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Sent: Friday, May 16, 2008 6:12 PM
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Cc: John Rycyna; Hanh Phan; Theresa Clark; Edward Fuller; Lynn Mrowca; Jim Xu; Joseph Colaccino
Subject: U.S. EPR Design Certification Application RAI No. 7
Attachments: RAI 7 SPLA 265.doc

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on May 9, 2008, and discussed with your staff on May 15, 2008. No change was made to the draft RAI as a result of that discussion. The schedule we have established for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs. For any RAIs that cannot be answered within 30 days, it is expected that a date for receipt of this information will be provided to the staff within the 30 day period so that the staff can assess how this information will impact the published schedule.

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Request for Additional Information No. 7, Revision 0
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U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation
Application Section: 19
SPLA Branch

QUESTIONS

19-56

Please provide an overview of the reactor coolant pump (RCP) seal failure model, including: (a) given loss of seal cooling, the assumed RCP seal leakage rate and its basis; (b) the basis for the standstill seal failure probability; and (c) the sensitivity of core damage frequency (CDF) for internal events to a ten-fold increase in the standstill seal failure probability.

19-57

Please describe why main steamline break inside containment with blowdown of multiple steam generators is a significant contributor to large release frequency (LRF). Identify any major conservatism in the modeling, including potential operator actions and mitigating features not modeled in the probabilistic risk assessment (PRA) that could preclude containment overpressure.

19-58

For each loss of component cooling water (LOCCW) initiating event, please provide additional information on: (a) the situation it represents, (b) the rationale for treating it as a separate initiator, and (c) a comparison of the initiating event frequency to comparable industry PRA models or other generic data sources. For example, please clarify why the LOCCW1 and LOCCW1L initiators, which appear to result in similar unavailability, are treated separately. Also, please clarify the statement on page 19.1-91 that LOCCW12 PM2 includes “maintenance unavailability for standby train.” What is the sensitivity of CDF and LRF for internal events to a ten-fold increase in the total LOCCW initiating event frequency?

19-59

Please provide the basis for the pressurizer safety valve (PSV) spurious opening frequency, stated as 2E-4 per year (/yr) in Final Safety Analysis Report (FSAR) Table 19.1-4.

19-60

Please provide additional information on the fast cooldown operation, including success criteria for both small and medium loss-of-coolant accidents (LOCA) and a discussion of the assumptions included in the human reliability analysis.

19-61

Please discuss why the evaluation of an interfacing systems LOCA (ISLOCA) due to a RCP thermal barrier tube leak does not consider dependence between operator failure to isolate the ISLOCA and operator failure to initiate secondary cooling and align residual heat removal (RHR) in four hours. This dependence appears to be evaluated for other ISLOCA events that do not lead directly to core damage.

19-62

Please describe how ventilation dependencies of equipment outside the safeguard buildings (e.g., emergency diesel generators (EDGs), essential service water pumps, and station blackout diesel generators) are modeled. If any dependencies are not modeled, please justify their exclusion and include sensitivity studies as appropriate.

19-63

According to section 8.3.1.1.1 of the FSAR, an alternate feed connection is provided between emergency power supply system (EPSS) divisional pairs to provide a normal and standby source of power when certain electrical components, including an EDG, are out of service. The discussion of station blackout power on page 19.1-9 of the FSAR indicates that this connection may be modeled in the PRA to some extent. Please discuss whether the alternate feed connection is credited in the PRA to provide power to certain equipment when the EDG in that division is out of service for maintenance.

19-64

Please discuss how accident sequences initiated by spurious opening of a main steam safety valve (MSSV) are modeled differently from other steam line breaks. The contribution of MSSV initiators appears not to be high enough to be included in Table 19.1-6 of the FSAR, but it is not clear how the mitigation strategy is different from the other steam line breaks with comparable initiating event frequencies.

19-65

Please describe how the steam line break outside containment (SLBO) frequency of 2.1E-3/yr was derived. Specifically, FSAR Table 19.1-4 indicates that leaks were excluded from the NUREG/CR-5750 value (1.0E-2/yr); please justify this exclusion.

19-66

Please provide a more detailed discussion of how feedwater line breaks are modeled in the PRA. Page 19.1-22 of the FSAR states that feedwater line breaks inside containment (FLBI) and steam line breaks inside containment (SLBI) are considered as a single initiator, but the SLBI initiating event frequency is taken directly from NUREG/CR-5750, which has a separate frequency for feedwater line breaks. The FSAR also states that feedwater line breaks outside containment are treated as total loss of feedwater (LOMFW) initiating events, but the LOMFW initiating event frequency is taken directly from NUREG/CR-6928, which does not appear to include feedwater line breaks.

19-67

Please provide the basis for the statement on page 19.1-56 of the FSAR that “[s]ome limited credit is given to the operators to recover from these software CCFs [common cause failures] (0.5).” How does this recovery action account for loss of instrumentation and the time available to the operators?

19-68

Please describe the results of additional sensitivity studies that illustrate the effect of digital instrumentation and control (DI&C) modeling uncertainties on CDF and risk insights. Sensitivity studies that may be appropriate are listed on pages 9 of the draft Interim Staff Guidance (ISG) on review of new reactor DI&C PRAs (DI&C-ISG-03, January 2008, Agencywide Document Access and Management System (ADAMS) Accession No. ML080350109).

19-69

Please provide the basic screening human error probability (HEP) used for pre-initiator human actions in the human reliability analysis (HRA). If the screening HEP of 0.03 from Accident Sequence Evaluation Program HRA Procedure (ASEP) is used, please perform a sensitivity study starting with the screening value of 0.05 recommended for cases where no plant visit or interaction is possible, as is the case at the design certification stage. (See page 3-32 of NUREG-1842 and page 4-2 of NUREG/CR-4772 for further details.)

19-70

In its discussion of post-initiator human actions, NUREG-1792 states that "[t]he total combined probability of all the HFEs in the same accident sequence/cut set should not be less than a justified value. It is suggested that the value not be below ~0.00001 since it is typically hard to defend that other dependent failure modes that are not usually treated (e.g., random events such as even a heart attack) cannot occur." Please discuss how this recommendation has been applied to the U.S. EPR PRA.

19-71

Please discuss how the use of a modern, digital control room has been considered in the HRA. Specifically, how is the use of touch screens and computerized procedures expected to affect operation? How is the failure of control room indication accounted for in the PRA?

19-72

Page 19.1-3 of the FSAR states that qualified analysts have performed each of the technical elements of the PRA. Please discuss the involvement of HRA practitioners and human factors specialists in the development of the U.S. EPR PRA, identified as a good practice in NUREG-1792.

19-73

(Follow-up to Question 19-04) Please provide additional information on the failure rates used for TELEPERM XS (TXS) components. How do the failure rates compare to both observed field experience and theoretical (e.g., part stress) estimates? How do the failure rates account for possible adverse environmental conditions (e.g., high temperature) in accident scenarios?

19-74

(Follow-up to Question 19-22) Please describe the expected shutdown sequence of events from entry into MODE 5 until the reactor cavity is flooded for refueling and during startup from the time when reactor cavity draining begins until entry into MODE 4.

19-75

(Follow-up to Question 19-24) Table 19.1-87 states that the reactor coolant system (RCS) is vented in Plant Operating State (POS) CAd2, CAd3, and CAu. Please clarify how the venting affects the availability of the steam generators for heat removal in these POS.

19-76

(Follow-up to Question 19-39) Please provide system information (including a description and system drawing or fault tree) as assumed in the PRA for the demineralized water system. This system appears to be modeled in the PRA, but no information on the system could be found in the rest of the FSAR.

19-77

(Follow-up to Question 19-46) Please discuss the success criteria and operator action timing for primary feed and bleed in various accident scenarios. The response to question 19-46 provided the probabilities for a modeling uncertainty case in which 1, 2, or 3 PSVs were needed; please discuss how these probabilities were selected. In addition, please clarify the statement on page 5-21 of the Applicant's Environmental Report that "[o]nly one valve is required for a successful feed and bleed," so larger valves are not necessary.