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Sent: Wednesday, April 09, 2008 4:16 PM
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Cc: Pederson Ronda M (AREVA NP INC); Jack Donohue; Walton Jensen; Christopher P Jackson; Joseph Colaccino
Subject: U.S. EPR Design Certification Application RAI No. 1
Attachments: RAI 1 SPCV 122 (4).doc

Attached please find the subject requests for additional information (RAIs). This RAI was discussed with your staff on April 2, 2008. 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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Mail Envelope Properties (3D6B1F4F51D25E4E9B6C93FF92DAE5170A90F70EC2)

Subject: U.S. EPR Design Certification Application RAI No. 1
Sent Date: 04/09/2008 4:16:09 PM
Received Date: 04/09/2008 4:16:00 PM
From: Getachew Tesfaye

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Files	Size	Date & Time
MESSAGE	65076	04/09/2008
RAI 1 SPCV 122 (4).doc	55038	

Options
Expiration Date:
Priority: olImportanceNormal
ReplyRequested: False
Return Notification: False

Sensitivity: olNormal
Recipients received:

Request for Additional Information No. 1, Revision 0
4/9/2008
U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 06.02.01 - Containment Functional Design
Application Section: 6.2
SPCV Branch

QUESTIONS

06.02.01-1

(FSAR Section 6.2.1) The NRC staff is reviewing the AREVA evaluation model for the containment functional design following a loss of coolant accident. The evaluation model includes methodology for determining the steam release and water spillage from the reactor system following a piping rupture as well as the methodology for determining the thermodynamic condition within the containment building. The EPR containment design is unlike that of operating PWRs in the US in that it does not include safety related containment sprays or fan coolers and instead relies on other containment heat removal mechanisms to a greater extent than credited in the containment analyses of operating PWRs. These heat removal mechanisms are the quenching of steam within the reactor system by incoming ECCS water and the removal of heat from the containment atmosphere by the internal containment structures.

Although the NRC staff has previously approved the AREVA evaluation model for use in safety analyses for the core reloads of operating plants, the staff did not review the methodology for application to the safety analyses of a plant without active safety related systems able to cool and mix the containment atmosphere following a postulated design basis LOCA. The NRC staff therefore requests that AREVA follow Regulatory Guide 1.203 in seeking approval of the EPR containment evaluation model. RG 1.203 describes a process that the NRC staff considers acceptable in developing and assessing evaluation models that will be used to analyze transient and accident behavior that is within the design basis of a nuclear power plant. The staff recognizes that portions of the EPR containment model have undergone a process similar to the one described in RG 1.203 for operating plants. For the portions of the model that have undergone a process similar to RG 1.203, the staff requests that AREVA provide a description or reference to the prior process and explain why the prior process bounds the application of the model for the EPR.

a. Regulatory Guide (RG)-1.203, “Transient and Accident Methods,” Regulatory Element 1 provides 20 steps for a process of evaluation model development and assessment. These elements discuss how computer codes will be assessed for adequacy for specific applications, describes their usage with other computer codes and their qualification for the specific applications for which they will be applied. Please address each of these 20 steps and show how the recommendations are met by the AREVA evaluation model for the US-EPR containment safety analysis.

b. Regulatory Element 2 of RG-1.203 deals with quality assurance. Appendix B to 10 CFR Part 50 describes NRC requirements regarding quality assurance for nuclear power plants. Please provide descriptions of how the evaluation model for US-EPR containment analysis meets these requirements.

c. Regulatory Element 3 of RG-1.203 deals with documentation. Please provide or provide available reference to the following documentation for the US-EPR containment evaluation model:

c.1. Requirements

The requirements for the evaluation model as developed in the phenomena identification and ranking table (PIRT) should be established and documented. Step 4 of Regulatory Position 1 to RG-1.203 deals with the development of the PIRT for the various applications for the evaluation model. The PIRT provides a means of determining those processes and phenomena for which code assessment should be demonstrated. Please provide PIRTs for the evaluation of the containment functional design following a LOCA. Provide the qualifications of the PIRT panel members.

c.2. Methodology

Please provide methodology documentation as described in the regulatory guide. You should include noding diagrams as well as the selection of input options and justify the selection of each option chosen. The noding arrangement for the containment building and the reactor system should be provided and justified.

c.3. User manuals and user guidelines

Provide documentation describing user instructions for the application of the evaluation model software for US-EPR containment analysis.

c.4. Scaling Reports

Scaling analyses should be conducted to ensure that the data and the models based on those data, will be applicable to the full-scale analyses performed for the US-EPR. Provide scaling reports for the test facilities used in the validation for the US-EPR containment evaluation model as discussed in the regulatory guide.

c.5. Assessment Reports

As described in the regulatory guide, the evaluation model should be assessed against applicable experimental data to ensure that the significant phenomena and processes for containment analysis as identified in the PIRT are being adequately modeled.

c.6. Uncertainty Analysis Reports

Please provide documentation of any uncertainty analysis performed for use of US-EPR containment evaluation model.

06.02.01-2

(FSAR Section 6.2.1) The following is a list of issues involving the mass and energy release calculations that the staff believes should be investigated in the application of RG 1.203 to the AREVA post LOCA containment evaluation model for US-EPR. Please indicate by cross-reference where each of the following items is addressed in the response to RAI #1.

a. Blowdown
   Demonstrate that the RELAP5 reactor core heat transfer assumptions are conservative for containment analysis.

b. Refill
   Historically no refill time was assumed so that the water level in the reactor vessel at the end of blowdown was set at the bottom of the core when the refill started. If AREVA makes other assumptions they should be justified.

c. Reflood
   The liquid predicted to be carried out of the top of the core for US-EPR to the steam generator tubes should be compared with carryout rate fraction (CRF) measurements for similar reactor core conditions such as those in the FLECHT series. SRP 6.2.1.3 defines the CRF as the mass ratio of liquid exiting the core to the liquid entering the core. Heat transfer from the steam generator tubes should be justified. Steam quenching in the cold legs by the incoming ECCS water should be justified. Core flow reversals predicted by RELAP5 which act to quench steam generated by the core in the lower plenum volume need to be justified by comparisons with experimental data or other methodology which does not produce flow reversals should be used.

   For a postulated double ended cold leg break provide graphs of the core inlet flow rate, the core exit liquid flow rate, and the CRF. Plot resolution should be sufficiently detailed so that flow oscillations are displayed eg. ½ second of transient time. Provide comparison plots of the CRF predicted by the RELAP5 code with the predictions of applicable experiments.

d. Post Reflood
   Boiling in the core at low pressure will cause a two phase level to rise into the steam generators. The level swell assumptions used in the RELAP5 and GOTHIC computer models of the reactor system should be justified. Heat transfer from the steam generator tubes should be justified. Steam quenching in the cold legs by the incoming ECCS water should be justified. Note that SRP 6.2.1.3 recommends that "Steam from decay heat boiling in the core should be assumed to flow to the containment by the path which produces the minimum amount of mixing with the ECCS injection water." If other assumptions are made for steam/water mixing justification is required.

06.02.01-3

(FSAR Section 6.2.1) The following issues involve the containment pressure and temperature calculations. The staff believes that these should be investigated in the application of RG 1.203 to the AREVA post LOCA containment evaluation model for US-EPR. Please indicate by cross-reference where each of these items is addressed in the response to RAI #1.

a. Models for determining the thermodynamic condition within the reactor building should be justified including the noding detail selected for the various reactor building compartments and for the internal flow paths.

b. Heat transfer assumptions for the internal containment structures should be justified. Separate validation should be performed for the vertical and horizontal heat transfer surfaces.

06.02.01-4

(FSAR Section 6.2.1.3) Please provide the staff with the SRELAP5 input used to determine M&E release for the double ended cold leg break at a reactor coolant pump suction that is described in the FSAR.

06.02.01-5

(FSAR Section 6.2.1) The staff has the following questions regarding the GOTHIC model (short-term LOCA, case 7A) which was provided to the staff by AREVA. There are some apparent inconsistencies between the FSAR descriptions of the containment analysis and the EPR GOTHIC input. Please provide clarification for the following:

a. FSAR Table 6.2.1-5 does not specify the presence of stainless steel; it appears from the GOTHIC input that the three IRWST heat sinks are lined with stainless steel. Stainless steel is also specified for the containment shell in the GOTHIC input and the stainless steel is input as painted; is that correct? Provide a material property table for the passive heat sinks which includes material density, thermal conductivity and heat capacity for each material type.

b. FSAR Section 6.2.1.1.3 indicates that the heat sinks listed are all exposed to the containment atmosphere; however it appears that in the GOTHIC input the three IRWST heat sinks are exposed to the pool; which is correct?

c. Containment initial and boundary conditions are listed in FSAR Table 6.2.1-4 which do not appear to reflect the intended operating conditions, i.e., the accessible areas of the containment have an operating range different from the inaccessible areas. The GOTHIC input for initial conditions uses the upper bound pressure and lower bound temperature based on the FSAR table. The heat sinks have various initial temperatures. Please justify that the initial conditions selected in the GOTHIC input are those which provide the most conservative result.

d. In FSAR Section 6.2.1.1.3, it appears that nitrogen gas from all 4 accumulators is assumed to be released into the containment. The GOTHIC input seems to assume only 3 accumulators release nitrogen. Please address this apparent inconsistency. Justify that the inputting of nitrogen from only 3 accumulators is acceptable.

06.02.01-6

(FSAR Section 6.2.1) To enable the staff to better compare the results from audit calculations with AREVA's containment analysis results for US-EPR, please provide clarification for the following:

a. FSAR Table 6.2.1-3 lists different operating characteristics for CCW trains 1 &4 from those of 2 &3. Which of the four CCW trains listed is used in the long-term GOTHIC peak containment pressure analyses?

b. FSAR Table 6.2.1-8, the peak pressure row indicates units of “psia” but the figures (6.2.1-18 and 6.2.1-20) infer the pressures should be “psig;” please correct the apparent inconsistencies.

c. FSAR Tables 6.2.1-7 and 6.2.1-8, specifically Cases 14D, 28 and 31, the IRWST initial temperatures are not 122F as are the other cases, please explain the basis for these specified temperatures of 170F and 248F.

06.02.01-7

(FSAR Section 6.2.1) In order to facilitate the review, the NRC staff needs certain design information as soon as possible. These include: (a) The design of rupture foils and convection foils; (b) details on the modeling of the containment in the multi-node GOTHIC calculations; and (c) detailed results for one of the multi-node GOTHIC calculations. (See detail below.)

a. Foils are installed in a steel framework. Does the framework separate the foil into many foils each of which has to rupture individually? What is the size of the individual foils? What is the total surface area of the foils? What is the available flow area once the foils rupture? Justify this flow area. What materials are the foils made of? What is the thickness of the foils? What is the weight of the foils per square foot? The above questions apply to both rupture foils and convection foils. In addition, how many fusible links are on the frame of a convection foil? Where are the links located?

b. Please provide a simplified sketch of the containment. Show internal walls, major components (steam generators, tanks, and so on) and the location and size of all mixing dampers, rupture foils and convection foils.

c. Provide the noding diagram that was used in the multi-node GOTHIC calculations (pages 47-50 of the U.S.-EPR Design Certification Acceptance Review presentation by AREVA of January 29, 2008). Provide the input data used in these calculations: volumes, elevations, cross sections, flow path dimensions, heat transfer surfaces. What was the break location and break size selected for the above referenced multi-node GOTHIC calculations?

d. Provide for one of the multi-node calculations (LB LOCA cold leg break if available) sufficient details of the results to permit visualization of flow patterns in the containment as well as heat transfer to the various heat sinks. Results should be given as a function of time for the duration of the accident. Please include flow in each flow path; content, temperature and pressure of each node; surface temperatures of significant heat sinks and heat transfer to each significant heat sink.

06.02.01-8

FSAR Section 6.2.1.2.3 states that the Homogeneous Equilibrium Model (HEM) is utilized in several of the cases to calculate the mass flux into subcompartments following a postulated piping break. The HEM is generally not acceptable for the calculation of liquid and two-phase critical flow from a piping break because it under predicts critical mass flow rates.

Please provide subcompartment analyses using break flow models which are conservative for the prediction of critical flow.

06.02.01-9

(FSAR Section 6.2.1.5) For the evaluation used to determine minimum containment pressure for ECCS analysis, please provide the following information:

a. The mass and energy release input that was utilized to produce the containment pressure as a function of time in Figure 15.6-50. Include the spilled ECCS water and the nitrogen accumulator gas release.

b. Describe and justify assumptions made for the mixing between containment steam and the spilled ECCS water.

c. FSAR Section 15.6.5.1.2 indicates that a condensing heat transfer coefficient of 1.7 times that predicted by the Uchida correlation is used. Provide a comparison of the containment pressure result during the core reflood period between this assumption and that produced by the heat transfer model recommended in BTP 6.2 which is 4 times the Tagami correlation followed by 1.2 times the Uchida correlation. Provide this justification specifically for the US-EPR analysis given in Figure 15.6-50.

d. FSAR Section 6.2.1.5.3 states that heat transfer between the IRWST water and the containment vapor space is not (Error in FSAR) considered in the analysis. Rewrite paragraph and include how this heat transfer path is considered in a conservative manner for minimum containment pressure analysis.

06.02.01-10

FSAR Section 6.2.3 describes the US-EPR Reactor Shield Building (RSB). Provide the evaluation for the RSB post LOCA pressure and temperature following design basis conditions. Provide the graphical results for the limiting case of the atmospheric pressure and temperature and the reactor containment building (RCB) wall temperature. The evaluation should include heat transfer from the RCB, the equipment heat loads within the RSB and the decrease in the RSB volume as the result of thermal and mechanical expansion of the RCB.