UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 15, 2010

LICENSEE: Energy Northwest

FACILITY: Columbia Generating Station

SUBJECT: SUMMARY OF TELEPHONE CONFERENCE CALL HELD ON JUNE 28, 2010, BETWEEN THE U.S. NUCLEAR REGULATORY COMMISSION AND ENERGY NORTHWEST CONCERNING DRAFT REQUESTS FOR ADDITIONAL INFORMATION PERTAINING TO THE SEVERE ACCIDENT MITIGATION ALTERNATIVES REVIEW OF THE COLUMBIA GENERATING STATION LICENSE RENEWAL APPLICATION

The U.S. Nuclear Regulatory Commission (NRC or the staff) and representatives of Energy Northwest (the applicant) held a telephone conference call on June 28, 2010, to discuss and clarify the staff's draft requests for additional information (D-RAIs) concerning the Severe Accident Mitigation Alternatives (SAMA) review of the Columbia Generating Station license renewal application. The telephone conference call was useful in clarifying the intent of the staff's D-RAIs.

Enclosure 1 provides a listing of the participants and Enclosure 2 contains a listing of the D-RAIs discussed with the applicant, including a brief description of the status of the items.

The applicant had an opportunity to comment on this summary.

Daniel I. Doyle, Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosures:
1. List of Participants
2. List of Draft Requests for Additional Information

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LIST OF PARTICIPANTS
COLUMBIA GENERATING STATION
LICENSE RENEWAL APPLICATION–SAMA REVIEW
TELEPHONE CONFERENCE CALL

JUNE 28, 2010

PARTICIPANTS	AFFILIATIONS
Daniel Doyle	U.S. Nuclear Regulatory Commission (NRC)
Tina Ghosh	NRC
Ray Gallucci	NRC
Steve Short	Pacific Northwest National Laboratory (PNNL)
Garill Coles	PNNL
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Abbas A. Mostala	Energy Northwest (EN)
Marsha G. Eades	EN
Jerral E. (Dusty) Rhoads	EN
Albert T. Chiang	EN
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Stanley Levinson	AREVA
Jennifer Butler	AREVA
James Metcalf	AREVA
Darvin Kapitz	AREVA
Eric J. Jorgenson	Maracor

ENCLOSURE 1

DRAFT REQUESTS FOR ADDITIONAL INFORMATION
COLUMBIA GENERATING STATION
LICENSE RENEWAL APPLICATION SAMA REVIEW

JUNE 28, 2010

The U.S. Nuclear Regulatory Commission (NRC or the staff) and representatives of the applicant, Energy Northwest (EN), held a telephone conference call on June 28, 2010, to discuss and clarify the following draft requests for additional information (D-RAIs) concerning the Severe Accident Mitigation Alternatives (SAMA) review of the Columbia Generating Station (CGS) license renewal application.

D-RAI 1a

Provide the following information regarding the Level 1 Probabilistic Safety Assessment (PSA) used for the SAMA analysis: Environmental Report (ER) Section E.3 explains that the SAMA evaluation is based on PSA Revision 6.2 models for Level 1 and 2 internal, fire, and seismic events. Table E.3-1 shows the completion dates for these different models to range from 1/2004 to 2/2007 and shows these models to be based on incorporation of plant modifications that occurred up through 8/2006 and CGS data/Bayesian updating through 6/2002. Identify any changes to the plant (physical and procedural modifications) since those dates that could have a significant impact on the results of the SAMA analyses, and provide a qualitative assessment of their impact on the PSA and on the results of the SAMA evaluation. Identify whether a newer PSA model is available, and if so, provide a brief description of the major changes relative to the PSA Revision 6.2, and provide an assessment of the impact on the results of the SAMA evaluation (e.g., increased benefit or additional SAMAs if the baseline core damage frequency (CDF) has increased; any new candidate SAMAs for newly-identified dominant sequences or risk-significant basic events).

Discussion:

The applicant noted that the PRA of record was Revision 6.2 and acknowledged that an upgraded PRA to RG 1.200 Rev 2 was recently available. The applicant was concerned with how that information could be used for the docketed SAMA analysis which was performed with Revision 6.2. NRC expressed concern that the insights from Revision 7.1 may lead to additional potentially cost beneficial SAMAs. The NRC indicated that the information from Revision 7.1 would be needed for validating the docketed SAMA analysis and some of the qualitative statements on the effect of plant changes that occurred after Revision 6.2.

D-RAI 1b

Provide the following information regarding the Level 1 PSA used for the SAMA analysis: ER Table E.3-3 shows the contribution of specific types of Transients and loss of coolant accidents (LOCAs) to the CDF and identifies specific initiators leading to anticipated transient without scram. In light of the fact that station blackout (SBO) contributes 33.1% to the CDF, identify the initiators that contribute to SBO.

Discussion:

The applicant asked for a clarification of the request to identify "initiators" that contribute to the SBO. The NRC reviewers responded that they were looking for the loss of offsite power (LOOP) contributors that comprise the grid, weather, and internal contributors to LOOP. The applicant stated that this was sufficient clarification.

D-RAI 1ci-iii

Provide the following information regarding the Level 1 PSA used for the SAMA analysis:

i) A summary of the scope of the 1997 owner's group peer review mentioned in Section E.5.5.

ii) A brief description of all unresolved B Level facts and observations (F&Os) from the 2004 internal events PSA peer review discussed in Section E.5.5 and an assessment of their impact on the SAMA evaluation.

iii) A summary of the scope of the other two peer reviews and the seven technical reviews discussed in Section E.5.2, a brief description of all unresolved issues/F&Os, and an assessment of their impact on the SAMA evaluation.

Discussion:

The applicant noted that (c)(ii) asks for only a discussion on unresolved LEVEL B F&Os, whereas (c)(iii) asks for a brief description of all unresolved issues/F&Os and that (c)(i) asks for only a summary of the scope of the Boiling-Water Reactor Owners Group (BWROG) 1997 peer review. The applicant asked if the NRC is asking for only the significant unresolved F&Os from the various peer reviews and technical assessments or for all of the F&Os and if, for the 1997 peer review, only a scope of the review was required. The NRC reviewer indicated that he is most interested in significant F&Os that were unresolved from these peer reviews and technical assessments that might impact the SAMA evaluations or potentially create new ones. The reviewer also noted that the expectation is that any unresolved F&Os from the 1997 peer review will have been addressed in the most recent peer review and, if not, need to be specifically discussed.

D-RAI 1e

Provide the following information regarding the Level 1 PSA used for the SAMA analysis: ER Table E.3-2 presents the truncation limits used when quantifying the Internal Events and Fire PSA fault trees to be 1E-8 to 1E-14 with a footnote explaining that "the truncation limit was adjusted to assure sufficient capture of the contributing basic events." The meaning of and need for different truncation limits for fault trees, event trees and a category referred to as "Global" is not clear. Explain the basis for the truncation limits selected and the meaning of the entries shown in Table E.3-2. Clarify for which cases a truncation limit of 1E-8 versus a truncation limit of 1E-14 was used.

Discussion:

The applicant asked for additional detail regarding the request regarding truncation limits. The reviewers stated that they were not familiar with the use of specific and global truncation limits. The applicant responded that this was a feature in the WinNUPRA program that allowed selective application of truncation limits to fault trees, event trees and to the overall equation. Additional information was provided by Eric Jorgenson of Maracor on the use of these in the quantification module and how sensitivity evaluations were performed to assure these selective truncation limits did not result in loss of potential contributors from the fault trees and events trees. The NRC reviewer emphasized that additional information was needed to understand how the range of different truncation limits was applied and the basis for selecting the limit. The applicant agreed to provide more information on the WinNUPRA models quantification and the Revision 6.2 sensitivity studies that validate that they do not truncate important contributors to the solution.

D-RAI 2b

Provide the following information relative to the Level 2 analysis: ER Section E.4.1.1 states that containment event trees (CETs) were developed for each plant damage state (PDS) and that quantification of the CETs was supported by fault tree analysis and assignment of split fractions. Clarify how probabilities were assigned to branches using split fractions for branches. In the response, specifically address how split fractions were developed for phenomenological branch points.

Discussion:

The applicant asked for clarification about the request for information regarding the development of the split fractions and especially how split fraction were developed for phenomenological branch points. The NRC reviewers responded that although the values were provided in the submittal for the split fractions, they were looking for additional information on how these were determined.

D-RAI 2c

Provide the following information relative to the Level 2 analysis: ER Section E.5.5.1 lists peer review findings and other self-identified areas that are in progress for the next revision, and characterizes them as not expected to significantly alter the SAMA analysis findings. Yet, a number of the recommendations address non-conservatism in the Level 1 and 2 PSA model, including:

- upgrading the LOCA outside containment modeling;
- refining the ISLOCA modeling;
- incorporating an initiating event for CCF of both 125-VDC power divisions;
- refining the impact of spray on equipment, the RCIC pump flood damage height, and flood isolation HEPs;
- including certain early phenomena that can lead to LERF;
- revising crew actions included in the LERF assessment;

- accounting for potential environmental impacts in the survivability of systems for Level 2 mitigation;
- reconsidering inclusion of source term scrubbing for non-LERF end states having no MAAP calculation.

Justify the conclusion that the unresolved findings are not expected to significantly alter the results of the SAMA analysis.

Discussion:

The applicant asked what additional justification would be acceptable to the NRC for those peer review findings and other self-identified areas that are in progress for the next revision. The applicant indicated that: (1) additional qualitative discussion of each item listed could be provided, (2) Revision 6.2 could be modified to bound the item, or (3) potentially this might be a use for Revision 7.1 as a sensitivity study against Revision 6.2 as these issues have been incorporated into Revision 7.1. The NRC replied that of the three options, the preferred one would be the third option.

D-RAI 2f

Provide the following information relative to the Level 2 analysis: The ER does not provide an importance list of either Level 1 or Level 2 basic events and so it is not possible to ascertain the significance of recovery events or operator actions in the PSAs. Discuss the extent to which recovery of systems or operator actions following the onset of core damage is credited in the Level 2 assessment and how recovery is modeled.

Discussion:

The applicant explained that Revision 6.2 could generate an importance list of basic events for level 1 (CDF) but, due to the structure of the interface between Level 1 and Level 2 (LERF), the importance at the basic events did not propagate through the Containment Event trees, and, therefore, the traditional Fussell-Vesely importance listing for level 2 basic events could not be provided. The applicant informed the NRC that they attempted to provide this information on Level 2 importance in narrative form in the submittal in section E9.5. The applicant stated that this was determined through inspection of those plant damage states that contribute most to LERF and identifying the major accident progression sequences and their important basic events. But Revision 6.2 would not generate the classic importance listing that the NRC was asking for in this question (and later in question 5c). The NRC reviewers emphasized that additional detail at the Level 2 basic event or cutset level was needed to conclude that the importance contributors had been considered. The NRC asked if Revision 7.1 would generate the importance. The applicant responded that it can and a sensitivity evaluation could be done to confirm the information in the level 2 importance discussion in the E9.5 section of the submittal. If additional basic events are identified in the sensitivity analysis, their consideration for additional SAMA candidates could be assessed.

D-RAI 3a

Provide the following information with regard to the treatment and inclusion of external events in the SAMA analysis: ER Section E.3.2.2 states that the seismic hazard analysis used for the seismic PSA is the same as submitted for the CGS Individual Plant Examination of External Events (IPEEE) except for an extrapolation from the maximum peak ground acceleration to 1.5g. The seismic hazard analysis used for the IPEEE was developed in 1994 and documented in "Probabilistic Seismic Hazard Analysis WNP-2 Nuclear Power Plant Hanford Washington." Justify the use of the seismic PSA model given: (1) since then the U.S. Geological Survey (USGS) has updated its assessment of seismic hazards across the U.S., including Washington State, (2) seismic hazard analysis was performed specifically for the Hanford area in 1994 which is documented in WHC-SD-W236A-TI-016, Seismic Design Spectra 200 West and East Areas DOE Hanford Site, Washington," to provide better evaluation of subsurface materials and (3) work was performed in 2005 which is documented in PNNL-15089, "Site-Specific Seismic Response Model for the Waste Treatment Plant, Hanford Washington" that better characterizes the effect from deep layers of sediments "interbedded" with basalt. Address whether consideration of the more current seismic hazard analysis could impact the results of the SAMA analysis (both SAMA identification and SAMA evaluation).

Discussion:

The applicant expressed concern that impact to the seismic hazard curve by newer geological studies conducted by others for the region and a facility about 11 miles away from CGS could not be addressed quantitatively. The applicant asked if the intent of the question was to have the applicant review those other studies and to provide a qualitative response. The NRC was concerned that the seismic hazard study was based on techniques and methodology from about 15 years ago and was concerned that recent studies might cause additional SAMA considerations for the seismic hazard that the PRA would not identify. The applicant responded that the Waste Treatment Plant study had been quickly reviewed and it was noted that, in the report, the primary geological features that caused higher response spectra in the 2-10 hertz region were layers of basalt and silt structures. In review of CGS core drills, these kinds of structures were not identified to be prominent. The applicant asked if a discussion on that level would be of value to the reviewers for this question. NRC reviewers responded that it could be useful.

D-RAI 6a

Provide the following with regard to the Phase II cost-benefit evaluations: Section E.8 provides the basis and results for the total averted cost, or maximum benefit, from internal events. Provide the same information (i.e., APE, AOC, AOE, AOSC) for the total averted cost from fire and seismic events.

Discussion:

The applicant noted that the APE, AOC, AOE, and AOSC for fire and seismic were located in Tables E.11-3 and E.11-4. The reviewer agreed and withdrew the question. Items 6b-k will be renumbered 6a-j.

D-RAI 6j

Provide the following with regard to the Phase II cost-benefit evaluations: ER Section E.12 discusses six sensitivity cases. Insufficient information is provided to understand the specific changes made to the baseline analysis assumptions for Cases 1 and 6. Provide a more detailed description of the analysis assumptions and methodology for these two cases.

Discussion:

The applicant asked for clarification on what additional information the NRC was looking for related to Sensitivity Cases 1 and 6. For Sensitivity Case 1 the NRC is looking for how repaired and refurbished equipment was credited in the cost-benefit analysis. For Sensitivity Case 6 the NRC is looking for what escalation rate was used to determine the increase in replacement power costs.

D-RAI 6k

Provide the following with regard to the Phase II cost-benefit evaluations: ER Section E.12 states that "no explicit uncertainty was performed since the number of conservative assumptions and input account for any uncertainties in the calculations," and goes on to delineate several sources of conservatism in the SAMA analysis. This is not consistent with NEI 05-01 wherein the delineation of conservatisms is used to offset an uncertainty factor based on the 95th percentile CDF. Given that the 95th percentile values are typically a factor of two to five higher than point estimates, identify and provide a further evaluation of those SAMAs that are within a factor of two to five of being cost-beneficial. This evaluation can be based on more realistic estimates of risk reduction and implementation costs, and deterministic considerations, including potential negative implications of candidate SAMAs.

Discussion:

The applicant asked if a qualitative discussion would be an acceptable response. The NRC clarified the question and stated that a quantitative analysis is needed. The quantitative analysis should determine the impact that the variability in the CDF has on the maximum benefit.

July 15, 2010

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Memorandum to Energy Northwest from Daniel I. Doyle dated July 15, 2010

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