Request for Additional Information No. 29, Revision 0

8/01/2008

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 15.04.01 - Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition
SRP Section: 15.04.02 - Uncontrolled Control Rod Assembly Withdrawal at Power
SRP Section: 15.04.03 - Control Rod Misoperation (System Malfunction or Operator Error)
SRP Section: 15.04.04-15.04.05 - Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate
SRP Section: 15.04.08 - Spectrum of Rod Ejection Accidents (PWR)
SRP Section: 15.05.01-15.05.02 - Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory
Application Section: Ch 15
SRSB Branch

QUESTIONS

15.04.01-1

Please provide plots of DNBR and peak fuel centerline temperature as a function of time during this event to demonstrate that these limits are met.

Regulatory basis: SRP 15.4.1 UNCONTROLLED CONTROL ROD ASSEMBLY WITHDRAWAL FROM A SUBCRITICAL OR LOW POWER STARTUP CONDITION REVIEW, III Review Procedures, 9 states: *“For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the final safety analysis report (FSAR) meets the acceptance criteria.”*

15.04.01-2

Please explain Figure 15.4-4—“Uncontrolled Control Bank Withdrawal from a Subcritical or Low Power Startup Condition - Primary System Temperature”. The figure shows the temperature excursion in the hot leg of loop 4 following reactor trip during coastdown of the RCPs. Explain the physical basis for the temperature excursion. Describe the S-RELAP model used in these calculations. Explain the consistency of these temperature predictions and the EPR non-LOCA vessel model shown in ANP-10263P-A, Rev 0, “Codes and Methods Applicability Report for the US EPR.”

Regulatory basis: SRP 15.4.1, review procedures Item 6 states: *The significant results of the analysis should be presented and should include maximum power levels reached for the reactor and the peak fuel rod, reactor temperatures and pressures, maximum heat flux levels, and the related fuel duty.*

15.04.02-1

Please provide plots of DNBR and peak fuel centerline temperature as a function of time during this event to demonstrate that these limits are met.

Regulatory basis: SRP 15.4.2 UNCONTROLLED CONTROL ROD ASSEMBLY WITHDRAWAL AT POWER, III Review Procedure 7 states: "*For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the final safety analysis report (FSAR) meets the acceptance criteria."*

15.04.02-2

For Table 15.4-4 "Uncontrolled Control Bank Withdrawal at Power" Include the scram reactivity used in the analysis in the Table.

Regulatory basis: SRP 15.4.2 UNCONTROLLED CONTROL ROD ASSEMBLY WITHDRAWAL AT POWER, III Review Procedure, 3 states *" For a PWR, the reviewer ascertains that a full range of Anticipated Operational Occurence conditions are analyzed; the AOO calculation models are adequate; and that scram response of the flux, temperature, or pressure instrumentation is correctly calculated."*

15.04.03-1

Please provide plots of DNBR and peak fuel centerline temperature as a function of time during this event to demonstrate that these limits are met.

Regulatory basis: SRP 15.4.3 CONTROL ROD MISOPERATION (SYSTEM MALFUNCTION OR OPERATOR ERROR) III. REVIEW PROCEDURES 4 states: *"For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the final safety analysis report (FSAR) meets the acceptance criteria."*

15.04.04-15.04.05-1

Provide the calculated DNBR as a function of time during the transient.

Regulatory basis: SRP 15.4.4 – 15.4.5 STARTUP OF AN INACTIVE LOOP OR RECIRCULATION LOOP AT AN INCORRECT TEMPERATURE, AND FLOW CONTROLLER MALFUNCTION CAUSING AN INCREASE IN BWR CORE FLOW *RATE includes 4. PWR without loop isolation valves: Startup of a pump in an inactive loop. III Procedure includes "The results of the analysis are reviewed and compared with the acceptance criteria presented in Subsection II of this SRP section regarding the maximum pressure in the reactor coolant and main steam systems, as well as minimum DNBR (PWR) or MCPR (BWR, if applicable). Time-related variations of the following parameters should be reviewed for consistency."*

15.04.08-1

It appears that the ejected rod accident has been performed assuming the center control rod is ejected. The ejection of an asymmetric control rod may result in a higher radial peaking factor. Justify that the analyzed cases are limiting with respect to local peaking factors.

15.04.08-2

Please explain the differences in the ejected rod worths shown in Table 15.4-15 and those shown in Tables 15.-17 and -18.

Regulatory basis: SRP 15.4.8 SPECTRUM OF ROD EJECTION ACCIDENTS (PWR) III Review Procedures 1. E. states: "*The applicant's analytical methods are reviewed.*"

15.04.08-3

Verify that the 110 cal/gm criterion shown in Table 15.4-14—"Rod Ejection Accident DNBR Analysis – Ejected Rod Analysis Limits for U.S. EPR" is the PCMI limit of ANP-10286P, Revision 0, "U.S. EPR Rod Ejection Accident Methodology Topical Report," AREVA NP Inc., November 2007.

Regulatory basis: SRP 15.4.8 SPECTRUM OF ROD EJECTION ACCIDENTS (PWR) II Acceptance Criteria.

15.04.08-4

Explain the physical basis of the loop 4 temperature excursion shown in Figure 15.4-35—HFP Rod Ejection Accident Overpressurization Analysis – Primary System Temperature, Figure 15.4-39—60% NP Rod Ejection Accident Overpressurization Analysis – Primary System Temperature, and Figure 15.4-43—HZP Rod Ejection Accident Overpressurization Analysis – Primary System Temperature. Consider the time delays in the nuclear power, heat transfer from the fuel to the coolant, and fluid mixing in the reactor pressure vessel.

Regulatory basis: SRP 15.4.8 SPECTRUM OF ROD EJECTION ACCIDENTS (PWR) III Review Procedures 1. E. states: "*The applicant's analytical methods are reviewed.*"

15.04.08-5

Please identify the computer codes used in the analysis. Explicitly or by reference provide a description of each code. Please provide the nodalization used in these numerical models.

15.04.08-6

Identify the core location of the limiting rod in both the initial and equilibrium core.

15.04.08-7

Provide the physical basis to show that the least negative Doppler temperature coefficient has been used in the calculations.

15.04.08-8

The FSAR states: “The plant simulation computer code S-RELAP5 (Reference 1) is used to determine the peak pressure response of the primary system to the RCCA ejection event.” Please identify the model and nodalization used.

15.04.08-9

Please describe the xenon condition at the start of the REA transient analysis. Justify that the assumed xenon condition conservative.

15.05.01-15.05.02-1

Table 6.8-1—Extra Borating System Design and Operating Parameters provides a nominal flow rate of 52 gpm / pump and a maximum allowable flow rate of 110.8 gpm (for the system). This corresponds to the flow rate of 15.5 lbm/sec shown in Figure 15.5-9—Inadvertent Operation of the EBS. Table 9.3.4-1—Major CVCS Component Design Data Sheet 1 of 5 shows 2 Centrifugal Charging Pumps with a normal flow rate of 176 gpm, and a maximum flow rate of 285 gpm. This does not correspond to the flow rate of 13 to 16 lbs/sec shown in Figure 15.5-18—CVCS Malfunction that Increases RCS Inventory. Please explain.

15.05.01-15.05.02-2

Explain the charging flow shown in Figure 15.5-18—CVCS Malfunction that Increases RCS Inventory – Charging Flow. Explain why the flow rate to loop 2 and 4 are different prior to RT. It is recognized that the analysis assumes that two pumps are de-energized at RT; one pump is immediately started and tripped again at about 1000 sec. Explain the rapid changes in flow shown in the figure between reactor trip and isolation of the changing pumps at 1019 seconds.