March 5, 2013

MEMORANDUM TO: Doug Coe, Acting Director
Division of Risk Analysis
Office of Nuclear Regulatory Research

THRU: Kevin Coyne, Chief ***/RA/***
Probabilistic Risk Assessment Branch
Division of Risk Analysis
Office of Nuclear Regulatory Research

FROM: Donald M. Helton ***/RA/***
Probabilistic Risk Assessment Branch
Division of Risk Analysis
Office of Nuclear Regulatory Research

SUBJECT: SUMMARY OF THE FOURTH CLOSED MEETING TO DISCUSS ONGOING OFFICE OF NUCLEAR REGULATORY RESEARCH CONFIRMATORY LEVEL 1 PROBABILISTIC RISK ASSESSMENT SUCCESS CRITERIA ANALYSES

On January 24th, 2013, NRC staff held a noticed, closed teleconference with Exelon/Byron station to discuss ongoing Office of Nuclear Regulatory Research confirmatory Level 1 probabilistic risk assessment (PRA) success criteria analyses. During the teleconference, participants discussed boundary conditions and qualitative results for sequence timing and success criteria aspects of selected medium loss-of-coolant accident scenarios, as well as boundary conditions for ongoing calculations related to loss of decay heat removal accidents in shutdown modes 4 and 5. These discussions (and the associated analyses) do not relate to any ongoing or anticipated regulatory actions; rather, they are to confirm specific underlying modeling aspects in the agency's standardized plant analysis risk models for 4-loop Westinghouse plants with large, dry containments (a continuation of an activity described further in NUREG-1953, "Confirmatory Thermal-Hydraulic Analysis to Support Specific Success Criteria in the Standardized Plant Analysis Risk Models – Surry and Peach Bottom").

This meeting was closed because (a) there has been no public interest expressed in past public activities related to this work, (b) the meeting is an information exchange not related to any specific regulatory decision, and (c) closure of the meeting facilitates the discussion of facility details (e.g., emergency operating procedures) that are not in the public domain. Should a member of the public wish to participate in any future discussions between NRC and Exelon associated with this project, they should contact Donald Helton, Senior Reliability and Risk Engineer, at 301-251-7594 or at Donald.Helton@nrc.gov.

CONTACT: Donald Helton, RES/DRA
301-251-7594

At the beginning of the call NRC staff highlighted the impending issuance of a draft NUREG/CR that investigates PRA end-state definition and success criteria modeling issues, which includes calculations performed using the NRC's Byron MELCOR model, amongst others. NRC staff described this separate, but related work, to minimize confusion about the relationship between that NUREG/CR and the NUREG report under development that was the main subject of the meeting. Note that, subsequent to the call, the expected issuance date for the draft NUREG/CR has changed from February 2013 to March 2013.

With regard to the medium loss-of-coolant analyses, two distinct scenarios were discussed. The first set investigated the minimal emergency core cooling system (ECCS) injection requirements for early injection considering different break sizes (ranging from 2- to 6-inch (5 to 15 cm) equivalent diameter), different minimal emergency core cooling system availabilities (1 safety injection plus 1 residual heat removal train; 2 of 4 cold-leg accumulators plus 1 residual heat removal train), different assumptions about containment spray availability (0 of 2 trains available; 2 of 2 trains available during the injection phase only; 2 of 2 trains available during the injection and recirculation phases), and variations on whether the residual heat removal heat exchanger is available. The preliminary key takeaways from these analyses are:

- One safety injection and 1 residual heat removal train were successful in preventing core damage during the early injection phase. Core damage eventually occurred for 2-inch (5 cm) breaks due to the lack of high pressure recirculation, while the other cases did not have core damage.

- Two accumulators and 1 residual heat removal train were not successful in preventing core damage for the 2-inch (5 cm) and 3.33-inch (8.5 cm) breaks. Core damage occurred before the accumulators began injecting. There was no core damage for the 4.67-inch (12 cm) and 6-inch (15 cm) cases.

- Containment heat removal systems had a negligible impact on the injection phase of the accident and only a limited impact during the later stages of the accident.

The second set of medium loss-of-coolant accident calculations investigated the timing to initiate cooldown activities in order to obviate the need for high-pressure recirculation, considering different break sizes (2- or 6-inch (5 or 15 cm) equivalent-diameter), cooldown initiation times (20 or 40 minutes), containment system availabilities (2 containment spray trains plus 0 reactor containment fan cooler (RCFC) units; 0 containment spray trains plus 1 RCFC unit; 4 RCFC units), and variations related to operator response upon observing temporarily rising reactor coolant system pressures. The preliminary key takeaways from these analyses are:

- For the 2-inch (5 cm) breaks, if operators stop the cooldown in response to rising cold leg temperatures, then core damage will occur regardless of the time of cooldown initiation (for the times studied) or the available containment systems. This is not the anticipated operator behavior, and calculations where cooldown is halted will be characterized as sensitivities.

- Otherwise, core damage is prevented. Continued cooldown eventually maintains reactor coolant system pressure below the residual heat removal pump shutoff head for 2-inch (5 cm) cases, while a 6-inch (15 cm) break is sufficient to keep the reactor coolant system pressure low enough to allow low-pressure recirculation to keep the core covered.

- Cooldown timing had no impact (for the timings studied) on whether or not core damage occurred, but it did affect the time of key events, particularly the time of refueling water storage tank depletion and the time of core damage (if applicable). The same was true of the availability of containment heat removal systems.

Once the thermal-hydraulic analyses have been reviewed and published, they will be evaluated for potential refinement of the standardized plant analysis risk modeling assumptions for Byron and other plants with similar design and operational characteristics. The specific modeling assumptions made for a given NRC application remain at the discretion of the cognizant NRC risk analysts.

With regard to the boundary conditions for ongoing calculations for loss-of-decay heat removal during shutdown operations, two sets of calculations were discussed. The calculation sets are similar in scope, addressing the specific system configurations, operational characteristics, and recovery actions associated with mode 4 and mode 5.

NRC staff highlighted some open questions from previous calls, as well as a desire for confirmation of some of the operational assumptions being made in the shutdown calculations. NRC followed up with a subsequent voluntary information request on January 29th, 2013.

The following people participated in the teleconference:

NRC		Exelon / Erin Engineering
James Corson	Margaret Tobin	Patrick Bozym
Donald Helton	Jeffery Wood	Andrew Dercher
Laura Kozak		

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OFFICE	RES/DRA/PRAB	RES/DRA/PRAB	RES/DRA
NAME	D. Helton	K. Coyne	D. Coe
DATE	2/26/13	2/1/13	3/5/13

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