ArevaEPRDCPEm Resource

From: BRYAN Martin (EXTERNAL AREVA) [Martin.Bryan.ext@areva.com]
Sent: Wednesday, September 08, 2010 4:33 PM
To: Tesfaye, Getachew
Cc: DELANO Karen (AREVA); ROMINE Judy (AREVA); BENNETT Kathy (AREVA); PANNELL George (AREVA)
Subject: Response to U.S. EPR Design Certification Application RAI No. 413, FSAR Ch. 7
Attachments: RAI 413 Response US EPR DC.pdf

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information RAI 413.

The following table indicates the respective pages in the response document, "RAI 413 Response US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

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A complete answer is not provided for 33 of the 33 questions. The schedule for a technically correct and complete response to these questions is provided below.

Question #	Response Date
RAI 413 07.08-10	March 15, 2011
RAI 413 07.08-11	March 15, 2011
RAI 413 07.08-12	March 15, 2011
RAI 413 07.08-13	March 15, 2011
RAI 413 07.08-14	March 15, 2011
RAI 413 07.08-15	December 17, 2010
RAI 413 07.08-16	March 15, 2011
RAI 413 07.08-17	March 15, 2011
RAI 413 07.08-18	December 17, 2010
RAI 413 07.08-19	January 28, 2011
RAI 413 07.08-20	December 17, 2010
RAI 413 07.08-21	January 28, 2011
RAI 413 07.08-22	December 17, 2010
RAI 413 07.08-23	December 17, 2010
RAI 413 07.08-24	January 28, 2011
RAI 413 07.08-25	December 17, 2010
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RAI 413 07.08-31	January 28, 2011
RAI 413 07.08-32	January 28, 2011
RAI 413 07.08-33	December 17, 2010
RAI 413 07.08-34	December 17, 2010
RAI 413 07.08-35	January 28, 2011
RAI 413 07.08-36	November 19, 2010
RAI 413 07.08-37	January 28, 2011
RAI 413 07.08-38	December 17, 2010
RAI 413 07.08-39	November 19, 2010
RAI 413 07.08-40	January 28, 2011
RAI 413 07.08-41	November 19, 2010
RAI 413 07.08-42	March 15, 2011

Sincerely,

Martin (Marty) C. Bryan
U.S. EPR Design Certification Licensing Manager
AREVA NP Inc.
Tel: (434) 832-3016
702 561-3528 cell
Martin.Bryan.ext@areva.com

From: Tesfaye, Getachew [mailto:Getachew.Tesfaye@nrc.gov]
Sent: Monday, August 09, 2010 3:46 PM
To: ZZ-DL-A-USEPR-DL
Cc: Mott, Kenneth; Spaulding, Deirdre; Jackson, Terry; Canova, Michael; Colaccino, Joseph; ArevaEPRDCPEm Resource
Subject: U.S. EPR Design Certification Application RAI No. 413(4772), FSAR Ch. 7

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on June 4, 2010, and discussed with your staff on July 22, 2010. Draft RAI Questions 07.08-19, 07.08-21, 07.08-23, and 07.08-41,were modified 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.

Thanks,
Getachew Tesfaye
Sr. Project Manager
NRO/DNRL/NARP
(301) 415-3361

Hearing Identifier: AREVA_EPR_DC_RAIs
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From: BRYAN Martin (EXTERNAL AREVA)

Created By: Martin.Bryan.ext@areva.com

Recipients:
"DELANO Karen (AREVA)" <Karen.Delano@areva.com>
Tracking Status: None
"ROMINE Judy (AREVA)" <Judy.Romine@areva.com>
Tracking Status: None
"BENNETT Kathy (AREVA)" <Kathy.Bennett@areva.com>
Tracking Status: None
"PANNELL George (AREVA)" <George.Pannell@areva.com>
Tracking Status: None
"Tesfaye, Getachew" <Getachew.Tesfaye@nrc.gov>
Tracking Status: None

Post Office: AUSLYNCMX02.adom.ad.corp

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Request for Additional Information No. 413(4772), Revision 1

8/9/2010

U.S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 07.08 - Diverse Instrumentation and Control Systems
Application Section: ANP-10304

QUESTIONS for Instrumentation, Controls and Electrical Engineering 1 (AP1000/EPR Projects) (ICE1)

Question 07.08-10:

Provide correct and unambiguous design descriptions for the PAS.

10CFR52.47(a)(2) states that an application must contain a “description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justification therefore, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. … The description shall be sufficient to permit understanding of the system designs and their relationship to the safety evaluations….”

Section 2.4.9, item 3.2, of the U.S. EPR FSAR, Tier 1 design descriptions, state that the PAS system hardware and system software are diverse from the PS and SAS system hardware and software. However, Section 7.1.1.4.6 of the U.S. EPR Tier 2, FSAR, Revision 2-markups, removed the diversity commitment from the PAS system. The design description now states that there “are no diversity requirements for the PAS.” The design information provided for the design basis items, taken alone and in combination, should have one and only one interpretation. Therefore, the staff requests the applicant to provide correct and unambiguous design descriptions for the PAS in order to complete its safety evaluation. The staff expects this RAI response to be included within the FSAR design descriptions.

Response to Question 07.08-10:

The response to this question will be provided by March 15, 2011.

Question 07.08-11:

Provide the design descriptions and design commitments for the credited PAS equipment diversity.

10CFR52.47(a)(2) states that an application must contain a “description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justification therefore, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. … The description shall be sufficient to permit understanding of the system designs and their relationship to the safety evaluations….” 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 22 requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.”

The applicant describes PAS equipment diversity attributes within Technical Report ANP-10304, “U.S. EPR Diversity and Defense-in-Depth Assessment Technical Report ,” Revision 1, as a way to show sufficient mitigation for a postulated PS software common-cause-failure (SWCCF) and to justify that adequate diversity has been provided in the design. However, Section 7.1.1.4.6 of the U.S. EPR FSAR, Tier 2, Revision 2 Interim markups, removed the diversity commitment from the PAS system. The design description now states that there “are no diversity requirements for the PAS.” The information provided for the design basis items, taken alone and in combination, should have one and only one interpretation. It should be possible to trace the information in each design basis item to the safety analyses, plant system design documents, regulatory requirements, applicant commitments, or other plant documents. Therefore, if the applicant’s FSAR PAS design does not commit to diverse PAS equipment, then the applicant is not able to take credit for PAS diversity in order to demonstrate that adequate and sufficient PAS diversity exist within their design.

Response to Question 07.08-11:

The response to this question will be provided by March 15, 2011.

Question 07.08-12:

Provide the PAS design descriptions that would demonstrate that the non-safety-related PAS meets the applicable regulatory requirements for manual operation of the safety-related main steam relief trains (MSRT) during a postulated software common-cause-failure (SWCCF) of the PS.

IEEE-603-1998, Clause 5.6, requires in part, independence between safety systems and other systems. 10CFR 50.62 requires, in part, that ATWS equipment must be designed to perform its function in a reliable manner. 10CFR52.47(a)(2) states that an application must contain a description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justification therefore, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to the safety evaluations. GDC 24, states, in part, that interconnection of the protection and control systems shall be limited to assure that safety is not significantly impaired.

The U.S. EPR Tier 2, FSAR Section 7.8.1.2.2, and Technical Report ANP-10304, "U.S. EPR Diversity and Defense-in-Depth Assessment Technical Report ," Revision 1, Appendix A, Section A.2.2, provide one line design descriptions which state that manual MSRT is available through the PAS. Point 3 of Item II.Q. of the Commission's SRM to SECY-93-087, states, in part, that the [credited] diverse or different function may be performed by a non-safety system if the system is of sufficient quality to perform the necessary function under the associated event conditions. Point 4 of Item II.Q. of the Commission's SRM to SECY-93-087, states, in part, that a set of controls located in the main control room should be provided for manual system-level actuation of critical safety functions. The applicant states in U.S. EPR Tier 2, FSAR Section 7.1.1.4.6 of the Revision 2 Interim mark-ups that the PAS equipment has no quality requirements. Therefore, the staff is not able to determine if the PAS design consist of sufficient quality. The staff could not identify design descriptions that would demonstrate that the credited PAS diverse manual controls for the MSRT are actuated at the system level. In addition, U.S. EPR Tier 2, FSAR Section 7.1.1.4.6 Revision 2 Interim mark-ups do not provide design descriptions to describe PAS outputs to safety-related actuators. PAS System Architecture Figure 7.1-12 of Tier 2 of the U.S. EPR FSAR does not provide sufficient design details such as proper isolation between PAS and safety-related components that would permit the staff to understand the design as it relates to the safety evaluations. Therefore, the staff requests the applicant to demonstrate PAS regulatory conformance for credited PAS actions within the applicant's D3 technical report.

Response to Question 07.08-12:

The response to this question will be provided by March 15, 2011.

Question 07.08-13:

Describe in detail the electrical isolation circuitry that is used between the PS and the DAS system, starting from the output of a sensor.

IEEE Std 603-1998, Section 5.6.3.1, Interconnected equipment, states, among other things, that no credible failure on the non-safety side of an isolation device shall prevent any portion of a safety system from meeting its minimum performance requirements during and following any design basis event requiring that safety function. A failure in an isolation device shall be evaluated in the same manner as a failure of other equipment in a safety system. GDC 24, states, in part, that Interconnection of the protection and control systems shall be limited to assure that safety is not significantly impaired.

BTP 7-11, “Guidance On Application And Qualification Of Isolation Devices,” Revision 5 (BTP 7-11), provides guidelines for reviewing the use of electrical isolation devices to allow connections between safety and non-safety related systems. BTP 7-11 deals with the criteria and methods used to confirm that the design of isolation devices assures that credible failures in the connected non-safety channels will not prevent the safety systems from meeting their required functions. In order to address this guidance to evaluate conformance to the Commission’s requirements, the staff requested design information about credited isolation devices during the second round of RAIs (ML081690513), ANP-10284Q2P, RAI question 19, for the review of the applicant’s D3 topical report, “U.S. EPR Instrumentation and Control Diversity and Defense-in-Depth Methodology Topical Report,” ANP-10284 (ML071760188). However, the applicant requested that the staff no longer issue a separate safety evaluation report (SER) for ANP-10284, but to incorporate the D3 review into the overall SER of the new D3 technical report, ANP-10304. The staff has utilized the entire response to RAI question 19 of ANP-10284Q2P (including all figures and schematics) when performing its safety evaluations. Therefore, the entire response should be submitted in relationship to the new D3 technical report, ANP-10304.

Response to Question 07.08-13:

The response to this question will be provided by March 15, 2011.

Question 07.08-14:

With the exception of the methodology contained in Section 4.0 of Revision 0, provide the missing information found in Revision 0 into Revision 1, of Technical Report ANP-10304, "U.S. EPR Diversity and Defense-in-Depth Assessment Technical Report."

10CFR52.47(a)(2) states that an application must contain a “description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justification therefore, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to the safety evaluations.

Response to Question 07.08-14:

The response to this question will be provided by March 15, 2011.

Question 07.08-15:

Provide the design basis and the detailed description of why the RCSL is credited for certain events and not credited for others.

10CFR52.47(a)(2) states that an application must contain a "description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justification therefore, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. … The description shall be sufficient to permit understanding of the system designs and their relationship to the safety evaluations…."

The RCSL design descriptions do not permit the staff to sufficiently understand when to credit the RCSL for event mitigation within the "U.S. EPR Diversity and Defense-in-Depth Assessment Technical Report" (D3-TR), ANP-10304, Revision 1 (ML093420199), event analysis. It states in Section 4.1 of the D3-TR that a "…decision [was] made not to credit the RCSL in the D3 assessment." This position is also stated in Section A.2.2 of Appendix A. However, Section A.3.5.5, CVCS Malfunction, takes credit for RCSL control rod (RCCA) insertion to alert the control room operation of a dilution event. Therefore, the staff could not identify clear design descriptions that will permit sufficient understanding of credited accident mitigation actions for the RCSL within the D3-TR analysis.

Response to Question 07.08-15:

The response to this question will be provided by December 17, 2010.

Question 07.08-16:

Provide the detailed design descriptions that would explain all of the systems available that would allow the operator to manually control the main steam relief trains (MSRT) from the main control room (MCR). Include within the description the signal path from the operator's manual input from within the MCR to the final actuation device (MSRT).

10CFR52.47(a)(2) states that an application must contain a "description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justification therefore, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. … The description shall be sufficient to permit understanding of the system designs and their relationship to the safety evaluations…."

The staff could not identify design descriptions that would explain how the non-safety related process automation system (PAS) would initiate safety-related MSRT operation. However within the applicant's "U.S. EPR Diversity and Defense-in-Depth Assessment Technical Report" (D3-TR), ANP-10304, Revision 1 (ML093420199), Section A.2.2 of Appendix A states that credit is taken for a manual function to control MSRT from the PICS and PAS. Therefore, the staff could not identify clear design descriptions that will permit sufficient understanding of credited manual functions for the MSRT operation.

Response to Question 07.08-16:

A response to this question will be provided by March 15, 2011.

Question 07.08-17:

Identify all RCSL and PAS actuations, for both a manual and automatic control, that rely on a PS output. For each of the conditions identified, evaluate compliance with 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 24 and its affect on the D3 analyses presented in Appendix A to ANP-10304 Rev 1.

10CFR52.47(a)(2) states that an application must contain a “description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justification therefore, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. … The description shall be sufficient to permit understanding of the system designs and their relationship to the safety evaluations….” GDC 29, “Protection Against Anticipated Operational Occurrences” requires that the protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

“U.S. EPR Diversity and Defense-in-Depth Assessment Technical Report” (D3-TR), ANP-10304, Revision 1 (ML093420199), Section A.1 of Appendix A states that “… SWCCF in the PS, concurrent with an AOO or PA, does not affect I&C functions outside the PS, if those functions do not rely on a PS output” and refers to Section 4 of ANP-10304 Rev 1. Section 4.12 states that due to the diversity attributes described in Section 4.2, “… control functions in the PAS that do not rely on PS output, can be assumed to function normally following a PS SWCCF concurrent with an AOO or PA.” While Section 4.2 addresses PAS diversity attributes, it does not say whether any PAS function relies on a PS output.

Response to Question 07.08-17:

A response to this question will be provided by March 15, 2011.

Question 07.08-18:

Provide the following additional information regarding the use of nominal DAS reactor trip setpoints in the D3 analyses:

a. Clarification of whether the DAS setpoint values listed in Table A.2-3 of ANP-10304 Rev 1 are used directly in the S-RELAP5 simulation of the D3 transient events,

b. Justification for the use of the Table A.2-3 DAS nominal trip setpoints without uncertainty in the D3 analyses, specifically addressing instrumentation channel uncertainty components such as sensor calibration tolerance and drift, basic sensor accuracy, and random process measurement uncertainty,

c. Identification of the setpoint methodology used to derive the DAS reactor trip setpoints, and

d. Identification of the design reference document that contains the DAS setpoint values.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

Table A.2-3 lists the PS and DAS reactor trip setpoints. The table implies that the DAS setpoints are used directly in the D3 safety analysis with no allowance for uncertainty, and several of the evaluation results sections (e.g., Section A.3.4.2) of ANP-10304 Rev 1 state that nominal reactor trip setpoints are used in their respective D3 analysis. Protection system setpoints are normally derived from the Analytical Limits that are utilized in the safety analysis. From the Analytical Limits the Limiting Trip Setpoints (corresponding to the Limiting Safety System Settings defined in 10CFR50.36) are derived taking into account total instrumentation channel uncertainty (e.g., calibration tolerance, drift, basic sensor accuracy). Additional margin is usually added to the Limiting Trip Setpoints, resulting in Nominal Trip Setpoints. The PS setpoints listed in ANP-10304 Rev 1 Table A.2-3 are consistent with the Limiting Trip Setpoints provided in the plant Technical Specifications, DCD FSAR Chapter 16 Table 3.3.1-1.

Response to Question 07.08-18:

A response to this question will be provided by December 17, 2010.

Question 07.08-19:

For each of the following, provide a description of how the best-estimate initial condition assumptions used in the D3 analyses were generated for:

a. Core average axial and core radial power distributions and the reactivity coefficient curves,

b. The core average axial power distributions used in the D3 analyses,

c. The reactivity coefficient curves (MTC, Doppler),

d. The hot channel factors $F_{\Delta h}$ and F_Q used in the D3 analyses,

e. The scram reactivity curves and the total scram reactivity worth (Δρ or pcm),

f. The non-fuel-related reactor core parameters affecting DNBR, including T_{inlet}, reactor system pressure, and total core flow, and

g. An assessment of the initial steady-state DNBR operating margin with the best-estimate assumptions and comparison to the expected Technical Specification (TS 3.2.3) DNBR Limiting Condition for Operation (LCO) value.

For each of the above b through f, describe or provide a comparison of each of the parameters to their respective design values as contained in the FSAR and justify any deviations. Also, state whether the same values of the above parameters are used consistently for all the D3 analyses.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

Response to Question 07.08-19:

The response to this question will be provided by January 28, 2011.

Question 07.08-20:

Provide the basis or a reference for the containment structure ultimate pressure capacity stated in Section A.2.4 of Appendix A to ANP-10304 Rev 1.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

Section A.2.4 of Appendix A to ANP-10304 states that the acceptance criterion for meeting containment structure integrity is the ultimate pressure capacity of the containment. The ultimate pressure capacity is cited in Section A.2.4 to be 156 psig, or 2.52 times the containment design pressure of 62 psig. US EPR FSAR Table 3.8-6 lists the containment ultimate capacity as 119 psig, which is also the value used in the severe accident evaluations in FSAR Section 19.2.4.4.2.4.

Response to Question 07.08-20:

A response to this question will be provided by December 17, 2010.

Question 07.08-21:

Provide additional explanation of the changes made to S-RELAP5 as utilized for the D3 analysis presented in ANP-10304 Rev 1, including the following:

a. A description of Heat Transfer modifications (i.e., fluid temperature, Inayatov multiplier, LIQHTC) made to the S-RELAP5 code and the purpose of the change, and

b. The validation basis for T_{avg} change made to the S-RELAP5 code.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

Section A.2.5 of Appendix A to ANP-10304 Rev 1 states that minor changes to the S-RELAP5 code were made to reflect improved heat transfer in the steam generator secondary system.

Response to Question 07.08-21:

A response to this question will be provided by January 28, 2011.

Question 07.08-22:

Provide a plot of DNBR normalized to SAFDL for the D3 analysis of the Increase in Steam Flow event, along with an explanation of the basis for the initial DNBR margin. In addition, address margin to the LPD SAFDL for the D3 analysis of the Increase in Steam Flow event.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

Section A.3.2.3 of Appendix A to ANP-10304 Rev 1 states that no fuel failure is predicted for the Increase in Steam Flow event. In comparing the D3 analysis of ANP-10304 Rev 1 to the corresponding FSAR Chapter 15 analysis, the maximum predicted power level is substantially higher in the D3 analysis (130% versus approximately 108%). Although initial DNBR margin is expected to be higher given the best-estimate assumptions of the D3 analysis, the decrease in DNBR due to the much higher power transient condition could be more than offsetting.

Response to Question 07.08-22:

A response to this question will be provided by December 17, 2010.

Question 07.08-23:

For the D3 analysis provided in "U.S. EPR Diversity and Defense-in-Depth Assessment Technical Report" (D3-TR), ANP-10304, Revision 1 (ML093420199), Appendix A, the following additional information is needed for the MSLB case with PS SWCCF at full power conditions:

a. Demonstrate that a return to criticality does not occur following reactor trip,

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

The FSAR Chapter 15 analysis identifies the main steam line break (MSLB) initiated from HZP as the limiting Steam System Piping Failure event, resulting in a small number of fuel failures due to the SAFDL on cladding strain (Local Power Density) being exceeded. [NOTE: HZP is typically more limiting than HFP because the absence of decay heat will allow greater cooldown of the RCS and therefore an increase in moderator temperature reactivity.] Therefore, additional information is needed for the case with PS SWCCF at full power conditions for the staff to complete its review.

Response to Question 07.08-23:

A response to this question will be provided by December 17, 2010.

Question 07.08-24:

Provide additional information to support the reliability of the predicted DAS trip actuation on Low SG Level, including the following:

a. A description of the S-RELAP5 SG level model and the SG narrow range level instrumentation model, including validation basis, and

b. An evaluation of the decalibration effects of the MSIV closure on the narrow range instrumentation and how decalibration is treated in the simulation model.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

The engineering analysis of the Inadvertent MSIV Closure event presented in Section A.3.3.2 of ANP-10304 Rev 1 shows that a DAS reactor trip actuation occurs on Low SG Level, terminating the secondary system pressure excursion at about 1640 psia, or 100 psia below the D3 analysis criterion of 120% of secondary system design pressure (1.20 x 1450 psia = 1740 psia). The Inadvertent Closure of MSIV event analyzed in the FSAR results in PS reactor trip actuation (at approximately 6 seconds) on High SG Pressure, which is not available through DAS. Therefore, an explicit D3 engineering analysis is provided in ANP-10304 Rev 1 Section A.3.3.2 for the Inadvertent Closure of MSIV event. The D3 analysis shows that at approximately 130 seconds DAS provides a reactor trip actuation on Low SG Level. The pressure in the affected SG reaches approximately 1640 psia, or 113% of the secondary system design pressure (design pressure is 1450 psia, per FSAR Table 10.3-1). Considering the rate of change of the SG pressure excursion and its calculated peak value relative to the D3 analysis criterion (1640 psia peak SG pressure versus 1740 psia criterion), additional information on the SG level model and the DAS Low SG Level trip function is required in order for staff to complete its review of the Inadvertent Closure of MSIV D3 analysis.

Response to Question 07.08-24:

A response to this question will be provided by January 28, 2011.

Question 07.08-25:

Identify the credited diverse means to address the loss of the LOOP emergency diesel generator (EDG) initiation start signal from the PS in order to operate the EFWS, in accordance with the D3 policy stated in SRM to SECY-93-087, Point 3. If the credited diverse means is a manual actuation, provide the detailed design descriptions that would address the guidance of Standard Review Plan (SRP) Appendix 18-A, "Crediting Manual Operator Actions in Diversity and Defense-In-Depth (D3) Analyses." Applying the credited diverse means, provide the following additional information:

a. Clarification of whether the EDGs or the SBO DGs are assumed to be utilized to power the EFW pumps in the D3 analysis,

b. A detailed description of how the EDGs are started following the occurrence of Loss of Non-Emergency AC Power to Station Auxiliaries event, including the following:

   I. Identification of the control systems (e.g., SICS, PICS) utilized to start the diesel generators and/or load the EFW pumps to the diesel generator buses,

   II. Identification of the location where the operator action to start diesel generators and/or load the EFW pumps to the diesel generator buses must take place, e.g., main control room or field local control panel,

   III. An assessment of the time required for the operator to achieve the necessary start of EFW, relative to the SG boil off time (reported as 1 ½ hours from the time of event initiation), and

   IV. An assessment of whether existing operating procedures (e.g., EPGs/EOPs) provide adequate symptom-based instructions to ensure the operator response assumed in the D3 analysis, or whether a special D3 coping procedure is required.

c. An evaluation of the results and consequences of the Loss of Non-Emergency AC Power to Station Auxiliaries event with SWCCF in the PS assuming that the steam generators boil dry (in approximately 1 ½ hours, according to the Section A.3.3.3 analysis) before EFW can be initiated.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events. Point 3 of the Commission's SRM to SECY 93-087, states, in part:

> *If a postulated common-[cause] failure could disable a safety function, then a diverse means, with a documented basis that the diverse means is unlikely to be subject to the same common-[cause] failure, shall be required to perform either the same function as or a different function….*

The Loss of Non-Emergency AC Power to Station Auxiliaries event addressed in Section A.3.3.3 of ANP-10304 Rev 1 explains that the EDGs must be manually started for powering the EFW pumps, consistent with the unavailability of a LOOP signal from the PS. FSAR Section

7.3.1.2.12 states that the EDGs are manually started through SICS in the main control room, but according to ANP-10304 Rev 1 Section 4.1, SICS is not credited in the D3 analysis. Section A.3.3.3 also states that the Station Blackout diesel generators (SBO DGs) start automatically upon loss of AC power, but the EFW pumps need to be manually loaded to the SBO DGs. According to the D3 analysis, the steam generators will boil dry within approximately 1 ½ hours from the start of the event. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to Question 07.08-25:

A response to this question will be provided by December 17, 2010.

Question 07.08-26:

Provide the following additional information regarding the Complete Loss of Forced Coolant Flow as described in ANP-10304 Rev 1 Section A.3.4.2:

a. A comparison of the RCS flow coastdown (normalized flow versus time from t=0 sec to 5 sec) for the FSAR Chapter 15 DBE analysis versus the D3 analysis presented in Section A.3.4.2 of ANP-10304 Rev 1,

b. A plot of core average heat flux (normalized core average heat flux versus time from t=0 sec to 10 sec) for the D3 analysis,

c. A plot of DNBR (DNBR normalized to the SAFDL) versus time (t=0 sec to 5 sec) for the D3 analysis, and

d. A table that lists the sequence of events for the Complete Loss of Forced RCS Flow, showing times of DAS reactor trip actuation, beginning of control rod insertion, minimum DNBR, and turbine trip for the D3 analysis.

In addition, explain the differences in initial DNBR margin between the FSAR Chapter 15 analysis and the D3 analysis, i.e., identify the best-estimate assumptions and assess their beneficial effects on DNBR.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

ANP-10304 Rev 1 Section A.3.4.2 provides an engineering analysis of the Complete Loss of Forced RCS Flow event (AOO). The D3 analysis shows that reactor trip actuation occurs on a DAS Low RCS Flow (Two Loops) signal, which is somewhat later that the PS reactor trip actuation on Low RCP Speed (DAS does not have a Low RCP Speed reactor trip function). The Complete Loss of Forced Coolant Flow is a rapidly occurring event with a calculated minimum DNBR occurring at around 3 seconds, and it is the limiting DNBR AOO. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to Question 07.08-26:

A response to this question will be provided by December 17, 2010.

Question 07.08-27:

For the RCP Rotor Seizure event described in ANP-10304 Rev 1 Section A.3.4.3, provide the following additional information:

a. A comparison of the RCS flow coastdown rate assumed for D3 evaluation versus the flow coastdown rate shown FSAR Figure 15.3-9;

b. The calculated initial (t=0 seconds) and minimum DNBR for the RCP Rotor Seizure event as analyzed in both Section 15.3.3 of the FSAR and the D3 analysis;

c. A table that lists the sequence of events for the RCP Rotor Seizure event, showing times of DAS reactor trip actuation, beginning of control rod insertion, minimum DNBR, and turbine trip.

In addition, explain the differences in initial DNBR margin between the FSAR Chapter 15 analysis and the D3 analysis, i.e., identify the best-estimate assumptions and assess their beneficial effects on DNBR.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

ANP-10304 Rev 1 Section A.3.4.3 provides an engineering argument for the RCP Rotor Seizure event (PA), stating that DAS will actuate a reactor trip on Low-Low RCS Flow (one loop), and therefore provide protection comparable to the PS as shown in the FSAR Section 15.3.3 analysis. The FSAR analysis shows that the DNBR SAFDL is exceeded, resulting in fuel damage. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to Question 07.08-27:

A response to this question will be provided by December 17, 2010.

Question 07.08-28:

For the Uncontrolled RCCA Withdrawal at Power event described in ANP-10304 Rev 1 Section A.3.5.2, provide the following additional information:

a. The calculated initial (t=0 seconds) and minimum DNBR for the Uncontrolled RCCA Withdrawal at Power event, and

b. A table that lists the sequence of events for the Uncontrolled RCCA Withdrawal at Power event, showing times of DAS reactor trip actuation, beginning of control rod insertion, minimum DNBR, and positioning of the RCCA bank being withdrawn.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

The Uncontrolled RCCA Withdrawal at Power event is analyzed assuming full power initial conditions at both BOC and EOC conditions, and with the RCCAs inserted to the Technical Specification Power Dependent Insertion Limit (PDIL). The time in cycle life-time (BOC vs. EOC) affects moderator temperature coefficient and control rod bank worth. The FSAR Section 15.4.2 analysis states that the reactor system is protected by PS Low DNBR, High LPD, Excore High Rate of Change, High Core Power Level, and High Pressurizer Level reactor trip functions, none of which are provided by DAS. The D3 engineering analysis provided in ANP-10304 Rev 1 Section A.3.5.2, however, shows that DAS actuates a reactor trip on Low SG Level and that reactor power peaks at approximately 108 percent. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to Question 07.08-28:

A response to this question will be provided by December 17, 2010.

Question 07.08-29:

For the Dropped RCCA event described in ANP-10304 Rev 1 Section A.3.5.3, provide the following additional information:

a. A description of the response of RCSL during the limiting Dropped RCCA event and its affect t on the calculated results of the transient,

b. The calculated initial (t=0 seconds) and minimum DNBR for the limiting Dropped RCCA event, and

c. An explanation of why the drop of a single RCCA is not more limiting than the drop of an RCCA bank, considering that the drop of a single RCCA results in a more severe radial peaking pattern (nonsymmetric core condition) with a similar return to full power conditions. Provide the calculated radial peaking for both the RCCA bank drop and single RCCA drop to support the explanation.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

In the FSAR analysis, the PS actuates a reactor trip on Low DNBR, which is unavailable in DAS. The limiting case, as reported in Section A.3.5.3, is the drop of a full bank of RCCAs from the Power Dependent Insertion Limit (PDIL) position. The D3 analysis of this event shows that following the drop of the RCCA bank, the reactor power returns to full power with no reactor trip actuation. The applicant reports that the best estimate conditions offset the failure of the PS trip functions, and no fuel failures occur. In addition, the D3 analysis assumes that RCSL is automatically controlling RCCAs, which worsens the transient. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to Question 07.08-29:

A response to this question will be provided by January 28, 2011.

Question 07.08-30:

Explain the assumption that RCSL will respond to the Boron Dilution event as described in Section A.3.5.5 of ANP-10304 Rev 1.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

Section A.3.5.5 of ANP-10304 Rev 1 describes a Boron Dilution event where RCSL responds to the reactivity transient by automatic insertion of RCCAs, thereby alerting the main control room operator of the dilution event. Sections 4.1, 4.12, and A.2.2 state that RCSL is assumed not to be available in the D3 analysis. The U.S. EPR, Tier 2, Section 7.7.2.3.10, state that "…RCS boron concentration is calculated in the PS…" and "Four redundant limitation signals from the PS are transferred to RCSL." Therefore, upon postulated failure of the PS, the "transfer" of PS limitation signals to the RCSL would not occur. The staff could not identify sufficient design descriptions that would clearly describe how the RCSL would respond to the Boron Dilution event as described in Section A.3.5.5 of ANP-10304 Rev 1.

Response to Question 07.08-30:

A response to this question will be provided by January 28, 2011.

Question 07.08-31:

Provide an explanation of difference in the DNBR transient between the no-rupture and with-rupture RCCA Ejection cases described in Section A.3.5.6, including:

a. Identification of any differences in analysis assumptions (e.g., initial conditions, reactivity parameters) that affect the transient,

b. A comparison of the key reactor parameters affecting DNBR, e.g., power level, peaking factors, reactor pressure, coolant temperatures, core flow.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

The applicant reports that the RCCA Ejection event, assuming no vessel rupture, does not exceed the DNBR SAFDL, whereas for the rupture cases the DNBR decreases below the SAFDL. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to Question 07.08-31:

A response to this question will be provided by January 28, 2011.

Question 07.08-32:

For the Steam Generator Tube Rupture (SGTR) event described in ANP-10304 Rev 1 Section A.3.7.2, provide the following additional information:

a. An explanation of why the D3 analysis of the SGTR results in SG overfill as compared to the FSAR Section 15.6.3 analysis which does not show SG overfill,

b. The design reference basis for main steam line loading with solid fill at a hydrostatic pressure of 1.25 design pressure, and

c. An assessment of the possibility of steam line water hammer occurrence during the SG overfill portion of the SGTR event.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

Section A.3.7.2 of ANP-10304 Rev 1 reports that the Steam Generator Tube Rupture (SGTR) event results in an overfill of the affected steam generator and discharge of liquid into the main steam line, whereas the SG does not overfill in FSAR Section Chapter 15 SGTR analysis. With essentially the same manual controls available and credited for the FSAR analysis and D3 analysis, plus availability of the DAS MFW isolation on SG High Level, it is unclear why the SGTR with PS SWCCF results in SG overfill. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to Question 07.08-32:

A response to this question will be provided by January 28, 2011.

Question 07.08-33:

For the large break loss of coolant accident (LBLOCA) event described in ANP-10304 Rev 1 Section A.3.7.3, provide the following additional information:

a. A summary of the sensitivity calculation results referred to in Section A.3.7.3.1 that demonstrate the conclusion that continued operation of RCPs does not significantly affect the LBLOCA results, and

b. An assessment of the effects of continued operation of the RCPs during two phase and vapor LBLOCA conditions on the RCPs themselves, i.e., cavitation or overspeed damage resulting in subsequent unavailability of the RCPs.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

Section A.3.7.3.1 of ANP-10304 Rev 1 states that an automatic RCP trip does not occur during a LBLOCA with SWCCF in the PS because the RCP trip is provided by the PS. It is further stated in Section A.3.7.3.1 that the RCPs are expected to continue in operation, and that such continued operation does not significantly affect the LBLOCA results, with reference to sensitivity calculation results. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to Question 07.08-33:

A response to this question will be provided by December 17, 2010.

Question 07.08-34:

Provide a resolution of whether the turbine bypass system may be relied upon for partial cooldown during a SBLOCA with SWCCF in the PS. If manual operator actions are required to be performed, provide the timing requirements and a description of the operator actions, including whether such operator actions are covered by existing plant operating procedures or symptom-based EPGs.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

ANP-10304 Rev 1 Section A.3.7.3.2 describes a condition where the MSRT partial cooldown function is unavailable during a SBLOCA with SWCCF in the PS and the turbine bypass system provides the partial cooldown function, as long as the MSIVs remain open. The applicant notes that an on-going review could affect the use of the turbine bypass system during SBLOCA. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to Question 07.08-34:

A response to this question will be provided by December 17, 2010.

Question 07.08-35:

Identify the credited diverse means to address the loss of the PS initiated safety functions of RT on a high pressurizer level and CVCS isolation on high pressurizer level, as discussed in ANP-10304, Revision 1, Section A.3.6.2," CVCS Malfunction that Increases RCS Inventory," in accordance with the D3 policy stated in SRM to SECY-93-087, Point 3. If the credited diverse means are manual actuations, provide the detailed design descriptions that would address the guidance of Standard Review Plan (SRP) Appendix 18-A, "Crediting Manual Operator Actions in Diversity and Defense-In-Depth (D3) Analyses." Applying the credited diverse means, provide the following additional information:

a. A description of the operator action sequence, starting with recognition of the increasing RCS inventory event and ending with isolation of CVCS,

b. The time within which the operator can accomplish the required actions to isolate CVCS and terminate the event, and

c. Identification of the procedure or procedure type (e.g., EPGs) that will prescribe the steps to accomplish the required operator action and whether a special D3 coping procedure is required.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

For the Increase in RCS Inventory event with SWCCF, as described in Section A.3.6.2 of ANP-10304 Rev 1, the transient does not terminate on Pressurizer High Level as is the case with the FSAR Chapter 15 analysis. In the FSAR analysis, the pressurizer high level causes a reactor trip and CVCS isolation. For the D3 analysis, neither the reactor trip nor the CVCS isolation occurs, and the pressurizer fills solid. Although Section A.3.6.2 of ANP-10304 Rev 1 states that pressurizer PSRVs are capable of relieving water, thus ensuring that the RCS pressure boundary is maintained, sufficient indication and procedures should be available to ensure that the plant operations personnel recognize and terminate the event in a timely manner. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to Question 07.08-35:

A response to this question will be provided by January 28, 2011.

Question 07.08-36:

For the D3 analysis of SBLOCA, described ANP-10304 Rev 1 Section A.3.7.3.2, provide the following additional information relative to RCP trip during SBLOCA with SWCCF in the PS:

a. The criteria for operator determination of the need for RCP trip, i.e., LOCA as confirmed by two-phase RCS flow conditions,

b. The displays available to the main control room operators to determine the need for RCP trip, and

c. Identification of the procedure or procedure type (e.g., EPGs) that will prescribe the steps to accomplish the required operator action and whether a special D3 coping procedure is required.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

The D3 analysis of SBLOCA, described ANP-10304 Rev 1 Section A.3.7.3.2, states that an RCP trip does not automatically occur. Although SBLOCA sensitivity analyses referred to in Section A.3.7.3.2 reportedly demonstrate that 10 CFR 50.46 criteria are met even without RCP trip, the staff's position is that the RCPs should be tripped during the occurrence of a SBLOCA either automatically or manually by operator action. The lack of automatic RCP trip for SBLOCA is not in conformance with TMI Action Plan requirement II.K.3.5 (NUREG-0737) as stated in FSAR Sections 15.6.5.2.2 and 15.6.5.2.6. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to Question 07.08-36:

A response to this question will be provided by November 19, 2010.

Question 07.08-37:

Provide information to justify that manual isolation of the main control room as described in Section A.3.9 of ANP-10304 Rev 1, will occur in a timely manner, including:

a. A description of how the need for manual isolation of the main control room is recognized,

b. The time line for manual control room isolation, including recognition of need for isolation and achievement of isolation,

c. Identification of the procedure or procedure type (e.g., EPGs) that will prescribe the steps to accomplish the required operator action and whether a special D3 coping procedure is required, and

d. A discussion of whether the operator actions required to manually isolate the main control room represent a diverse means of protective action to ensure the D3 radiological analysis acceptance criteria are met.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

For the Radiological Consequences of accidents with SWCCF in the PS, Section A.3.9 of ANP-10304 Rev 1 states that DAS does not provide automatic control room isolation. Analyses performed by the applicant, however, indicate that manual isolation of the main control room should take place within 30 minutes of an event initiation. The staff is concerned that manual isolation of the main control room may represent a required D3 coping action and is a considered a vulnerability which should address the guidance contained in the BTP-7-19 acceptance criteria, which states, in part:

> *The applicant/licensee should (1) demonstrate that sufficient diversity exists to achieve these goals, (2) identify the vulnerabilities discovered and the corrective actions taken, or (3) identify the vulnerabilities discovered and provide a documented basis that justifies taking no action.*

Therefore, the staff request additional information to address this vulnerability.

Response to Question 07.08-37:

A response to this question will be provided by January 28, 2011.

Question 07.08-38:

For Section A.3.2.3 of ANP-10304 Rev 1, describe the neutronics calculations that are done to determine the decalibration. In particular, address the issue of the temperature decrease in neutron reflector coolant. Also describe how the decalibration is implemented in the S-RELAP5 model.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

Section A.3.2.3 of ANP-10304 Rev 1 states that the reduction in downcomer water temperature that results from the Increase in Steam Flow event causes the excore neutron detectors to become decalibrated, thereby delaying reactor trip. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to Question 07.08-38:

A response to this question will be provided by December 17, 2010.

Question 07.08-39:

For Section A.2.5 of Appendix A to ANP-10304, Rev 1, describe the modeling changes, including the following:

a. A description of the model for each additional available system,

b. Any nodalization changes that were made relative to the FSAR S-RELAP5 model, and

c. Provide nodalization diagrams.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

Section A.2.5 of Appendix A to ANP-10304 states that system modeling is changed to reflect available systems and expected behavior during best estimate conditions. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to Question 07.08-39:

A response to this question will be provided by November 19, 2010.

Question 07.08-40:

For Section A.2.5 of ANP-10304 Rev 1, describe how the models for pressurizer pressure and level control are validated to assure that they accurately describe the plant response.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

Section A.2.5 of ANP-10304 Rev 1 states that the pressurizer pressure and level control systems are included in the S-RELAP5 best-estimate non-LOCA model. The staff is not able to identify design descriptions that would permit sufficient understanding in order to complete the safety evaluation.

Response to Question 07.08-40:

A response to this question will be provided by January 28, 2011.

Question 07.08-41:

Provide the S-RELAP5 input base deck and the other required input deck files to perform a confirmatory run of the following events:

a. The limiting increase in steam flow event,

b. The inadvertent Closure of an MSIV event,

c. CVCS Malfunction resulting in decreased RCS boron event,

d. RCCA ejection with rupture event,

e. SBLOCA event.

Provide S-RELAP5 source code and relevant documentation used for D3 analysis. Documentation should include, but not limited to, source code modification record, description of the altered numerical algorithm, benchmark and validation calculation, etc.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protective function. The Staff Requirements Memorandum to SECY 93-087, Item II.Q, states that the vendor or applicant shall analyze each postulated common-mode failure for each event and shall demonstrate adequate diversity within the design for each of these events.

Response to Question 07.08-41:

A response to this question will be provided by November 19, 2010.

Question 07.08-42:

Since signal diversity is credited between the two sub-systems of the Protection System (PS), identify the reactor trip and engineered safety feature actuation functions that will be assigned to sub-systems A and B.

10 CFR Part 50, Appendix A, GDC 22, requires, in part, that design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function. The guidance for signal diversity is identified in NUREG/CR-6303, Section 3.2.5. The description of signal diversity for the PS subsystems A and B is provided in Section 4.2 of Technical Report ANP-10304, "U.S. EPR Diversity and Defense-in-Depth Assessment Technical Report," Revision 1. The signal diversity provides for measurements from different sensors measuring different process parameters, to perform reactor trip functions that can protect against the same events. However, when reviewing the design rules for PS subsystems contained in ANP-10281P, "U.S. EPR Digital Protection System Technical Report," Revision 0, sentence number 7 of Section 10.2 states that

7. If a signal is required in both subsystems, it is implemented twice, once in each subsystem.

In addition, sentence number 2 of the same section is specific only for RT signal diversity (versus both RT and ESFAS signal diversity). Sentence 2 states:

2. A sensor used for a primary RT initiation signal in a given subsystem cannot be used by the second, diverse initiating signal (if one exists) in the other subsystem.

Therefore, it appears that the design rules for plant parameters assigned to PS subsystems could possibly allow for identical signal sharing between the subsystems and this would not meet the signal diversity guidance of NUREG/CR-6303. If this is the case, then either (1) signal diversity might not be able to be credited for the PS subsystems or (2) the Technical Report ANP-10304 may have to define and describe the particular conditions and/or events for when signal diversity can be credited for the PS subsystems. While a methodology for assigning reactor trip and engineering safety feature actuation functions to the sub-systems is provided in ANP-10281P, the staff finds that the actual assignment of the functions and how they achieve signal diversity is necessary for the staff to complete its review.

Response to Question 07.08-42:

A response to this question will be provided by March 15, 2011.