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UniStar
NUCLEAR ENERGY

10 CFR 50.4
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August 19, 2011

UN#11-235

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: UniStar Nuclear Energy, NRC Docket No. 52-016
Response to Request for Additional Information for the
Calvert Cliffs Nuclear Power Plant, Unit 3,
RAI 306, Reliability Assurance Program

References:
1) James Steckel (NRC) to Robert Poche (UniStar Nuclear Energy), "FINAL RAI 306 SPRA 5606," email dated June 1, 2011

2) UniStar Nuclear Energy Letter UN#11-199, from Greg Gibson to Document Control Desk, U.S. NRC, Response to Request for Additional Information 306, Reliability Assurance Program, dated June 30, 2011

The purpose of this letter is to respond to the request for additional information (RAI) identified in the NRC e-mail correspondence to UniStar Nuclear Energy, dated June 1, 2011 (Reference 1). This RAI addresses the Reliability Assurance Program, as discussed in Section 17.4 of the Final Safety Analysis Report (FSAR), as submitted in Part 2 of the Calvert Cliffs Nuclear Power Plant (CCNPP) Unit 3 Combined License Application (COLA), Revision 7.

The Enclosure provides our responses to RAI No. 306, Questions 17.04-10 and 17.04-12, and includes revised COLA content. A Licensing Basis Document Change Request has been initiated to incorporate these changes into a future revision of the COLA.

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Reference 2 stated that a schedule for RAI 306, Question 17.04-11 would be provided by August 19, 2011. A response to RAI 306, Question 17.04-11 will be provided to the NRC by November 11, 2011.

There are no regulatory commitments identified in this letter. This letter does not contain any sensitive or proprietary information.

If there are any questions regarding this transmittal, please contact me at (410) 470-4205, or Mr. Wayne A. Massie at (410) 470-5503.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on August 19, 2011

Greg Gibson

Enclosure: Response to NRC Request for Additional Information RAI No. 306, Questions 17.04-10 and 17.04-12, Reliability Assurance Program, Calvert Cliffs Nuclear Power Plant, Unit 3

cc: Surinder Arora, NRC Project Manager, U.S. EPR Projects Branch
Laura Quinn, NRC Environmental Project Manager, U.S. EPR COL Application
Getachew Tesfaye, NRC Project Manager, U.S. EPR DC Application (w/o enclosure)
Charles Casto, Deputy Regional Administrator, NRC Region II (w/o enclosure)
Silas Kennedy, U.S. NRC Resident Inspector, CCNPP, Units 1 and 2
U.S. NRC Region I Office

Enclosure

Response to NRC Request for Additional Information

RAI No. 306, Questions 17.04-10 and 17.04-12, Reliability Assurance Program
Calvert Cliffs Nuclear Power Plant, Unit 3

RAI No. 306

Question 17.04-10

Follow-up to Question 17.04-5

Since the RAP structures, systems, and components (SSCs) are subjected to QA controls and the system boundary is one of the key sources of information for D-RAP ITAAC closure, therefore, revise the final safety analysis report (FSAR) to clearly provide a common basis for defining and understanding the system boundaries and the associated interfaces as briefly discussed in the response to Question 17.04-5.

Response

The response to Calvert Cliffs Nuclear Power Plant (CCNPP) Unit 3 RAI 224, Question 17.04-5[1], sixth paragraph, provided a comprehensive definition of U.S. EPR system boundaries and interfaces. The definition and interfaces will be incorporated into FSAR Section 17.4 in a future revision of the COLA.

COLA Impact

CCNPP Unit 3 FSAR Section 17.4 will be supplemented to include new Subsection 17.4.4.1.2.6, "Identification of SSC Boundaries and Interfaces," in a future revision of the COLA, as shown below:

17.4.4.1.2.6 Identification of SSC Boundaries and Interfaces

Plant components have a unique tag number, including designators that group these components by function. The determination of these groupings and identifiers is accomplished through the application of engineering conventions to standardize the approach and to ensure that system interfaces are consistently and completely identified. As an example, heat exchangers are generally included as part of the higher pressure system.

Instrumentation sensors are included as part of the mechanical system they are a part of (as opposed to being included in I&C systems that the sensors are connected to). These groupings are referred to as systems. As such, each system has system boundaries that are defined as the point of demarcation at interface points where the physical transition is made from one system to another. A typical example of these system boundary points is the isolation valve between two mechanical systems.

In the U.S. EPR design, system boundaries are shown on piping and instrumentation diagrams (P&IDs) and can be clearly identified by making reference to the tag numbers for the components. These system boundaries and associated interfaces are described in detailed design documents.

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[1] Greg Gibson (UniStar Nuclear Energy) to Document Control Desk (NRC), Letter UN#10-094, "RAI 224, Reliability Assurance Program, and Clarification of the Response to RAI 194, Question 17.04-4," dated April 16, 2010 (ML101090293)

Question 17.04-12

Follow-up to Question 17.04-6

The response to Question 17.04-6 states that, "The selection criteria used to establish an expert panel for the RAP to screen SSCs is found in ASME RA-Sb-2005, Section 6, which provides general information on the selection process and personnel qualifications ..." The staff finds that the criteria in Section 6 of ASME RA-Sb-2005 are not all suitable to make up the D-RAP expert panel since they were intentionally developed to establish a team to review a PRA, focusing on PRA models, database, assumptions, etc.

In the light of industry guidelines such as NEI 00-04 on Integrated Decision-making Panel, please provide the rationale for the criteria used for selecting the D-RAP panel and justify that a panel of 3 individual with five years experience would be sufficient to review the results of the initial risk-significance determinations and finalize the D-RAP list.

Response

There is regulatory precedence, as shown below, for the statement in CCNPP Unit 3 FSAR Section 17.4.4.1.3 that, as a minimum, the combined expert panel and working groups include at least three individuals with a minimum of five years experience at similar nuclear plants:

- The NRC Advanced Final Safety Evaluation for Chapter 17 of the South Texas Project Combined License Application (Accession Number ML110280134) states:

  > FSAR Section 17.4S.1 describes the use of an expert panel to identify risk-significant SSCs that are not modeled in the PRA to augment PRA techniques in ranking the risk of SSCs using deterministic techniques, operating experience, and expert judgment and to act as a final approver of risk-significant SSCs. FSAR Subsection 17.4S.1.3 describes the qualification requirements for members of the expert panel. The expert panel and designated working group (or groups) consists of designated individuals with expertise in the areas of risk assessment, operations, maintenance, engineering, QA, and licensing. At a minimum, the combined expert panel and working group(s) should include at least three individuals with a minimum of 5 years of experience at the STP or at a similar nuclear plant. There should also be at least one individual who has worked on modeling and updating the PRA for the STP or at a similar plant for a minimum of 3 years. When utilized, expert panel representatives from contracting design organizations are required to have a minimum of 3 years of experience establishing risk rankings for nuclear plant.

- Draft NRC Inspection Procedure 37060 (Accession Number ML110480844), Section 02.02(g) states:

  > The IDP [independent decision-making panel] must be composed of experienced personnel who possess diverse knowledge and insights in plant design and operation, and who are capable in the use of deterministic knowledge and risk insights in making SSC classifications. At least three members of the IDP should have a minimum of 5 years experience at the

plant, and there should be at least one member of the IDP who has worked on the modeling and updating of the plant specific PRA for a minimum of 3 years.

It is also noted that, as shown in the U.S. EPR FSAR, Tier 2, Table 1.9-2, that Regulatory Guide 1.201 (issued for trial use by the NRC and which endorses NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline") does not apply to the U.S. EPR. The U.S. EPR design is "deterministic" and does not invoke 10 CFR 50.69 to risk-classify SSC and does not implement risk-based programs (e.g., IST, ISI, Fire Protection). As noted in U.S. EPR FSAR, Tier 2, Section 19.1.1.1, "The PRA is not used for any formal risk-informed applications, such as 10CFR50.69, Risk-Informed Categorization and Treatment of structures, systems and components (SSC) and 10CFR50.48, Fire Protection."

However, the U.S. EPR design plans to utilize the suggested composition requirements for an Integrated Decision-making Panel described in NEI 00-04, with the composition requirements described in ASME RA-Sb-2005 Section 6.

Section 9.1 of NEI 00-04 provides the recommended knowledge and experience makeup of the IDP which may also be satisfied by requirements for panel members established in ASME RA-Sb-2005, Section 6. CCNPP Unit 3 FSAR Section 17.4.4.1.3 is conservative with regards to the criteria of ASME RA-Sb-2005 and requires expert panel members with expertise in the areas of risk assessment, operations, maintenance, engineering, quality assurance, and licensing.

NEI 00-04 goes further by suggesting a minimum number of expert panel members of at least five persons and states that the precise makeup of the panel is up to the licensee. CCNPP Unit 3 FSAR Section 17.4.4.1.3 states that "at least three individuals with a minimum of five years experience at similar nuclear plants, and at least one individual who has worked on the modeling and updating of the PRA for similar plants for a minimum of three years."

Consistent with the NEI 00-04 recommendation, FSAR Section 17.4 will be revised to reflect the industry guidance suggesting a minimum number of DRAP expert panel members of at least five persons.

COLA Impact

CCNPP Unit 3 FSAR Section 17.4.4.1 will be revised in a future revision of the COLA, as shown below:

17.4.4.1 Identification of Site-Specific SSCs for D-RAP

Section 17.4.2 describes a methodology for ensuring site-specific SSCs are identified and included in the RAP.

The initial list of site-specific SSCs and their risk rankings are included in Section 17.4.2. The PRA model will continue to be refined over the life of the plant and this will require periodic adjustment to the risk rankings of SSCs in Section 17.4.2.

As D-RAP enters the detailed design, procurement, fabrication and construction phase, an expert panel with {Calvert Cliffs 3 Nuclear Project, LLC and UniStar Nuclear Operating Services, LLC} representation will be established and utilized to:

- Augment PRA techniques in the risk ranking of SSCs using deterministic techniques, operating experience and expert judgment.
- Identify risk significant SSCs not modeled in the PRA (if any).
- Act as the final approver of risk significant SSCs.
- Recommend design changes where appropriate to reduce risk.
- Revise/adjust recommend operations phase maintenance/testing activities for risk-significant SSCs described in Section 17.4.2.
- Designate and chair NSSS and Architect Engineer working groups as necessary to assist in accomplishing the objectives of the expert panel.
- Review and approve the recommendations of the working groups.
- Assess the overall station risk impact due to SSC performance and all implemented risk-informed programs (including D-RAP) after each plant-specific data update of the PRA.

The expert panel is made up of members with diverse backgrounds in engineering, operations, maintenance, risk and reliability analysis, operating experience and work control. The expert panel will have a minimum complement of five persons. During the detailed design phase of D-RAP, each major engineering organization performing detailed design will be represented on the panel (or working groups) as deemed necessary. The composition of the panel will change during the period leading up to fuel load and operations. The panel will continue to function during operations for the life of the plant.