ArevaEPRDCPEm Resource

From: WELLS Russell (AREVA) [Russell.Wells@areva.com]
Sent: Thursday, March 31, 2011 7:53 AM
To: Tesfaye, Getachew
Cc: LENTZ Tony (EXTERNAL AREVA); BENNETT Kathy (AREVA); DELANO Karen (AREVA); ROMINE Judy (AREVA); RYAN Tom (AREVA)
Subject: Response to U.S. EPR Design Certification Application RAI No. 315, FSAR Ch. 16 OPEN ITEM, Supplement 9
Attachments: RAI 315 Supplement 9 Response US EPR DC.pdf

Getachew,

AREVA NP Inc. provided responses to the four questions of RAI No. 315 on April 5, 2010. Supplement 1 and Supplement 2 responses to RAI No. 315 were sent on May 20, 2010 and August 27, 2010, respectively, to provide a revised schedule. Supplement 3 response to RAI No. 315 was sent on October 12, 2010 to provide a response to one of the remaining four questions, 16-321. Supplement 4 response to RAI No. 315 was sent on October 20, 2010 to provide a revised schedule. Supplement 5 response to RAI No. 315 was sent on November 18, 2010 to provide a revised schedule. Supplement 6 response to RAI No. 315 was sent on December 16, 2010 to provide a revised schedule. Supplement 7 response to RAI No. 315 was sent on January 26, 2011 to provide a revised schedule. Supplement 8 response to RAI No. 315 was sent on March 22, 2011 to provide a revised schedule.

The attached file, "RAI 315 Supplement 9 US EPR DC.pdf," provides a response to two of the remaining three questions.

The following table indicates the respective pages in the response document, "RAI 315 Supplement 9 US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 315 — 16-318	2	12
RAI 315 — 16-319	13	14

The schedule for a technically correct and complete response to the remaining question remains unchanged and is provided below.

Question #	Response Date
RAI 315 — 16-320	April 26, 2011

Sincerely,

Russ Wells
U.S. EPR Design Certification Licensing Manager
AREVA NP, Inc.
3315 Old Forest Road, P.O. Box 10935
Mail Stop OF-57
Lynchburg, VA 24506-0935
Phone: 434-832-3884 (work)
434-942-6375 (cell)
Fax: 434-382-3884
Russell.Wells@Areva.com

From: WELLS Russell (RS/NB)
Sent: Tuesday, March 22, 2011 1:04 PM
To: Tesfaye, Getachew
Cc: BENNETT Kathy (RS/NB); DELANO Karen (RS/NB); ROMINE Judy (RS/NB); RYAN Tom (RS/NB)
Subject: Response to U.S. EPR Design Certification Application RAI No. 315, FSAR Ch. 16 OPEN ITEM, Supplement 8

Getachew,

AREVA NP Inc. provided responses to the four questions of RAI No. 315 on April 5, 2010. Supplement 1 and Supplement 2 responses to RAI No. 315 were sent on May 20, 2010 and August 27, 2010, respectively, to provide a revised schedule. Supplement 3 response to RAI No. 315 was sent on October 12, 2010 to provide a response to one of the remaining four questions, 16-321. Supplement 4 response to RAI No. 315 was sent on October 20, 2010 to provide a revised schedule. Supplement 5 response to RAI No. 315 was sent on November 18, 2010 to provide a revised schedule. Supplement 6 response to RAI No. 315 was sent on December 16, 2010 to provide a revised schedule. Supplement 7 response to RAI No. 315 was sent on January 26, 2011 to provide a revised schedule.

A revised schedule is provided below to allow additional time to address comments and have additional interaction with the staff on the three remaining questions.

A complete answer is not provided for the remaining 3 questions. The schedule for a technically correct and complete response to these questions is changed and is provided below.

Question #	Response Date
RAI 315 — 16-318	April 26, 2011
RAI 315 — 16-319	April 26, 2011
RAI 315 — 16-320	April 26, 2011

Sincerely,

Russ Wells
U.S. EPR Design Certification Licensing Manager
AREVA NP, Inc.
3315 Old Forest Road, P.O. Box 10935
Mail Stop OF-57
Lynchburg, VA 24506-0935
Phone: 434-832-3884 (work)
434-942-6375 (cell)
Fax: 434-382-3884
Russell.Wells@Areva.com

From: BRYAN Martin (External RS/NB)
Sent: Wednesday, January 26, 2011 3:04 PM
To: Tesfaye, Getachew
Cc: DELANO Karen (RS/NB); ROMINE Judy (RS/NB); BENNETT Kathy (RS/NB); RYAN Tom (RS/NB)
Subject: Response to U.S. EPR Design Certification Application RAI No. 315, FSAR Ch. 16 OPEN ITEM, Supplement 7

Getachew,

AREVA NP Inc. provided responses to the four questions of RAI No. 315 on April 5, 2010. Supplement 1 and Supplement 2 responses to RAI No. 315 were sent on May 20, 2010 and August 27, 2010, respectively, to provide a revised schedule. Supplement 3 response to RAI No. 315 was sent on October 12, 2010 to provide a response to one of the remaining four questions, 16-321. Supplement 4 response to RAI No. 315 was sent on October 20, 2010 to provide a revised schedule. Supplement 5 response to RAI No. 315 was sent on November 18, 2010 to provide a revised schedule. Supplement 6 response to RAI No. 315 was sent on December 16, 2010 to provide a revised schedule.

A revised schedule is provided below to allow additional time to address comments and have additional interaction with the staff on the three remaining questions.

A complete answer is not provided for the remaining 3 questions. The schedule for a technically correct and complete response to these questions is changed and is provided below.

Question #	Response Date
RAI 315 — 16-318	March 24, 2011
RAI 315 — 16-319	March 24, 2011
RAI 315 — 16-320	March 24, 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: BRYAN Martin (External RS/NB)
Sent: Thursday, December 16, 2010 3:28 PM
To: Tesfaye, Getachew
Cc: DELANO Karen (RS/NB); ROMINE Judy (RS/NB); BENNETT Kathy (RS/NB); RYAN Tom (RS/NB); Miernicki, Michael
Subject: Response to U.S. EPR Design Certification Application RAI No. 315, FSAR Ch. 16 OPEN ITEM, Supplement 6

Getachew,

AREVA NP Inc. provided responses to the four questions of RAI No. 315 on April 5, 2010. Supplement 1 and Supplement 2 responses to RAI No. 315 were sent on May 20, 2010 and August 27, 2010, respectively, to provide a revised schedule. Supplement 3 response to RAI No. 315 was sent on October 12, 2010 to provide a response to one of the remaining four questions, 16-321. Supplement 4 response to RAI No. 315 was sent on October 20, 2010 to provide a revised schedule. Supplement 5 response to RAI No. 315 was sent on November 18, 2010 to provide a revised schedule.

A revised schedule is provided below to allow additional time to address comments and have additional interaction with the staff on the three remaining questions.

A complete answer is not provided for the remaining 3 questions. The schedule for a technically correct and complete response to these questions is changed and is provided below.

Question #	Response Date
RAI 315 — 16-318	January 27, 2011

RAI 315 — 16-319	January 27, 2011
RAI 315 — 16-320	January 27, 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: BRYAN Martin (External RS/NB)
Sent: Thursday, November 18, 2010 7:59 AM
To: Tesfaye, Getachew
Cc: DELANO Karen (RS/NB); ROMINE Judy (RS/NB); BENNETT Kathy (RS/NB); RYAN Tom (RS/NB); Miernicki, Michael
Subject: Response to U.S. EPR Design Certification Application RAI No. 315, FSAR Ch. 16 OPEN ITEM, Supplement 5

Getachew,

AREVA NP Inc. provided responses to the four questions of RAI No. 315 on April 5, 2010. Supplement 1 and Supplement 2 responses to RAI No. 315 were sent on May 20, 2010 and August 27, 2010, respectively, to provide a revised schedule. Supplement 3 response to RAI No. 315 was sent on October 12, 2010 to provide a response to one of the remaining four questions, 16-321. Supplement 4 response to RAI No. 315 was sent on October 20, 2010 to provide a revised schedule.

A revised schedule is provided below to allow additional time to address comments and have additional interaction with the staff on the three remaining questions.

A complete answer is not provided for the remaining 3 questions. The schedule for a technically correct and complete response to these questions is changed and is provided below.

Question #	Response Date
RAI 315 — 16-318	December 21, 2010
RAI 315 — 16-319	December 21, 2010
RAI 315 — 16-320	December 21, 2010

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: BRYAN Martin (External RS/NB)
Sent: Wednesday, October 20, 2010 3:40 PM
To: 'Tesfaye, Getachew'

Cc: DELANO Karen (RS/NB); ROMINE Judy (RS/NB); BENNETT Kathy (RS/NB); RYAN Tom (RS/NB)
Subject: Response to U.S. EPR Design Certification Application RAI No. 315, FSAR Ch. 16 OPEN ITEM, Supplement 4

Getachew,

AREVA NP Inc. provided responses to the four questions of RAI No. 315 on April 5, 2010. Supplement 1 and Supplement 2 responses to RAI No. 315 were sent on May 20, 2010 and August 27, 2010, respectively, to provide a revised schedule. Supplement 3 response to RAI No. 315 was sent on October 12, 2010 to provide a response to one of the remaining four questions, 16-321.

A revised schedule is provided below to allow additional time to address comments and have additional interaction with the staff on the three remaining questions.

A complete answer is not provided for the remaining 3 questions. The schedule for a technically correct and complete response to these questions is changed and is provided below.

Question #	Response Date
RAI 315 — 16-318	November 22, 2010
RAI 315 — 16-319	November 22, 2010
RAI 315 — 16-320	November 22, 2010

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: BRYAN Martin (External RS/NB)
Sent: Tuesday, October 12, 2010 4:59 PM
To: 'Tesfaye, Getachew'
Cc: DELANO Karen (RS/NB); ROMINE Judy (RS/NB); BENNETT Kathy (RS/NB); RYAN Tom (RS/NB); LENTZ Tony (External RS/NB)
Subject: Response to U.S. EPR Design Certification Application RAI No. 315, FSAR Ch. 16 OPEN ITEM, Supplement 3

Getachew,

AREVA NP Inc. provided responses to the four questions of RAI No. 315 on April 5, 2010. Supplement 1 and Supplement 2 responses to RAI No. 315 were sent on May 20, 2010 and August 27, 2010, respectively, to provide a revised schedule.

The attached file, “RAI 315 Supplement 3 US EPR DC.pdf,” provides a partial response.

The following table indicates the respective pages in the response document, “RAI 315 Supplement 3 US EPR DC.pdf,” that contain AREVA NP’s response to the subject question.

Question #	Start Page	End Page
RAI 315 — 16-321	2	3

The schedule for a technically correct and complete response to the remaining three questions remains unchanged and will be provided on October 21, 2010.

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: BRYAN Martin (External RS/NB)
Sent: Friday, August 27, 2010 12:01 PM
To: 'Tesfaye, Getachew'
Cc: DELANO Karen (RS/NB); ROMINE Judy (RS/NB); BENNETT Kathy (RS/NB); RYAN Tom (RS/NB)
Subject: Response to U.S. EPR Design Certification Application RAI No. 315, FSAR Ch. 16 OPEN ITEM, Supplement 2

Getachew,

AREVA NP Inc. provided a schedule for technically correct and complete responses to the 4 questions in RAI No. 315 on April 5, 2010. AREVA provided an updated schedule for the remaining 4 responses on May 20, 2010 to allow for additional interaction with the NRC.

A revised schedule is provided below to allow additional time to address comments and have additional interaction with the staff on the four remaining questions.

A complete answer is not provided for the remaining 4 questions. The schedule for a technically correct and complete response to these questions is changed and is provided below.

Question #	Response Date
RAI 315 — 16-318	October 21, 2010
RAI 315 — 16-319	October 21, 2010
RAI 315 — 16-320	October 21, 2010
RAI 315 — 16-321	October 21, 2010

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: BRYAN Martin (EXT)
Sent: Thursday, May 20, 2010 12:18 PM
To: 'Tesfaye, Getachew'
Cc: DELANO Karen V (AREVA NP INC); ROMINE Judy (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); RYAN Tom (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 315, FSAR Ch. 16 OPEN ITEM, Supplement 1

Getachew,

AREVA NP Inc. provided a schedule for technically correct and complete responses to the 4 questions in RAI No. 315 on April 5, 2010. As agreed with the NRC, additional time is needed for the NRC to review and discuss the draft responses to these questions with AREVA.

The schedule for technically correct and complete responses to these questions has been revised as provided below.

Question #	Response Date
RAI 315 — 16-318	August 31, 2010
RAI 315 — 16-319	August 31, 2010
RAI 315 — 16-320	August 31, 2010
RAI 315 — 16-321	August 31, 2010

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: BRYAN Martin (EXT)
Sent: Monday, April 05, 2010 5:01 PM
To: 'Tesfaye, Getachew'
Cc: DELANO Karen V (AREVA NP INC); ROMINE Judy (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); PANNELL George L (AREVA NP INC); LENTZ Tony F (EXT)
Subject: Response to U.S. EPR Design Certification Application RAI No. 315 (3878), FSARCh. 16 OPEN ITEM

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 315 Response US EPR DC.pdf" provides a schedule since a technically correct and complete response to the 4 questions is not provided.

The following table indicates the respective pages in the response document, "RAI 315 Response US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 315 — 16-318	2	2
RAI 315 — 16-319	3	3
RAI 315 — 16-320	4	4
RAI 315 — 16-321	5	5

A complete answer is not provided for the 4 questions. The schedule for a technically correct and complete response to these questions is provided below.

Question #	Response Date
RAI 315 — 16-318	May 20, 2010
RAI 315 — 16-319	May 20, 2010
RAI 315 — 16-320	May 20, 2010

Sincerely,
Martin (Marty) C. Bryan
Licensing Advisory Engineer
AREVA NP Inc.
Tel: (434) 832-3016
Martin.Bryan@areva.com

From: Tesfaye, Getachew [mailto:Getachew.Tesfaye@nrc.gov]
Sent: Wednesday, November 18, 2009 6:59 PM
To: ZZ-DL-A-USEPR-DL
Cc: Le, Hien; DeMarshall, Joseph; Kowal, Mark; Hearn, Peter; Colaccino, Joseph; ArevaEPRDCPEm Resource
Subject: U.S. EPR Design Certification Application RAI No. 315 (3878), FSARCh. 16 OPEN ITEM

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on October 21, 2009, and discussed with your staff on November 18, 2009. No changes were made to the draft RAI questions as a result of that discussion. The question in this RAI is an OPEN ITEM in the safety evaluation report for Chapter 16 for Phases 2 and 3 reviews. As such, the schedule we have established for your application assumes technically correct and complete responses prior to the start of Phase 4 review. For any RAI that cannot be answered prior to the start of Phase 4 review, it is expected that a date for receipt of this information will be provided so that the staff can assess how this information will impact the published schedule.

Thanks,

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

Hearing Identifier: AREVA_EPR_DC_RAIs
Email Number: 2785

Mail Envelope Properties (1F1CC1BBDC66B842A46CAC03D6B1CD41042B9609)

Subject: Response to U.S. EPR Design Certification Application RAI No. 315, FSAR Ch. 16 OPEN ITEM, Supplement 9
Sent Date: 3/31/2011 7:53:01 AM
Received Date: 3/31/2011 7:53:05 AM
From: WELLS Russell (AREVA)

Created By: Russell.Wells@areva.com

Recipients:
"LENTZ Tony (EXTERNAL AREVA)" <Tony.Lentz.ext@areva.com>
Tracking Status: None
"BENNETT Kathy (AREVA)" <Kathy.Bennett@areva.com>
Tracking Status: None
"DELANO Karen (AREVA)" <Karen.Delano@areva.com>
Tracking Status: None
"ROMINE Judy (AREVA)" <Judy.Romine@areva.com>
Tracking Status: None
"RYAN Tom (AREVA)" <Tom.Ryan@areva.com>
Tracking Status: None
"Tesfaye, Getachew" <Getachew.Tesfaye@nrc.gov>
Tracking Status: None

Post Office: AUSLYNCMX02.adom.ad.corp

Files	**Size**	**Date & Time**
MESSAGE	17138	3/31/2011 7:53:05 AM
RAI 315 Supplement 9 Response US EPR DC.pdf		112142

Options
Priority: Standard
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date:
Recipients Received:

Response to

Request for Additional Information No. 315 (3878), Supplement 9

11/18/2009

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 16 - Technical Specifications
Application Section: TS 3.3

QUESTIONS for Technical Specification Branch (CTSB)

Question 16-318:

OPEN ITEM

Follow-up to RAI 103, Question 16-137.

In RAI-SRP16-CTSB-103/137, the staff requested a technical justification regarding the omission of safety-related Reactor Trip (RT) signals in Table 3.3.1-2, Section A (Reactor Trip). FSAR Section 7.2.1.2 identifies the Safety Injection System (SIS) Actuation, Emergency Feedwater System (EFWS) Actuation, and the Manual RT signals from the Safety Information and Control System (SICS), as safety-related RT initiation signals. The applicant concludes that these RT initiation signals should not be included in Technical Specifications on the basis that 1) they are not credited in the EPR safety analysis as implied by their absence from Chapter 15 Tables 15.0-7 and 15.0-8, and 2) they do not satisfy Criterion 3 of 10 CFR 50.36 with regard to being part of the primary success path of a safety sequence analysis. NUREG-1431 includes both the Manual RT and the SIS Actuation initiation signals in comparable LCO 3.3.1, Reactor Trip System Instrumentation. The Manual RT initiation ensures that the control room operator has the capability to initiate a reactor trip at any time. This capability is critical whenever a parameter is rapidly trending toward its Trip Setpoint. Regarding the SIS Actuation, NUREG-1431 Bases B 3.3.1 specifically states that initiation of a reactor trip upon any signal that initiates a safety injection is a condition of acceptability for the LOCA. The EFWS Actuation is the primary success path which functions to mitigate the effects of a loss of Main Feedwater (MFW) event, providing a safety classified means to remove residual heat via the steam generators (SGs). FSAR Section 7.3.1.2.2 identifies a number of failure mechanisms that can result in a loss of MFW, including a Loss of Offsite Power, which is a highly credible event. In addition, it remains unclear how the applicant intends to ensure that surveillance testing requirements associated with the referenced safety-related trip signals will be met if they are not included in the Technical Specifications. The staff finds that the response does not provide the requisite technical justification to warrant exclusion of the safety-related RT initiation signals from Technical Specifications. This issue has been identified as an open item in the SER w/OI for Chapter 16 of the EPR FSAR

Response to Question 16-318:

This issue was further clarified on Page 16-20 of the NRC's March 10, 2010 Safety Evaluation, which states:

- In RAI 103, Question 16-137, the staff requested that the applicant provide a technical justification regarding the omission of safety-related RT signals in FSAR Tier 2, Table 3.3.1-2, Section A. FSAR Tier 2, Section 7.2.1.2, “Reactor Trip Functional Description,” identifies the Safety Injection System (SIS) Actuation, Emergency Feedwater System (EFWS) Actuation, and the Manual RT signals from the Safety Information and Control System (SICS), as safety-related RT initiation signals. In a March 19, 2009, response to RAI 103, Question 16-137, the applicant concluded that these RT initiation signals should not be included in TS on the basis that (1) they are not credited in the U.S. EPR safety analysis as implied by their absence from FSAR Tier 2, Chapter 15, “Transient and Accident Analyses,” FSAR Tier 2, Tables 15.0-7 and 15.0-8, and (2) they do not satisfy Criterion 3 of 10 CFR 50.36 with regard to being part of the primary success path of a safety-sequence analysis. NUREG-1431 includes both the Manual RT and the SIS Actuation initiation signals in comparable LCO 3.3.1, Reactor

Trip System Instrumentation. The Manual RT initiation ensures that the control room operator has the capability to initiate a reactor trip at any time. This capability is critical whenever a parameter is rapidly trending toward its Trip Setpoint. Regarding the SIS Actuation, NUREG-1431, Bases B 3.3.1 specifically states that initiation of a reactor trip upon any signal that initiates a safety injection is a condition of acceptability for the loss-of-coolant accident (LOCA). The EFWS Actuation is the primary success path, which functions to mitigate the effects of a loss of Main Feedwater (MFW) event, providing a safety classified means to remove residual heat via the steam generators. FSAR Tier 2, Section 7.3.1.2.2, “Emergency Feedwater System Actuation,” identifies a number of failure mechanisms that can result in a loss of MFW, including a loss of offsite power, which is a highly credible event. In addition, it remains unclear how the applicant intends to ensure that surveillance testing requirements associated with the referenced safety-related trip signals will be met if they are not included in the TS. The staff determined that the response does not provide the requisite technical justification to warrant exclusion of the safety-related RT initiation signals from TS.

Due to similarities in the two NRC Questions, this response will address both 16-318 and 16-319. This response supersedes AREVA's previous responses to RAI 103, Questions 16-137 and 16-160.

BACKGROUND

As specified in NUREG-0800, *Standard Review Plan*, Section 16.0, *Technical Specifications*:

> 10 CFR 52.47(a)(11) and 52.79(a)(30) provides that a design certification (DC) applicant and an combined license (COL) applicant are to propose Technical Specifications (TS) prepared in accordance with 10 CFR 50.36 and 50.36a. COL applicants that reference a certified plant design should propose TS based on the TS approved during the design certification (DC) review. The certified generic TS serve as the standard TS for the certified NSSS design.

As previously discussed in the response to RAI 103, Question 16-137:

> The required content of the Technical Specifications is specified in 10 CFR 50.36. The U.S. EPR Protection System and its reactor trip isolation signals satisfy Criterion 3 of 10 CFR 50.36:
>
> > "A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier."
>
> As discussed in the Final Policy Statement on Technical Specifications Improvements for Nuclear Power Plants (FR Doc. 93-17344):
>
> > "Discussion of Criterion 3: A third concept in the adequate protection of the public health and safety is that in the event that a postulated Design Basis Accident or Transient should occur, structures, systems, and components are available to function or to actuate in order to mitigate the consequence of the Design Basis Accident or Transient. Safety sequence analyses or their equivalent have been performed in recent years and

provide a method of presenting the plant response to an accident. These can be used to define the primary success paths.

> A safety sequence analysis is a systematic examination of the actions required to mitigate the consequences of events considered in the plant's Design Basis Accident and Transient analyses, as presented in Chapters 6 and 15 of the plant's FSAR (or equivalent chapters). Such a safety sequence analysis considers all applicable events, whether explicitly or implicitly presented. The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criteria), so that the plant response to Design Basis Accidents and Transients limits the consequences of these events to within the appropriate acceptance criteria.
>
> It is the intent of this criterion to capture into Technical Specifications only those structures, systems, and components that are part of the primary success path of a safety sequence analysis. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function. The primary success path for a particular mode of operation does not include backup and diverse equipment (e.g., rod withdrawal block which is a backup to the average power range monitor high flux trip in the startup mode, safety valves which are backup to low temperature overpressure relief valves during cold shutdown)."

<u>BASIS FOR U.S EPR PROTECTION SYSTEM TECHNICAL SPECIFICATONS</u>

In order to clarify the fundamental differences between the proposed U.S. EPR Protection System Technical Specifications and the Westinghouse Standard Technical Specifications for reactor trip and ESF functions, it should be noted that the proposed U.S. EPR Protection System Technical Specifications are component-based and address the components necessary for reactor trip and ESFAS functions. The Westinghouse Standard Technical Specifications are function-based and provide several LCOs for the reactor trip and ESF functions. This change in approach was necessary since:

1. In the Protection System, sensors and signal processors support multiple functions. Since failures in the plant would occur on a component basis, a component-based Technical Specification approach was determined to be advantageous since it specifies the required actions for the operators based on what component(s) have failed. Functional based Technical Specifications for the U.S. EPR Protection System would require additional operator analysis to determine what functions are affected by the failed component, which could delay implementation of the required actions. Delays would be especially important if the required action is to be taken immediately.

2. The Protection System for the U.S. EPR performs both the reactor trip and ESF functions. In many cases the same sensors and signal processors are utilized in both reactor trip and ESF functions. The Protection System Technical Specifications consist of one single Limiting Condition for Operation (LCO). Utilizing several LCOs, as is used in the Westinghouse Standard Technical Specifications, would result in needless duplications and could potentially result in conflicting requirements.

U.S. EPR FSAR Tier 2 Chapter 16 Limiting LCO 3.3.1 requires specific Protection System sensors, manual actuation switches, signal processors, and actuation devices to be operable. Thus, surveillance testing of these components is used in a series of sequential, overlapping or total divisional steps to ensure the operability of the Protection System, including the credited reactor trip and ESF functions that the system performs. The Protection System components that are required to be operable and the required surveillance testing for each component are specified in U.S. EPR FSAR Tier 2 Chapter 16, Technical Specifications, Table 3.3.1-1, *Protection* System *Sensors, Manual Actuation Switches, Signal Processors, and Actuation Devices*. The required number of components, required modes, and surveillance testing specified for each component envelopes the requirements for the credited reactor trip and ESF functions supported by each component. A summary of the generic strategy for periodic surveillance testing of the Protection System was provided in response to RAI 103, Question 16-193. In the Westinghouse Standard Technical Specification, required modes and surveillances are specified on a function by function basis. Both approaches ensure that reactor trip and ESF functions will maintain the Safety Limits during all design basis events and anticipated operational occurrences in the required modes.

However, as a result of the component-based approach taken for the U.S. EPR Protection System Technical Specifications, the listing of functions in U.S. EPR FSAR Tier 2 Chapter 16, Technical Specifications, Table 3.3.1-2, *Acquisition and Processing Unit Requirements Referenced from Table 3.3.1-1*, does not serve the same purpose as the listing of reactor trip and ESF functions in the function-based Westinghouse Standard Technical Specifications. In the Westinghouse Standard Technical Specifications, LCOs 3.3.1, "Reactor Trip System (RTS) Instrumentation," 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," 3.3.5, "Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation," LCO 3.3.6, "Containment Purge and Exhaust Isolation Instrumentation," LCO 3.3.7, "Control Room Emergency Filtration System (CREFS) Actuation Instrumentation," 3.3.8, "Fuel Building Air Cleanup System (FBACS) Actuation Instrumentation," and 3.3.9, "Boron Dilution Protection System (BDPS)," require "instrumentation for each Function" to be operable and surveillances are specified on a function by function basis. As discussed above, the U.S. EPR Protection System Technical Specifications requires the specific Protection System sensors, manual actuation switches, signal processors, and actuation devices specified in Table 3.3.1-1 to be operable. Table 3.3.1-2 is referenced from Table 3.3.1-1 and is only entered when the number of operable Acquisition and Processing Units (APUs) drops below the minimum number required for functional capability. There were two purposes served by including the listing of reactor trip and ESF functions in Table 3.3.1-2 of the U.S. EPR Technical Specifications when they were originally developed:

- The primary purpose for Table 3.3.1-2 was to provide the location for setpoint values for the specific reactor trip and ESF functions. 10 CFR 50.36(d) states that the Technical Specifications will include items in the following categories: (1) Safety limits, limiting safety system settings, and limiting control settings. Regulatory Guide 1.105, Setpoints for Safety-Related Instrumentation, Revision 3, Regulatory Position C.3 states:

  > Section 4.3 of ISA-S67.04-1994 states that the limiting safety system setting (LSSS) may be maintained in technical specifications or appropriate plant procedures. However, 10 CFR 50.36 states that the technical specifications will include items in the categories of safety limits, limiting safety system settings, and limiting control settings. Thus, the LSSS may not be maintained in plant procedures. Rather, the LSSS must be specified as a technical-specification-defined limit in order to satisfy

> the requirements of 10 CFR 50.36. The LSSS should be developed in accordance with the setpoint methodology set forth in the standard, **with the LSSS listed in the technical specifications**. (Emphasis added)

> The U.S. EPR utilizes a digitally-based Protection System. In the U.S. EPR, setpoints are not physically adjusted parameters in hardwired components. Rather, the setpoints are values programmed into the software contained in the APU. Hence, the setpoints for each function performed by the APU are listed in the table that contains the required actions (Conditions) for the APUs. Thus, Table 3.3.1-2 satisfies the legal requirements for including setpoints in the Technical Specifications and would require the APU associated with the function to be declared inoperable if the setpoint for the function was determined to be incorrect or programmed into the APU incorrectly.

There are five APUs in each division of the Protection System (APUs A1, A2, A3, B1, and B2). Reactor trip and ESF functions are allocated amongst the five APUs as part of the detailed design process. The functions performed by each APU are the same for that APU in each division (i.e., APU A1 performs the same functions in each of the four divisions). Table 3.3.1-2 specifies that three divisions of APUs are required for the performance of each function. The structure of Table 3.3.1-2 implicitly allows a different APU to be inoperable in each division as long as three of the same APUs are operable (i.e., it is permissible to have APU A1 inoperable in Division 1, APU A2 inoperable in Division 2, and APU A3 inoperable in Division 3, since there are always three divisions of APUs operable for each function). Thus, inclusion of Table 3.3.1-2 provides flexibility in requiring three divisions of APUs be operable to support each function, while not requiring the functions to be performed by each APU to be specified at this time.

In practical terms, if the setpoints were relocated out of Table 3.3.1-2 (e.g., using a Setpoint Control Program type approach), Table 3.3.1-2 could be deleted and required actions for inoperable APUs could be specified in Table 3.3.1-1. This is the approach taken for Actuation Logic Units (ALUs) in Table 3.3.1-1. There would only be a loss in flexibility in that the failure of different APUs in different divisions (as described in the second bullet, above) would not be permitted without entering a Condition.

Additional discussions regarding the Reactor Trip on Safety Injection System (SIS) Actuation, Reactor Trip on Emergency Feedwater System (EFWS) Actuation, Manual Reactor Trip, and Emergency Feedwater System (EFWS) Isolation on High SG Level (Affected SGs) Engineered Safety Features Actuation System (ESFAS) functions discussed in the NRC's Questions are provided below:

<u>SPECIFIC FUNCTIONS DISCUSSED BY NRC</u>

Reactor Trip on Safety Injection System (SIS) Actuation

In the follow-up question, the NRC states that:

> FSAR Section 7.2.1.2 identifies the Safety Injection System (SIS) Actuation as a safety-related Reactor Trip initiation signal. …

> NUREG-1431 includes both the Manual Reactor Trip and the SIS Actuation initiation signals in comparable LCO 3.3.1, Reactor Trip System Instrumentation. … NUREG-1431 Bases B

> 3.3.1 specifically states that initiation of a reactor trip upon any signal that initiates a safety injection is a condition of acceptability for the LOCA. …
>
> In addition, it remains unclear how the applicant intends to ensure that surveillance testing requirements associated with the referenced safety-related trip signals will be met if they are not included in the Technical Specifications.

The Reactor Trip on SIS Actuation is a safety related design feature of the U.S. EPR. It is described in U.S. EPR FSAR Tier 2 Section 7.2.1.2.20 and is depicted in Figure 7.3-2. Section 7.2 reflects the design of the U.S. EPR Protection System. The Protection System design includes features that are not credited in the safety analysis. If the Reactor Trip on SIS Actuation function was deleted from the U.S. EPR design, there would be no impact to the safety analysis as summarized in Chapter 15 of the U.S. EPR FSAR. 10 CFR 50.36, Criterion 3, does not require safety related design features to be incorporated into the Technical Specifications unless the design features are credited in the safety analysis.

While AREVA has no direct knowledge of the proprietary safety analysis that supports the Westinghouse Standard Technical Specifications, AREVA assumes that the Westinghouse Standard Technical Specifications reflect the safety analysis Westinghouse used to support its design. The safety analysis used by AREVA for the U.S. EPR is summarized in U.S. EPR FSAR Tier 2 Chapter 15. Specifically, the analysis of Loss of Coolant Accidents (LOCAs) resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary are summarized in U.S. EPR FSAR Tier 2 Section 15.6.5. The following reactor trip functions are credited to mitigate LOCAs in the U.S. EPR safety analysis:

- Low Pressurizer Pressure,
- Low Hot Leg Pressure, and
- High Containment pressure.

The Reactor Trip on SIS Actuation is not credited in the U.S. EPR safety analysis to mitigate the consequences of a LOCA. 10 CFR 50.36 requires an applicants Technical Specifications to reflect its safety analysis. It does not require an applicant to incorporate functions in its Technical Specifications that another vendor assumed in their safety analysis. Incorporation of functions included by Westinghouse in their Standard Technical Specification into the U.S. EPR Technical Specifications is not supported by the requirements specified in 10 CFR 50.36 with regards to the content of U.S. EPR Technical Specifications.

With regards to surveillance testing, there are three Engineered Safety Features Actuation System Signals associated with Safety Injection System (SIS) actuation listed in the U.S. EPR Protection System Technical Specifications:

- SIS Actuation on Low Pressurizer Pressure,
- SIS Actuation on Low Delta Psat, and
- SIS Actuation on Low RCS Loop Level.

The sensors necessary to detect the conditions that necessitate an SIS actuation, the signal processors that generate the SIS actuation signal, and their surveillance requirements are listed in Table 3.3.1-1 of the Protection System Technical Specifications. Also included is a surveillance requirement to periodically verify the setpoints have been properly loaded into the signal processors. Similarly, the sensors, signal processors, and actuation devices necessary to perform reactor trip functions are also listed in Table 3.3.1-1 of the Protection System Technical Specifications. While the Reactor Trip on SIS Actuation function is not required to be included in the U.S. EPR Technical Specifications, the components necessary to perform this function and the verification of the actuation setpoints are already the subject of other surveillance requirements for other credited functions.

In addition, U.S. EPR FSAR Tier 2 Table 1.8-2 contains U.S. EPR Combined License Information Item 13.5-5, which requires a Combined License applicant that references the U.S. EPR design certification provide site-specific information for administrative, operating, emergency, maintenance, and other operating procedures. Information regarding testing procedures not required to be included in the Technical Specifications may be requested from the Combined License applicant.

Reactor Trip on Emergency Feedwater System (EFWS) Actuation

The logic for not including the Reactor Trip on EFWS Actuation in the U.S. EPR Technical Specifications is generally the same as that for the Reactor Trip on SIS Actuation discussed above.

In the follow-up question, the NRC states that:

> FSAR Section 7.2.1.2 identifies the EFWS Actuation as a safety-related Reactor Trip initiation signal. …
>
> The EFWS Actuation is the primary success path which functions to mitigate the effects of a loss of Main Feedwater (MFW) event, providing a safety classified means to remove residual heat via the steam generators (SGs). FSAR Section 7.3.1.2.2 identifies a number of failure mechanisms that can result in a loss of MFW, including a Loss of Offsite Power, which is a highly credible event. …
>
> In addition, it remains unclear how the applicant intends to ensure that surveillance testing requirements associated with the referenced safety-related trip signals will be met if they are not included in the Technical Specifications.

Reactor trip functions for the U.S. EPR insert the Rod Cluster Control Assemblies (RCCAs) into the reactor core. ESF functions are used to initiate Emergency Core Cooling Systems. There are two credited ESF functions that actuate EFW:

- EFWS Actuation on Low-Low SG Level (Affected SG), and
- EFWS Actuation on Loss of Offsite Power and SIS Actuation (All SGs).

The Reactor Trip on EFWS Actuation is a safety related design feature of the U.S. EPR. It is described in U.S. EPR FSAR Tier 2 Section 7.2.1.2.21 and is depicted in Figure 7.3-3. As discussed in the NRC's Final Policy Statement on Technical Specifications Improvements for Nuclear Power Plants, 10 CFR 50.36, Criterion 3, does not require safety related design features to be incorporated into the Technical Specifications unless the features are credited in the safety analysis.

Section 7.2 reflects the Protection System design. The Protection System design includes features that are not credited in the safety analysis. The safety analysis used by AREVA for the U.S. EPR is summarized in U.S. EPR FSAR Tier 2 Chapter 15. Specifically, the loss of normal Feedwater flow is addressed in Section 15.2.7 and Feedwater piping breaks inside and outside containment are addressed in Section 15.2.8.

The following reactor trip functions are credited to mitigate the loss of normal Feedwater flow in the U.S. EPR safety analysis:

- Low Steam Generator Level, and
- Low DNBR

The following reactor trip functions are credited to mitigate the entire spectrum of Feedwater piping breaks inside and outside containment in the U.S. EPR safety analysis:

- Steam Generator Pressure Drop,
- Low Steam Generator Pressure,
- Low Steam Generator Level,
- High Containment Pressure, and
- Low DNBR.

The Reactor Trip on EFWS Actuation is not credited in the U.S. EPR safety analysis to mitigate the consequences of a LOCA or any other design basis event or abnormal operational occurrence. 10 CFR 50.36 requires an applicants Technical Specifications to reflect its safety analysis. The implication by the NRC that AREVA needs to incorporate into its Technical Specifications functions that are not credited in their safety analysis is not supported by the requirements specified in 10 CFR 50.36.

With regards to surveillance testing, the sensors credited with detecting the conditions that necessitate a reactor trip to mitigate the loss of normal Feedwater flow or Feedwater piping breaks inside and outside containment, the signal processors that generate the reactor trip signal, the reactor trip actuation devices, and their surveillance requirements are listed in Table 3.3.1-1 of the Protection System Technical Specifications. Also included is a surveillance requirement to periodically verify the reactor trip setpoints have been properly loaded into the signal processors. Also listed in Table 3.3.1-1 are the sensors, signal processors, and actuation devices credited in the safety analysis for initiating EFWS Actuation on Low-Low SG Level (Affected SG) and EFWS Actuation on Loss of Offsite Power and SIS Actuation (All SGs) ESFAS functions. While the Reactor Trip on EFWS Actuation function is not required to be included in the U.S. EPR Technical Specifications, the components necessary to perform this function and the verification of the actuation setpoints are already the subject of other surveillance requirements for other credited functions.

In the NRC Question, the Staff states that it is unclear how the applicant intends to ensure that surveillance testing requirements associated with the referenced safety-related trip signals will be met if they are not included in the Technical Specifications. The details of periodic testing for non-credited equipment are not the responsibility of the DC applicant. U.S. EPR FSAR Tier 2 Table 1.8-2 contains U.S. EPR Combined License Information Item 13.5-5, which requires a Combined License applicant that references the U.S. EPR design certification provide site-specific information for administrative, operating, emergency, maintenance, and other operating procedures. Information regarding testing procedures not included in the Technical Specifications may be requested from the Combined License applicant.

Manual Reactor Trip

The context of the original Question 16-137 was in regard to the manual reactor trip signal not being listed as a separate function in Table 3.3.1-2.

In the follow-up question, the NRC states that:

> FSAR Section 7.2.1.2 identifies the manual reactor trip signal from the Safety Information and Control System (SICS) as a safety-related reactor trip initiation signal. …
>
> The Manual RT initiation ensures that the control room operator has the capability to initiate a reactor trip at any time. This capability is critical whenever a parameter is rapidly trending toward its Trip Setpoint.
>
> In addition, it remains unclear how the applicant intends to ensure that surveillance testing requirements associated with the referenced safety-related trip signals will be met if they are not included in the Technical Specifications.

The manual reactor trip is a safety related design feature of the U.S. EPR and is described in U.S. EPR FSAR Tier 2 Section 7.2.1.2.22 and depicted in Figure 7.2-3. As shown in Figure 7.2-3, the manual reactor trip is initiated by a switch in the Main Control Room which goes directly to the Reactor Trip Breakers. The manual reactor trip actuation switches and their surveillance requirements are listed in Table 3.3.1-1 of the Protection System Technical Specifications. Similarly, the actuation devices necessary to perform reactor trip functions are also listed in Table 3.3.1-1 of the Protection System Technical Specifications. The surveillances on the manual reactor trip actuation switches and the Reactor Trip Breakers ensure that a manual reactor trip will occur when initiated.

The manual reactor trip switches do not provide a signal to the APUs and there is no software "function" for the manual reactor trip loaded in the APUs. The list of functions in Table 3.3.1-2 only contains the credited reactor trip and ESF software functions performed by the APUs. Since the U.S. EPR Protection System Technical Specifications are component based, the format does not readily allow the listing of a function in Table 3.3.1-2 that is not performed by the APUs.

EFWS Isolation on High SG Level (Affected SG) ESFAS Signal

The logic for not including the automatic EFWS Isolation on High SG Level (Affected SG) ESFAS function in Table 3.3.1-2 of the U.S. EPR Technical Specifications is generally the same as that for the Reactor Trip on SIS Actuation and Reactor Trip on EFWS Actuation signals discussed above.

Both the automatic EFWS Isolation on High SG Level (Affected SG) ESFAS and manual SG Isolation signals are safety related design features of the U.S. EPR. The automatic EFWS Isolation on High SG Level (Affected SG) function is described in U.S. EPR FSAR Tier 2 Section 7.3.1.2.3 and depicted in Figures 7.3-5 through 7.3-7. The manual SG Isolation function is discussed in U.S. EPR FSAR Tier 2 Section 7.3.1.2.14 and depicted in Figures 7.3-25 and 7.3-26. The manual SG Isolation function includes the following actions:

- MSIV, MSIV bypass, and SG blowdown closure,
- MFW and SSS isolation, and
- EFWS isolation (Confirmatory action).

Section 7.3 reflects the design of the U.S. EPR Protection System. The Protection System design includes features that are not credited in the safety analysis. If the automatic EFWS Isolation on High SG Level (Affected SG) ESFAS function was deleted from the U.S. EPR design, there would be no impact to the credited safety analysis path summarized in Chapter 15 of the U.S. EPR FSAR. 10 CFR 50.36, Criterion 3, does not require safety related design features to be incorporated into the Technical Specifications unless the design features are credited in the safety analysis.

In the follow-up question, the NRC states that:

> If the EFWS system is actuated to mitigate the effects of a loss of Main Feedwater (MFW) event, then isolation of the EFWS system is considered the primary success path for mitigating a SGTR.

The safety analysis for the U.S. EPR Steam Generator Tube Rupture (SGTR) event is summarized in U.S. EPR FSAR Tier 2 Section 15.6.3.

The EFWS Isolation function is not credited with automatically mitigating the effects of a SGTR. Rather, manual operator action to isolate the EFWS is the primary success path credited in Chapter 15 for mitigating a SGTR. Please note that in RAI 285, Question 07.03-27, AREVA was requested to clarify its technical position concerning manual initiation of a steam generator isolation due to a steam generator tube rupture event.

With regards to surveillance testing, the U.S. EPR Technical Specifications contain surveillance requirements that ensure credited automatic and manual actions are operable. The credited manual actuation switches used for SG Isolation are specified in Table 3.3.1-1 of the Protection System Technical Specifications. As discussed in U.S. EPR FSAR Tier 2 Section 7.3.1.2.14, SG Isolation includes the isolation of Main Feedwater, Startup and Shutdown, and Emergency Feedwater systems, Main Steam Isolation Valve, Main Steam Isolation Valve bypass, and SG blowdown closure. The surveillance requirements specified in Table 3.3.1-1 for the manual SG Isolation switches and in Limiting Condition for Operation 3.7.5, "Emergency Feedwater System," are sufficient to ensure the ability of the operator to isolate the EFWS. The surveillances for the instrumentation in LCO 3.3.1 ensure that the manual isolation signal is operable and the surveillances for the mechanical equipment in LCO 3.7.5 ensure the EFW isolation valves will operate. The manual isolation signal is not routed through the APU. There is no software "function" for the manual SG Isolation loaded in the APUs. As previously discussed, the list of functions in Table 3.3.1-2 only applies to the reactor trip and ESF software functions performed by the APUs. Since the U.S. EPR Protection System Technical Specifications are component based, the format does not readily allow the listing of a function in Table 3.3.1-2 that is not performed by the APUs. Therefore, credited manual actions are addressed in the Instrumentation Technical Specifications by the inclusion of the associated components (i.e., the manual switches) in Table 3.3.1-1. Manual actions are not specified as functions, which are listed in Table 3.3.1-2.

While the automatic EFWS Isolation on High SG Level (Affected SG) ESFAS function is not required to be included in the U.S. EPR Technical Specifications, the components necessary to perform this function (i.e., Steam Generator Level (Wide Range) sensors, APUs, ALUs, and the Hot Leg Temperature (Wide Range) sensors that provide input to associated Permissive P13) are listed in Table 3.3.1-1 of the Protection System Technical Specifications and are already the subject of other surveillance requirements for other credited functions.

In the NRC Question, the Staff states that it is unclear how the applicant intends to ensure that surveillance testing requirements associated with the referenced safety-related trip signals will be met if they are not included in the Technical Specifications. The details of periodic testing for non-credited equipment are not the responsibility of the DC applicant. U.S. EPR FSAR Tier 2 Table 1.8-2 contains U.S. EPR Combined License Information Item 13.5-5, which requires a Combined License applicant that references the U.S. EPR design certification provide site-specific information for administrative, operating, emergency, maintenance, and other operating procedures. Information regarding testing procedures not included in the Technical Specifications may be requested from the Combined License applicant.

SUMMARY

In summary, the Reactor Trip on SIS Actuation, Reactor Trip on EFWS Actuation, and EFWS Isolation on High SG Level functions are not credited in the safety analysis and are not required by 10 CFR 50.36, Criterion 3, to be included in the U.S. EPR Technical Specifications. However, the components that perform these functions are already demonstrated operable by surveillances specified in the Protection System Technical Specifications for other credited reactor trip and ESFAS functions.

The switches that initiate the manual reactor trip and EFWS isolation functions via the SG Isolation switches are demonstrated operable by surveillances specified in the component based Protection System Technical Specifications. The component based format utilized for the Protection System Technical Specifications does not allow a manual reactor trip function to be listed, since it is not a software function performed by the APUs.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 16-319:

OPEN ITEM

Follow-up to RAI 103, Question 16-160.

In RAI-SRP16-CTSB-103/160, the staff requested an explanation regarding the mode applicability for Hot Leg Temperature Wide Range (WR) instrumentation with respect to ESFAS Function B.6.c, Emergency Feedwater System (EFWS) Isolation on High SG Level (Affected SGs). Although this issue is identified and addressed under RAI-SRP16-CTSB-103/144, the staff questions the applicant's removal of the EFWS Isolation on High SG Level function from Technical Specifications as indicated in the response to Question 16-160 on page 30 of 63. The applicant concludes that ESFAS Function B.6.c should not be included in Technical Specifications on the basis that 1) the function is no longer credited in U.S. EPR FSAR Tier 2, Table 15.0-8 and 2) Manual operator action is assumed to mitigate a SG tube rupture (SGTR) event with no automatic actions. The EFWS Isolation function automatically mitigates the effects of a SGTR. The EFWS is isolated at a high level setpoint to avoid an uncontrolled SG level increase, subsequent SG overfill, and potential radioactive water discharge via the main steam relief train. If the EFWS system is actuated to mitigate the effects of a loss of Main Feedwater (MFW) event, then isolation of the EFWS system is considered the primary success path for mitigating a SGTR. In addition, the applicant has not demonstrated that the surveillance testing requirements associated with the EFWS Isolation function are met if they are not included in the Technical Specifications. Exclusion from Table 15.0-8 and reliance upon manual operator action to avoid an uncontrolled SG level increase and potential radioactive discharge, do not necessarily warrant exclusion of the EFWS Isolation function from the Technical Specifications. This issue has been identified as an open item in the SER w/OI for Chapter 16 of the EPR FSAR.

Response to Question 16-319:

This issue was further clarified on Page 16-20 of the NRC's March 10, 2010 Safety Evaluation, which states:

- In RAI 103, Question 16-160, the staff requested that the applicant provide an explanation regarding the mode applicability for Hot Leg Temperature Wide Range instrumentation with respect to ESFAS Function B.6.c, "Emergency Feedwater System Isolation on High-SG Level (Affected SGs)." Although this issue is identified and addressed under RAI 103, Question 16-144, the staff questions the applicant's removal of the EFWS Isolation on High-SG Level function from Technical Specifications as indicated in the response to RAI 103, Question 16-160 on Page 30 of 63. In a June 30, 2009, response to RAI 103, Question 16-160, the applicant concluded that ESFAS Function B.6.c should not be included in TS on the basis that (1) the function is no longer credited in FSAR Tier 2, Table 15.0-8, "Engineered Safety Features Functions Used in the Accident Analysis," and (2) Manual operator action is assumed to mitigate a steam generator tube rupture (SGTR) event with no automatic actions. The EFWS Isolation function automatically mitigates the effects of a SGTR. The EFWS is isolated at a high-level setpoint to avoid an uncontrolled SG level increase, subsequent SG overfill, and potential radioactive water discharge via the main steam relief train. If the EFWS system is actuated to mitigate the effects of a loss of Main Feedwater event, then isolation of the EFWS system is considered the primary success path for mitigating a

> SGTR. In addition, it remains unclear how the applicant intends to ensure that surveillance testing requirements associated with the EFWS Isolation function will be met if they are not included in the TS. Exclusion from FSAR Tier 2, Table 15.0-8 and reliance upon manual operator action to avoid an uncontrolled SG level increase and potential radioactive discharge do not necessarily warrant exclusion of the EFWS Isolation function from the Technical Specifications.

Refer to the response to Question 16-318.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.