even though the SRVs may be reclosed, RCS mass has been lost to the suppression pool and subsequent depressurizations may be required (e.g., plant operators may reclose the SRVs after having initiated an Emergency RPV Depressurization to preserve steam-driven injection systems and complete the depressurization some time later). For clarity, the threshold basis should be revised to indicate that plant operators may reclose the SRVs following an emergency RPV depressurization.

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Question – The most recent update of the BWROG Severe Accident Guidelines (SAGs) (R3) changes the containment flooding strategy. As changed, this strategy would not be employed based solely on the inability to maintain or restore RPV water level. Rather, based on Fukushima OE, primary containment flooding would be directed when the core is ex-RPV or as a discretionary action. How should this change be addressed vis-à-vis NEI 99-01, BWR FPB Table, Primary Containment Potential Loss threshold, #2 Reactor Vessel (or RPV) Water Level?

Answer - This SAG change impacts the associated FPB threshold and basis. The emergency classification point has been moved from potential for/onset of core melt, the current threshold basis, to, under the most likely conditions, melting has occurred and the RPV has been breached (i.e., the corium has migrated to a location ex-RPV). The migration of corium to a location outside the RPV can be expected to present a significant challenge to primary containment integrity. The determination of a potential loss of primary containment should occur sooner.

This threshold, and the associated basis, should be changed to indicate that a potential loss of the primary containment has occurred if RPV water level cannot be restored and maintained above the minimum steam cooling reactor water level. The inability to maintain RPV water level above the minimum steam cooling reactor water level places the plant on a trajectory for core melt and a subsequent challenge to the primary containment (i.e., it represents a potential loss). This change also impacts Fuel Clad Barrier Loss threshold Reactor Vessel (RPV) Water Level #2.A. That threshold should also be revised to read the same as the containment potential loss threshold. The associated basis should also be revised as needed.

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Question – With respect to the NEI 99-01, BWR FPB Table, Containment Loss threshold #3, Primary Containment Isolation Failure or Bypass (which involves a failure of all valves in any one line to close and a direct downstream pathway to the environment exists after primary containment isolation signal), should a release through the wetwell be considered a direct release path?

Answer – Yes; within the context of this threshold, a release through the wetwell is a "direct" release path. The answer reflects consideration of the large amount of noble gases that could be released if there were a failure to isolate primary containment. Wetwell "scrubbing" of the release would not affect the noble gas concentration.

Question - The most recent update of the BWROG Emergency Procedure Guidelines (R3) allows for anticipatory venting to address conditions other than those associated with an immediate challenge to primary containment resulting from high pressure (i.e., pressure at the drywell design limit) or combustible gas reaching a deflagration concentration. For example, venting may be performed early to address an adverse trend in primary containment pressure. How should this change be addressed vis-à-vis the NEI 99-01, BWR FPB Table, Containment Loss threshold #3, Primary Containment Isolation Failure or Bypass?

Answer – There is no impact to the threshold intent. The relationship between the operationally significant action and the containment barrier status is unchanged, i.e., conditions have degraded to the point that the control room staff has made a decision to perform a controlled (intentional) venting of the containment, whether for anticipated or immediate reasons. This venting action results in a bypass of the primary containment. For clarity, the threshold and basis should be revised to indicate that a loss of the containment barrier occurs when there is a controlled venting of primary containment (due to this action causing a bypass of the primary containment).

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Question – Concerning NEI 99-01 IC SG2 (R4/R5) or IC SS5 (R6), should the EALs also address steam flow above the minimum core steam flow as an alternate strategy to achieve adequate core cooling?

Answer – No; core steam flow cooling would be employed only if RPV water level cannot be restored and maintained above the top of the active fuel, cannot be determined, or must be intentionally lowered below the top of the active fuel. It is relied upon as a contingency core cooling method, and its use and effectiveness is subject to a number of factors. During an anticipated transient without scram, the fact that minimum steam cooling RPV water level cannot be maintained is sufficient to meet the EAL criterion that core cooling is extremely challenged.

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Question - NEI 99-01 R6 contains the following Developer Note guidance for ICs CU2, CA2, SA1 and SS1:

> “The EAL and/or basis section may specify use of a non-safety-related power source provided that operation of this source is recognized in AOPs and EOPS, or beyond design basis accident response guidelines (e.g., FLEX support guidelines). Such power sources should generally meet the “Alternate ac source” definition provided in 10 CFR 50.2.”

FLEX support guidelines are not mentioned in earlier revisions of NEI 99-01 or NUMARC-007. Plants have added, or are in the process of adding, new FLEX capabilities in response to NRC Order EA-12-049. These capabilities will allow a plant to maintain or restore key safety functions for an indefinite period of time following an extended loss of AC power. Should EALs or Bases be revised to recognize/credit FLEX capabilities (e.g., a plant now has the ability to re-energize a bus from a FLEX generator)?

Answer – Consistent with the developer note guidance cited above, a FLEX power source may be reflected in an EAL and/or basis if the source meets the “Alternate ac power source” definition criteria in 10 CFR 50.2. Beyond that allowance, no other FLEX equipment should be recognized/credited in any EAL or basis. This answer reflects the conditions under which FLEX equipment would most likely be utilized - a beyond-design-basis event of sufficient magnitude to render all installed AC power sources unavailable for an extended period of time concurrent with a loss of normal access to the normal heat sink. Such an extreme event can be expected cause extensive plant damage and likely degradation of infrastructure in the local plant environs (e.g., roads and bridges).