ArevaEPRDCPEm Resource

From: WILLIFORD Dennis (AREVA) [Dennis.Williford@areva.com]
Sent: Wednesday, May 25, 2011 12:11 PM
To: Tesfaye, Getachew
Cc: BENNETT Kathy (AREVA); DELANO Karen (AREVA); ROMINE Judy (AREVA); RYAN Tom (AREVA)
Subject: Response to U.S. EPR Design Certification Application RAI No. 442, FSAR Ch. 7, Supplement 11
Attachments: RAI 442 Supplement 11 Response US EPR DC.pdf

Getachew,

On November 19, 2010, AREVA NP Inc. (AREVA NP) provided a schedule for a technically correct and complete response to the questions in RAI 442. Supplement 1 response was sent on January 7, 2011 to provide a revised schedule for four of the questions. Supplement 2 response was sent on February 9, 2011 to provide a revised schedule. Supplement 3 response was sent on February 18, 2011 to provide technically correct and complete responses to four questions. Supplement 4 response was sent on February 25, 2011 to provide technically correct and complete response to one question. Supplement 5 response was sent on March 2, 2011 to provide technically correct and complete responses to three of the 12 remaining questions. Supplement 6 response was sent on March 15, 2011 to provide technically correct and complete responses to two of the 9 remaining questions. Supplement 7 response was sent on April 5, 2011 to provide a revised schedule. Supplement 8 response was sent on April 25, 2011 to provide a revised schedule. Supplement 9 response was sent on May 20, 2011 to provide a technically correct and complete response to one of the 7 remaining questions. Supplement 10 response was sent on May 25, 2011 to provide a technically correct and complete response to one of the 6 remaining questions.

The attached file, "RAI 442 Supplement 11 Response US EPR DC.pdf" provides a technically correct and complete response to Question 07.01-28, as committed. Appended to this file are affected pages of the U.S. EPR Final Safety Analysis Report in redline-strikeout format to support the response to RAI 442, Question 07.01-28.

The following table indicates the respective pages in the attachment that contain AREVA NP's response to the subject question.

Question #	Start Page	End Page
RAI 442 07.01-28	2	3

AREVA NP's schedule for providing a technically correct and complete response to the remaining 4 questions in RAI 442 is unchanged and is provided below.

Question #	Response Date
RAI 442 — 7.1-26	May 27, 2011
RAI 442 — 7.1-30	May 27, 2011
RAI 442 — 7.1-31	May 27, 2011
RAI 442 — 7.3-32	May 27, 2011

Sincerely,

Dennis Williford, P.E.
U.S. EPR Design Certification Licensing Manager
AREVA NP Inc.

7207 IBM Drive, Mail Code CLT 2B
Charlotte, NC 28262
Phone: 704-805-2223
Email: Dennis.Williford@areva.com

From: WILLIFORD Dennis (RS/NB)
Sent: Wednesday, May 25, 2011 7:42 AM
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. 442, FSAR Ch. 7, Supplement 10

Getachew,

On November 19, 2010, AREVA NP Inc. (AREVA NP) provided a schedule for a technically correct and complete response to the questions in RAI 442. Supplement 1 response was sent on January 7, 2011 to provide a revised schedule for four of the questions. Supplement 2 response was sent on February 9, 2011 to provide a revised schedule. Supplement 3 response was sent on February 18, 2011 to provide technically correct and complete responses to four questions. Supplement 4 response was sent on February 25, 2011 to provide technically correct and complete response to one question. Supplement 5 response was sent on March 2, 2011 to provide technically correct and complete responses to three of the 12 remaining questions. Supplement 6 response was sent on March 15, 2011 to provide technically correct and complete responses to two of the 9 remaining questions. Supplement 7 response was sent on April 5, 2011 to provide a revised schedule. Supplement 8 response was sent on April 25, 2011 to provide a revised schedule. Supplement 9 response was sent on May 20, 2011 to provide a technically correct and complete response to one of the 7 remaining questions.

The attached file, "RAI 442 Supplement 10 Response US EPR DC.pdf" provides a technically correct and complete response to Question 07.01-27, as committed. Appended to this file are affected pages of the U.S. EPR Final Safety Analysis Report in redline-strikeout format which support the response to RAI 442, Question 07.01-27.

The following table indicates the respective pages in the attachment that contain AREVA NP's response to the subject question.

Question #	Start Page	End Page
RAI 442 07.01-27	2	2

AREVA NP's schedule for providing a technically correct and complete response to the remaining 5 questions in RAI 442 remains unchanged and is provided below.

Question #	Response Date
RAI 442 — 7.1-26	May 27, 2011
RAI 442 — 7.1-28	May 27, 2011
RAI 442 — 7.1-30	May 27, 2011
RAI 442 — 7.1-31	May 27, 2011
RAI 442 — 7.3-32	May 27, 2011

Sincerely,

Dennis Williford, P.E.
U.S. EPR Design Certification Licensing Manager
AREVA NP Inc.
7207 IBM Drive, Mail Code CLT 2B
Charlotte, NC 28262
Phone: 704-805-2223
Email: Dennis.Williford@areva.com

From: WILLIFORD Dennis (RS/NB)
Sent: Friday, May 20, 2011 5:32 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. 442, FSAR Ch. 7, Supplement 9

Getachew,

On November 19, 2010, AREVA NP Inc. (AREVA NP) provided a schedule for a technically correct and complete response to the questions in RAI 442. Supplement 1 response was sent on January 7, 2011 to provide a revised schedule for four of the questions. Supplement 2 response was sent on February 9, 2011 to provide a revised schedule. Supplement 3 response was sent on February 18, 2011 to provide technically correct and complete responses to four questions. Supplement 4 response was sent on February 25, 2011 to provide technically correct and complete response to one question. Supplement 5 response was sent on March 2, 2011 to provide technically correct and complete responses to three of the 12 remaining questions. Supplement 6 response was sent on March 15, 2011 to provide technically correct and complete responses to two of the 9 remaining questions. Supplement 7 response was sent on April 5, 2011 to provide a revised schedule. Supplement 8 response was sent on April 25, 2011 to provide a revised schedule.

The attached file, "RAI 442 Supplement 9 Response US EPR DC.pdf" provides a technically correct and complete response to Question 07.09-64, as committed. Appended to this file are affected pages of the U.S. EPR Final Safety Analysis Report in redline-strikeout format which support the response to RAI 442, Question 07.09-64.

The following table indicates the respective pages in the enclosure that contain AREVA NP's response to the subject question.

Question #	Start Page	End Page
RAI 442 07.09-64	2	3

AREVA NP's schedule for providing a technically correct and complete response to the remaining 6 questions in RAI 442 remains unchanged and is provided below.

Question #	Response Date
RAI 442 — 7.1-26	May 27, 2011
RAI 442 — 7.1-27	May 27, 2011
RAI 442 — 7.1-28	May 27, 2011
RAI 442 — 7.1-30	May 27, 2011
RAI 442 — 7.1-31	May 27, 2011
RAI 442 — 7.3-32	May 27, 2011

Sincerely,

Dennis Williford, P.E.
U.S. EPR Design Certification Licensing Manager
AREVA NP Inc.
7207 IBM Drive, Mail Code CLT 2B
Charlotte, NC 28262
Phone: 704-805-2223
Email: Dennis.Williford@areva.com

From: WELLS Russell (RS/NB)
Sent: Monday, April 25, 2011 4:43 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. 442, FSAR Ch. 7, Supplement 8

Getachew,

On November 19, 2010, AREVA NP Inc. (AREVA NP) provided a schedule for a technically correct and complete response to the questions in RAI 442. Supplement 1 response was sent on January 7, 2011 to provide a revised schedule for four of the questions. Supplement 2 response was sent on February 9, 2011 to provide a revised schedule. Supplement 3 response was sent on February 18, 2011 to provide technically correct and complete responses to four questions. Supplement 4 response was sent on February 25, 2011 to provide technically correct and complete response to one question. Supplement 5 response was sent on March 2, 2011 to provide technically correct and complete responses to three of the 12 remaining questions. Supplement 6 response was sent on March 15, 2011 to provide technically correct and complete responses to two of the 9 remaining questions. Supplement 7 response was sent on April 5, 2011 to provide a revised schedule.

To provide additional time to interact with the NRC, a revised schedule is provided in this e-mail.

AREVA NP's schedule for providing a technically correct and complete response to the remaining questions in RAI 442 is provided below.

Question #	Response Date
RAI 442 — 7.1-26	May 27, 2011
RAI 442 — 7.1-27	May 27, 2011
RAI 442 — 7.1-28	May 27, 2011
RAI 442 — 7.1-30	May 27, 2011
RAI 442 — 7.1-31	May 27, 2011
RAI 442 — 7.3-32	May 27, 2011
RAI 442 — 7.9-64	May 27, 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, April 05, 2011 10:56 AM
To: 'Getachew.Tesfaye@nrc.gov'
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. 442, FSAR Ch. 7, Supplement 7

Getachew,

On November 19, 2010, AREVA NP Inc. (AREVA NP) provided a schedule for a technically correct and complete response to the questions in RAI 442. Supplement 1 response was sent on January 7, 2011 to provide a revised schedule for four of the questions. Supplement 2 response was sent on February 9, 2011 to provide a revised schedule. Supplement 3 response was sent on February 18, 2011 to provide technically correct and complete responses to four questions. Supplement 4 response was sent on February 25, 2011 to provide technically correct and complete response to one question. Supplement 5 response was sent on March 2, 2011 to provide technically correct and complete responses to three of the 12 remaining questions. Supplement 6 response was sent on March 15, 2011 to provide technically correct and complete responses to two of the 9 remaining questions.

To provide additional time to interact with the NRC, a revised schedule is provided in this e-mail.

AREVA NP's schedule for providing a technically correct and complete response to the remaining questions in RAI 442 is provided below.

Question #	Response Date
RAI 442 — 7.1-26	**April 28, 2011**
RAI 442 — 7.1-27	**April 28, 2011**
RAI 442 — 7.1-28	**April 28, 2011**
RAI 442 — 7.1-30	April 28, 2011
RAI 442 — 7.1-31	**April 28, 2011**
RAI 442 — 7.3-32	**April 28, 2011**
RAI 442 — 7.9-64	April 28, 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 15, 2011 12:51 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. 442, FSAR Ch. 7, Supplement 6

Getachew,

On November 19, 2010, AREVA NP Inc. (AREVA NP) provided a schedule for a technically correct and complete response to the questions in RAI 442. Supplement 1 response was sent on January 7, 2011 to provide a revised schedule for four of the questions. Supplement 2 response was sent on February 9, 2011 to provide a revised schedule. Supplement 3 response was sent on February 18, 2011 to provide technically correct and complete responses to four questions. Supplement 4 response was sent on February 25, 2011 to provide technically correct and complete response to one question. Supplement 5 response was sent on March 2, 2011 to provide technically correct and complete responses to three of the 12 remaining questions. Based on discussions with NRC, the attached file, "RAI 442 Supplement 6 Response US EPR DC.pdf" provides technically correct and complete responses to two of the 9 questions, as committed.

The following table indicates the respective pages in the response document, "RAI 442 Supplement 6 Response US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 442 07.01-32	2	3
RAI 442 07.09-67	4	5

AREVA NP's schedule for providing a technically correct and complete response to the remaining questions in RAI 442 remains unchanged and is provided below.

Question #	Response Date
RAI 442 — 7.1-26	April 21, 2011
RAI 442 — 7.1-27	April 14, 2011
RAI 442 — 7.1-28	April 7, 2011
RAI 442 — 7.1-30	April 28, 2011
RAI 442 — 7.1-31	April 7, 2011
RAI 442 — 7.3-32	April 14, 2011
RAI 442 — 7.9-64	April 28, 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: Wednesday, March 02, 2011 4:52 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. 442, FSAR Ch. 7, Supplement 5

Getachew,

On November 19, 2010, AREVA NP Inc. (AREVA NP) provided a schedule for a technically correct and complete response to the questions in RAI 442. Supplement 1 response was sent on January 7, 2011 to provide a revised schedule for four of the questions. Supplement 2 response was sent on February 9, 2011 to provide a revised schedule. Supplement 3 response was sent on February 18, 2011 to provide technically correct and complete responses to four questions. Supplement 4 response was sent on February 25, 2011 to provide technically correct and complete response to one question. Based on discussions with NRC, the attached file, "RAI 442 Supplement 5 Response US EPR DC.pdf" provides technically correct and complete responses to three of the 12 questions, as committed.

The following table indicates the respective pages in the response document, "RAI 442 Supplement 5 Response US EPR DC.pdf," that contain AREVA NP's response to the subject question.

Question #	Start Page	End Page
RAI 442 07.03-33	2	2
RAI 442 07.03-34	3	4
RAI 442 07.09-61	5	8

AREVA NP's schedule for providing a technically correct and complete response to all questions in RAI 442 remains unchanged and is provided below.

Question #	Response Date
RAI 442 — 7.1-26	April 21, 2011
RAI 442 — 7.1-27	April 14, 2011
RAI 442 — 7.1-28	April 7, 2011
RAI 442 — 7.1-30	April 28, 2011
RAI 442 — 7.1-31	April 7, 2011
RAI 442 — 7.1-32	April 7, 2011
RAI 442 — 7.3-32	April 14, 2011
RAI 442 — 7.9-64	April 28, 2011
RAI 442 — 7.9-67	April 7, 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: Friday, February 25, 2011 8:07 AM
To: Tesfaye, Getachew
Cc: BRYAN Martin (External RS/NB); 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. 442, FSAR Ch. 7, Supplement 4

Getachew,

On November 19, 2010, AREVA NP Inc. (AREVA NP) provided a schedule for a technically correct and complete response to the questions in RAI 442. Supplement 1 response was sent on January 7, 2011 to provide a revised schedule for four of the questions. Supplement 2 response was sent on February 9, 2011 to provide a revised schedule. Supplement 3 response was sent on February 18, 2011 to provide technically correct and complete responses to four questions. Based on discussions with NRC, the attached file, "RAI 442 Supplement 4 Response US EPR DC.pdf" provides technically correct and complete responses to one of the 13 questions, as committed.

Appended to this file are affected pages of the U.S. EPR Final Safety Analysis Report and Technical Report ANP-10309P, in redline-strikeout format which support the response to RAI 442 Question 07.09-63.

The following table indicates the respective pages in the response document, "RAI 442 Supplement 4 Response US EPR DC.pdf," that contain AREVA NP's response to the subject question.

Question #	Start Page	End Page
RAI 442 07.09-63	2	2

Based upon the information presented to the NRC during the February 15, 2011, Public Meeting, the schedule for the remaining questions has been changed.

AREVA NP's schedule for providing a technically correct and complete response to all questions in RAI 442 is provided below.

Question #	Response Date
RAI 442 — 7.1-26	April 21, 2011
RAI 442 — 7.1-27	April 14, 2011
RAI 442 — 7.1-28	April 7, 2011
RAI 442 — 7.1-30	April 28, 2011
RAI 442 — 7.1-31	April 7, 2011
RAI 442 — 7.1-32	April 7, 2011
RAI 442 — 7.3-32	April 14, 2011

RAI 442 — 7.3-33	April 7, 2011
RAI 442 — 7.3-34	April 7, 2011
RAI 442 — 7.9-61	April 7, 2011
RAI 442 — 7.9-64	April 28, 2011
RAI 442 — 7.9-67	April 7, 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: Friday, February 18, 2011 12:21 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. 442, FSAR Ch. 7, Supplement 3

Getachew,

On November 19, 2010, AREVA NP Inc. (AREVA NP) provided a schedule for a technically correct and complete response to the questions in RAI 442. Supplement 1 response was sent on January 7, 2011 to provide a revised schedule for four of the questions. Supplement 2 response was sent on February 9, 2011 to provide a revised schedule. Based on discussions with NRC, the attached file, "RAI 442 Supplement 3 Response US EPR DC.pdf" provides technically correct and complete responses to four of the 17 questions, as committed.

Appended to this file are affected pages of the U.S. EPR Final Safety Analysis Report and Technical Report ANP-10281P, in redline-strikeout format which support the response to RAI 442 Question 07.01-29.

The following table indicates the respective pages in the response document, "RAI 442 Supplement 3 Response US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 442 07.01-29	2	2
RAI 413 07.09-62	3	4
RAI 413 07.09-65	5	5
RAI 413 07.09-66	6	6

The schedule for technically correct and complete responses to the remaining 13 questions is unchanged and provided below:

AREVA NP's schedule for providing a technically correct and complete response to all questions in RAI 442 is provided below.

Question #	Response Date
RAI 442 — 7.1-26	March 15, 2011
RAI 442 — 7.1-27	March 15, 2011
RAI 442 — 7.1-28	March 15, 2011
RAI 442 — 7.1-30	March 15, 2011
RAI 442 — 7.1-31	March 15, 2011
RAI 442 — 7.1-32	March 15, 2011
RAI 442 — 7.3-32	March 15, 2011
RAI 442 — 7.3-33	March 15, 2011
RAI 442 — 7.3-34	March 15, 2011
RAI 442 — 7.9-61	March 15, 2011
RAI 442 — 7.9-63	March 15, 2011
RAI 442 — 7.9-64	March 15, 2011
RAI 442 — 7.9-67	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: BRYAN Martin (External RS/NB)
Sent: Wednesday, February 09, 2011 5:07 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. 442, FSAR Ch. 7, Supplement 2

Getachew,

On November 19, 2010, AREVA NP Inc. (AREVA NP) provided a schedule for a technically correct and complete response to the questions in RAI 442. Supplement 1 response was sent on January 7, 2011 to provide a revised schedule for four of the questions. To allow additional time to interact with the staff and to process the responses a revised schedule is provided below. It should be noted that the dates below may need to be adjusted following the February 15, 2011 public meeting between AREVA and the NRC on digital instrumentation and controls.

AREVA NP's schedule for providing a technically correct and complete response to all questions in RAI 442 is provided below.

Question #	Response Date
RAI 442 — 7.1-26	March 15, 2011

RAI 442 — 7.1-27	March 15, 2011
RAI 442 — 7.1-28	March 15, 2011
RAI 442 — 7.1-29	March 15, 2011
RAI 442 — 7.1-30	March 15, 2011
RAI 442 — 7.1-31	March 15, 2011
RAI 442 — 7.1-32	March 15, 2011
RAI 442 — 7.3-32	March 15, 2011
RAI 442 — 7.3-33	March 15, 2011
RAI 442 — 7.3-34	March 15, 2011
RAI 442 — 7.9-61	March 15, 2011
RAI 442 — 7.9-62	March 15, 2011
RAI 442 — 7.9-63	March 15, 2011
RAI 442 — 7.9-64	March 15, 2011
RAI 442 — 7.9-65	March 15, 2011
RAI 442 — 7.9-66	March 15, 2011
RAI 442 — 7.9-67	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: BRYAN Martin (External RS/NB)
Sent: Friday, January 07, 2011 11:15 AM
To: Tesfaye, Getachew
Cc: DELANO Karen (RS/NB); ROMINE Judy (RS/NB); BENNETT Kathy (RS/NB); RYAN Tom (RS/NB); PANNELL George (CORP/QP)
Subject: Response to U.S. EPR Design Certification Application RAI No. 442, FSAR Ch. 7, Supplement 1

Getachew,

On November 19, 2010, AREVA NP Inc. (AREVA NP) provided a schedule for a technically correct and complete response to the questions in RAI 442. To allow additional time to interact with the staff a revised schedule is provided below for questions 7.1.29, 7.1.32, 7.9-65 and 7.9-67. The schedule for the other questions remains unchanged.

AREVA NP's schedule for providing a technically correct and complete response to all questions in RAI 442 is provided below.

Question #	Response Date
RAI 442 — 7.1-26	March 15, 2011
RAI 442 — 7.1-27	March 15, 2011
RAI 442 — 7.1-28	March 15, 2011
RAI 442 — 7.1-29	**February 9, 2011**
RAI 442 — 7.1-30	February 9, 2011
RAI 442 — 7.1-31	March 15, 2011

RAI 442 — 7.1-32	**February 9, 2011**
RAI 442 — 7.3-32	February 9, 2011
RAI 442 — 7.3-33	February 9, 2011
RAI 442 — 7.3-34	March 15, 2011
RAI 442 — 7.9-61	February 9, 2011
RAI 442 — 7.9-62	February 9, 2011
RAI 442 — 7.9-63	February 9, 2011
RAI 442 — 7.9-64	March 15, 2011
RAI 442 — 7.9-65	**March 15, 2011**
RAI 442 — 7.9-66	February 9, 2011
RAI 442 — 7.9-67	**February 9, 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: Friday, November 19, 2010 5:12 PM
To: 'Tesfaye, Getachew'
Cc: DELANO Karen (RS/NB); ROMINE Judy (RS/NB); BENNETT Kathy (RS/NB); PANNELL George (CORP/QP)
Subject: Response to U.S. EPR Design Certification Application RAI No. 442, FSAR Ch. 7

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 442 Response US EPR DC.pdf" provides a schedule since a technically correct and complete response to the 17 question (s) is not provided.

The following table indicates the respective pages in the response document, "RAI 442 Response US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 442 — 7.1-26	2	2
RAI 442 — 7.1-27	3	3
RAI 442 — 7.1-28	4	4
RAI 442 — 7.1-29	5	5
RAI 442 — 7.1-30	6	6
RAI 442 — 7.1-31	7	8
RAI 442 — 7.1-32	9	9
RAI 442 — 7.3-32	10	10
RAI 442 — 7.3-33	11	11
RAI 442 — 7.3-34	12	12
RAI 442 — 7.9-61	13	13
RAI 442 — 7.9-62	14	14
RAI 442 — 7.9-63	15	15

RAI 442 — 7.9-64	16	16
RAI 442 — 7.9-65	17	17
RAI 442 — 7.9-66	18	18
RAI 442 — 7.9-67	19	19

A complete answer is not provided for the 17 questions. The schedule for a technically correct and complete response to these questions is provided below.

Question #	Response Date
RAI 442 — 7.1-26	March 15, 2011
RAI 442 — 7.1-27	March 15, 2011
RAI 442 — 7.1-28	March 15, 2011
RAI 442 — 7.1-29	January 7, 2011
RAI 442 — 7.1-30	February 9, 2011
RAI 442 — 7.1-31	March 15, 2011
RAI 442 — 7.1-32	January 7, 2011
RAI 442 — 7.3-32	February 9, 2011
RAI 442 — 7.3-33	February 9, 2011
RAI 442 — 7.3-34	March 15, 2011
RAI 442 — 7.9-61	February 9, 2011
RAI 442 — 7.9-62	February 9, 2011
RAI 442 — 7.9-63	February 9, 2011
RAI 442 — 7.9-64	March 15, 2011
RAI 442 — 7.9-65	January 7, 2011
RAI 442 — 7.9-66	February 9, 2011
RAI 442 — 7.9-67	January 7, 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: Wednesday, October 20, 2010 8:09 AM
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Cc: Zhao, Jack; Morton, Wendell; Mott, Kenneth; Spaulding, Deirdre; Truong, Tung; Zhang, Deanna; Jackson, Terry; Canova, Michael; Colaccino, Joseph; ArevaEPRDCPEm Resource
Subject: U.S. EPR Design Certification Application RAI No. 442(4295,5076,5068,5067), FSAR Ch. 7

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on September 10, 2010, and discussed with your staff on October 13, 2010. Drat RAI Questions 07.01-26 and 07.03-33 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,

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Response to

Request for Additional Information No. 442, Supplement 11

10/20/2010

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 07.01 - Instrumentation and Controls - Introduction
SRP Section: 07.03 - Engineered Safety Features Systems
SRP Section: 07.09 - Data Communication Systems

Application Section: FSAR Ch 7

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

Question 07.01-28:

Clarify the actuation path for system and component level control of safety systems from the Process Information and Control System (PICS).

10 CFR 52.47(a)(2) requires, in part, a description of structures, systems, and components sufficient to permit understanding of the system designs. During the review of Technical Report ANP-10309, "U.S. EPR Digital Protection System", Revision 0, the staff observed that in Section 8.5, the applicant states that, " While the US EPR design includes the ability to manually manipulate these actuators at the individual component level from the non-safety-related PICS (the component level manipulations are not processed through the PS), the system level actuations addressed in this section are implemented through Class 1E actuation paths and are single failure tolerant." The applicant did not state where in the ESF logic chain that the system level manual actuation originates from the PICS. In a teleconference the staff had with the applicant, the applicant states that the manual actuation signal goes through the PACS. During this teleconference, the applicant committed to clarifying this Section 8.5 of Technical Report ANP-10309, U.S. EPR Digital Protection System, Revision 0, in order to make it clearer that the system level manual actuation from the PICS goes through the PACS.

Response to Question 07.01-28:

System level manual initiation of engineered safety feature actuation system (ESFAS) functions is not performed from the process information and control system (PICS).

U.S. EPR FSAR Tier 2, Section 7.3.1.1 states:

> “The capability for manual system-level ESF actuations is available to the operator through the safety information and control system (SICS) in the main control room (MCR). These manual actuations are acquired by the actuation logic units (ALUs) in the protection system and combined with the automatic actuation logic.”

U.S. EPR FSAR Tier 2, Section 7.3.1.1 also states:

> “The capability for component-level control of ESF system actuators is available to the operator on both the PICS and the SICS. Commands from the PICS are processed by the process automation system (PAS) and sent to the priority and actuator control system (PACS) for prioritization.”

The discussion regarding manual controls in Technical Report ANP-10309P, “U.S. EPR Digital Protection System,” will be revised for clarification. Technical Report ANP-10309P revisions described in this response will be submitted by separate letter.

Proposed changes to the instrumentation and controls (I&C) architecture were communicated to the NRC staff in the February 15, 2011 public meeting. U.S. EPR FSAR Tier 2, Section 7.5 and Section 7.7 were revised to incorporate the revised I&C architecture. These sections are provided in their entirety with this response to facilitate NRC review.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 7.5 and Section 7.7 will be revised as described in the response and indicated on the enclosed markup.

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U.S. EPR Final Safety Analysis Report Markups

7.5 Information Systems Important to Safety

The information necessary to monitor the nuclear steam supply systems (NSSS), the containment systems, and the balance of plant is displayed on the operator console and the various screens and panels located within the main control room (MCR). Information systems important to safety are those systems that provide information to control and operate the unit safely through all operating conditions, including anticipated operational occurrences (AOO), accident and post-accident conditions. This section is limited to the description of those display instruments that provide information to enable the operator to assess reactor status, the onset and severity of accident conditions, and engineered safety feature (ESF) actuation status and performance, or to enable the operator to reliably perform vital manual actions such as safe shutdown and initiation of manual ESF actuation.

This section also provides information on the classification of monitored variables, which is based on the guidance provided by RG 1.97, Revision 4, which endorses IEEE Std 497-2002 (Reference 1), with certain clarifying regulatory positions. A methodology for selecting the post-accident monitoring (PAM) variables based on IEEE Std 497-2002 (Reference 1) is presented in Section 7.5.2.2.1.

7.5.1 Description

This section discusses the instrumentation and controls (I&C) used to provide information important to safety and to provide a means for manual operator action related to accident mitigation.

7.5.1.1 Annunciator Systems

The annunciator system consists of alarms and functions to enable operators to silence, acknowledge, reset, and test alarms. The non-safety-related process information and control system (PICS) is the primary annunciator system. In the event of an abnormal plant or system condition, the operator will receive an indication of the abnormal event. Icons will be displayed on the PICS screens to allow the operator to acknowledge the alarm and to view system diagrams of the affected system.

The safety information and control system (SICS) provides ~~some~~ limited ~~backup~~ annunciation functions to support accident mitigation~~if the PICS is unavailable~~.

The architecture and functions of the PICS and SICS are described in Section 7.1.

7.5.1.2 Post-Accident Monitoring Instrumentation

The ~~accident monitoring~~ PAM instrumentation ~~(AMI)~~ provides plant process variable information and system status~~, known as post-accident monitoring (PAM) variables,~~ to the operator in the MCR to permit the operator to perform the following:

- Preplanned manual safety functions.
- Capability to assess plant conditions, safety system performance, and determine appropriate actions to take to respond to abnormal events.
- Capability to bring the plant to a safe shutdown condition.

The ~~AMI~~PAM instrumentation utilizes the components of existing safety-related and non-safety-related I&C systems to accomplish ~~AMI~~PAM functions. ~~The primary operator interface in the MCR for displaying all~~The PAM variables are displayed on~~is~~ the non-safety-related PICS. ~~If the PICS is not available, the safety-related SICS is used.~~ Additionally, Type A, B, and C PAM variables are displayed on the safety-related SICS. The ~~protection system (PS),~~ priority and actuator control system (PACS),~~safety automation system (SAS), process automation system (PAS),~~ SICS, and PICS contain the hardware to obtain and display the safety-related and non-safety-related PAM variables. The selection of PAM variables is described in Section 7.5.2.2.1.

7.5.1.3 Emergency Response ~~Facilities~~Information

The description of the emergency response ~~facilities (ERF)~~ information capability in this section is limited to the system interface with the plant I&C systems. The ~~ERF consists of the~~ safety parameter display system (SPDS), Emergency Response Data System (ERDS), and technical support center (TSC)~~. These systems and facilities~~ are designed and implemented in accordance with NUREG-0696 (Reference 4), NUREG-0654 (Reference 5) and NUREG-0737 (Reference 6).

The PICS provides a means of transmitting data ~~via~~through a firewall to systems external to the plant I&C systems. Details of the architecture of PICS are provided in Section 7.1.

The TSC contains PICS workstations that display pertinent information for plant management and technical support personnel. These workstations do not send control signals to the PICS. The PICS provides the ~~primary~~ SPDS display ~~and the SICS provides a backup SPDS display~~.

7.5.1.4 Bypass and Inoperable Status Indication

Bypassed and inoperable status indication (BISI) of safety-related systems is provided by the PICS. BISI is also discussed in Section 7.5.2.1.1, Section 7.5.2.2.4, and Section 7.5.2.2.5.

7.5.2 Analysis

The human factors engineering (HFE) program described in Chapter 18 provides a design process that reasonably assures that plant operators can access the required

information and controls to enable safe and efficient control and monitoring of plant processes and equipment. As part of the HFE program, verification and validation evaluations will confirm that the human system~~s~~ interfaces provide the operator with sufficient information to perform required manual safety functions and sufficient time to make reasoned judgments and take action where operator action is essential for maintaining the plant in a safe condition.

7.5.2.1 Acceptance Criteria

The following acceptance criteria ~~requirements~~guidance listed in NUREG-0800 (Reference 10), Section 7.5, apply to the I&C systems listed in Section 7.5.1.

~~Compliance to~~Compliance with the following requirements is discussed in Section 7.1:

- 10 CFR 50.55a(a)(1), "Quality Standards".
- 10 CFR 50.55a(h), "Protection and Safety Systems".
- GDC 1, "Quality Standards and Records".
- GDC 2, "Design Basis for Protection against Natural Phenomena".
- GDC 4, "Environmental and Missile Design Basis".
- GDC 19, "Control Room".
- GDC 24, "Separation of Protection and Control Systems".
- ~~10 CFR 52.47 (b)(1), "ITAAC for Standard Design Certification".~~

7.5.2.1.1 10 CFR 50.34(f), "Additional TMI-Related Requirements"

The following ~~are~~ TMI-related requirements ~~that~~ apply ~~to the AMI described in Section 7.5.1~~.

10 CFR 50.34(f)(2)(v) Bypassed and Inoperable Status Indication

If any ~~PAM~~ Type A, B, and C PAM variable is bypassed or rendered inoperable, an indication is provided to the operator in the MCR. Description of the bypassed and inoperable status of safety systems is provided in Section 7.5.2.2.4.

10 CFR 50.34(f)(2)(xi) Direct Indication of Relief and Safety Valve Indication

Three pressurizer safety relief valves (PSRV) are arranged at the top of the pressurizer (PZR) for overpressure protection of the reactor coolant system (RCS). Each PSRV is provided with a position sensor. The position (open or closed) for each valve is indicated in the MCR. The PSRVs are described in Section 5.4.13.

10 CFR 50.34(f)(2)(xii) Auxiliary Feedwater Flow Indication

Indication of emergency feedwater (EFW) flow to each steam generator (SG) is provided in the MCR. ~~Details on the~~ EFW flow sensors are shown in Figure 10.4.9-1~~described in Section 10.4.9~~.

10 CFR 50.34(f)(2)(xvii) Accident Monitoring Instrumentation

The following instrumentation is available for readout in the MCR:

- Containment pressure sensors are provided by the containment ventilation system described in Section 9.4.7.
- Level sensors for the in-containment refueling water storage tank (IRWST) are provided by the safety injection system described in Section 6.3.
- Containment hydrogen sensors are provided by the hydrogen monitoring system described in Section 6.2.5.
- Containment radiation intensity (high level) monitors are provided by the containment high range monitors described in Section 12.3.4.1.3 and Table 12.3-3.~~radiation monitoring system (RMS) described in Section 11.5.~~
- Noble gas effluent monitoring at all potential accident release points are provided by the RMS described in Section 11.5 and Table 11.5-1.
- Continuous sampling of radioiodines and particulates from all potential accident release points will be provided by the process sampling system as described in Section 11.5 and Table 11.5-1. Additional details on the process sampling system are described in Section 9.3.2.

10 CFR 50.34(f)(2)(xviii) Inadequate Core Cooling Instrumentation

The following instrumentation provides an indication in the MCR of inadequate core cooling:

- A combination of RCS hot leg wide range (WR) pressure and the core outlet thermocouples (COT) described in Section 7.1 is used to determine inadequate core cooling. In addition, the reactor vessel water level indication is provided by the reactor pressure vessel water level measurement system described in Section 7.1.

10 CFR 50.34(f)(2)(xix) Instruments for Monitoring Plant Conditions Following Core Damage

The ~~post-accident monitoring~~PAM variables discussed in Section 7.5.2.2.1 and the severe accident monitoring variables discussed in Section 7.5.2.2.3 provide for monitoring plant conditions following core damage.

10 CFR 50.34(f)(2)(xx) Power for Pressurizer Level Indication

Each of the four PZR level sensors generates a signal that is received in one of the four divisions of the PS. The PZR level sensors are powered from the Class 1E bus of the PS division in which the sensor signal is received. PZR level indication is provided by both the PICS and ~~is backed by~~ the safety-related SICS.

Each division of the PS and the SICS is supplied by an independent Class 1E, uninterruptible electrical bus. These busses are backed by the emergency diesel generators to cope with loss of offsite power. Inside a division, the PS cabinets are supplied by two redundant, uninterruptible 24 Vdc feeds. To cope with loss of onsite and offsite power, the feeds to the PS cabinets are supplied with two-hour batteries.

7.5.2.1.2 GDC 13, "Instrumentation and Control"

The PICS and SICS provide the capability for monitoring ~~variables, including post-accident monitoring~~PAM variables and system variables over their anticipated ranges for normal operation, for AOOs, and for postulated accident conditions as appropriate. This monitoring provides reasonable assurance of safety by including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, or the containment and its associated systems. The PICS and SICS also provide a means of manual control capabilities for maintaining these variables and systems within prescribed operating ranges.

7.5.2.2 Discussion

7.5.2.2.1 Conformance ~~with~~to Regulatory Guide 1.97 and BTP 7-10

The guidance of RG 1.97, Revision 4, will be used to confirm the ~~accident monitoring~~PAM variables during detailed design. With clarifying regulatory positions, RG 1.97, Revision 4, endorses IEEE Std 497-2002 (Reference 1), which provides performance-based criteria for selecting variables and recommends determining the variable type according to its accident management function. The accident management function is to be identified by its use in the Emergency Procedure Guidelines (EPG), Emergency Operating Procedures (EOP), and Abnormal Operating Procedures (AOP). The development of these guidelines and procedures is discussed in Section 13.5. When these procedures are complete and verified, they will be used to confirm the ~~accident monitoring instrumentation~~PAM variables list.

Development of the ~~Accident Monitoring~~PAM Variables List

Section 13.5.2.1.2 describes the EOP development process. Preparation of EOPs and AOPs for the U.S. EPR plant requires detailed design of systems to be completed. An evaluation was performed to develop the ~~initial~~ list of ~~accident monitoring~~PAM variables for the U.S. EPR plant. The evaluation included:

- A ~~review~~step-by-step evaluation of Volume 1 of the Babcock and Wilcox (B&W) Owners Group Emergency Operating Procedures ~~B&W Unit EOP~~ Technical Basis Document (Reference 11) was performed to identify required supporting instrumentation. The evaluation considered the differences in the U.S. EPR and the B&W plant designs.

- A review of the operator manual actions listed in ~~Section 15.0.0.3.7~~Chapter 15 for which no automatic control is provided to determine instrumentation required to support those actions.

- A review of radiation monitoring system~~s~~ design to identify instruments necessary to support post-accident monitoring.

- Identification of additional instrumentation based on engineering judgment considering differences between the U.S. EPR and the B&W plant designs.

- A gap evaluation was performed to confirm that critical safety functions and fission product barriers described in IEEE Std 497-2002 were adequately monitored by the list of instruments developed~~comparison of the list with the critical safety functions and fission product barriers described in Reference 1~~.

The list of ~~accident monitoring~~PAM variables is provided in Table 7.5-1—~~Initial~~ Inventory of Post-Accident Monitoring Variables.

Confirmation of the ~~Accident Monitoring~~PAM Variables

To meet the guidance of RG 1.97, Revision 4 and Reference 1, a systematic step-by-step review of the plant-specific EOPs for the U.S. EPR is required. See Section 13.5 for more information on U.S. EPR procedure development.

A COL applicant that references the U.S. EPR design certification will ~~update~~ confirm the ~~initial~~ inventory list of ~~accident monitoring~~PAM variables in Table 7.5-1 ~~with a final list~~ upon completion of the emergency operating and abnormal operating procedures prior to fuel loading.

~~To meet the guidance of RG 1.97 and Reference 1, a systematic step-by-step review of the plant-specific EOPs for the U.S. EPR is required. See Section 18.8 and Section 13.5 for more information on U.S. EPR procedure development.~~

The ~~complete accident monitoring instrumentation~~confirmation of the PAM variables list by the COL Applicant will be documented in a table format that includes the following:

- Variable name that indicates the variable function.

- Variable Type (A, B, C, D or E).

- Range.

- Safety classification (1E or non-1E).
- Environmental and Seismic Qualification.
- Minimum number of instruments required.
- Monitoring duration for the variable.

~~**Performance Assessment**~~

~~A performance assessment of each of the following performance criteria specified in Reference 1 will be conducted and the results documented:~~

- ~~Range.~~
- ~~Accuracy.~~
- ~~Response time.~~
- ~~Required instrument duration.~~
- ~~Reliability.~~

~~In conformance with Reference 1, a graded approach will be used to develop setpoints associated with accident monitoring variables based on their importance to safety.~~

Criteria for Selection of Variable Types

In accordance with RG 1.97, Revision 4, and ~~Reference 1~~IEEE Std 497-2002, the ~~accident monitoring~~PAM variables are selected and the variable types are determined according to its accident management function. These variables are the primary source of accident monitoring information. Five types of variables exist and the selection criteria are described as follows:

Type A Variables

Type A variables are those variables that provide the primary information required to permit the control room operating staff to:

- Take specific pre-planned manually-controlled actions for which no automatic control is provided and that are required for safety systems to perform their safety-related functions as assumed in the plant Accident Analysis Licensing Basis.
- Take specific planned manually-controlled actions for which no automatic control is provided and that are required to mitigate the consequences of an AOO.

As recommended by RG 1.97, Revision 4, ~~t~~Type A variables include those variables that are associated with contingency actions that are within the plant licensing basis and may be identified in written procedures.

Type B Variables

Type B variables are those variables that provide primary information to the control room operators to assess the accomplishing or maintaining of plant critical safety functions.

Type C Variables

Type C variables are those variables that provide primary information to the control room operators of the potential for breach, or the actual breach, of the three fission product barriers (extended range): fuel cladding, reactor coolant system pressure boundary, and containment pressure boundary.

The selection of these variables represents a minimum set of plant variables that provide the most direct indication of the integrity of the three fission product barriers. They also provide the capability for monitoring beyond the normal operating range.

Type D Variables

Type D variables are those variables that are required in procedures and licensing basis documentation to:

- Indicate the performance of those safety systems and auxiliary supporting features necessary for the mitigation of design basis events (DBE).
- Indicate the performance of other systems necessary to achieve and maintain a safe shutdown condition.
- Verify safety system status.

Type D variables are based upon the plant accident analysis licensing basis and those necessary to implement the following procedures, which are applicable to the plant design:

- Event-specific EPGs or plant-specific EOPs.
- Functional restoration EPGs or plant-specific EOPs.
- Plant AOPs.

Type E Variables

Type E variables are those variables required for use in determining the magnitude of the release of radioactive materials and continually assessing such releases.

These variables are selected to:

- Monitor the magnitude of releases of radioactive materials through identified pathways.
- Monitor the environmental conditions used to determine the impact of releases of radioactive materials through identified pathways.
- Monitor radiation levels and radioactivity in the plant environs.
- Monitor radiation levels and radioactivity in the control room and selected plant areas where access may be required for plant recovery.

Post-Accident Monitoring Instrumentation Criteria

The ~~I&C systems that perform the AMI functions~~PAM instrumentation are designed in accordance with the performance (criterion 5), design (criterion 6), qualification (criterion 7), and display criteria (criterion 8) of ~~Reference 1~~IEEE Std 497-2002 with the modifications specified in RG 1.97, Revision 4, and in accordance with the supplemental guidance provided in BTP 7-10 (Reference 7).

7.5.2.2.2 Use of Digital Systems

~~The supervisory control system level (level 2) consists of data processing related to HMI for process control and monitoring.~~The human machine interface (HMI) systems provide the ability to control and monitor the plant operation. The HMI systems consist of the SICS and PICS. ~~are the systems that make up the supervisory control system level and are~~ The PICS is implemented using a digital computer platform~~s~~. The SICS is implemented using hardwired I&C and the qualified display system (QDS).

The QDS is an HMI that is qualified to non-safety-related supplemented grade (NS-AQ) to meet Seismic Class II criteria.~~The software and hardware development of the SICS is performed in accordance with the quality criterion for digital computers specified in IEEE Std 7-4.3.2-2003 (Reference 2).~~

To minimize the potential for non-safety-related digital control system failures that could challenge safety systems, non-safety-related digital control system hardware and software is developed using a structured process similar to that applied to safety-related system software; however, the process is tailored to account for the lower safety significance. The hardware and software development process for PICS is described in Section 7.1~~Section 7.7~~.

7.5.2.2.3 Monitoring for Severe Accidents

Instrumentation used to monitor severe accident conditions are identified in Table 19.2-3. The severe accident response instrumentation is designed so there is reasonable assurance that the instrumentation will operate in the severe accident environment for which they are intended and over the time span for which they are needed.

7.5.2.2.4 Conformance to Regulatory Guide 1.47

If a protective function of some part of a safety-related system has been bypassed or deliberately rendered in-operative, continued indication of the bypassed condition ~~for each affected safety group~~ is provided in the MCR.

The PS and the SAS are the safety-related system level automation systems. Both systems provide display signals to the PICS. Outputs to PICS from safety-related systems are supplied through qualified isolation devices. If the PS or SAS is operated in a bypassed mode or inoperable condition, an output is automatically provided to the PICS for indication of the bypass or inoperable condition in accordance with the guidance of RG 1.47, and Clause 5.8.3 of IEEE Std 603-1998 (Reference 3).

7.5.2.2.5 Scope of Bypassed and Inoperable Status Indications

The BISI in the MCR includes bypasses of the reactor trip (RT) functions described in Section 7.2 and ESF functions described in Section 7.3. In addition, BISI is provided for the safety injection system (SIS) accumulator isolation valves and the residual heat removal (RHR) system suction isolation valves. If any SIS accumulator isolation valve comes off its open seat during conditions that require the valve to be open, a bypass indication will be provided in the MCR. If any RHR system suction isolation valve comes off its closed seat during conditions that require the valves to be closed, a bypass indication will be provided in the MCR.

7.5.2.2.6 Redundancy and Diversity of Display

Type A, B, and C PAM variables are sent directly from the SCDS to the SICS via hardwired connections bypassing software based components. Diverse display of variables is not required. The same variables are processed through the PAS and PICS to provide a redundant path.~~Diversity is provided for the processing and display of indications and alarms necessary to alert the operator to abnormal plant conditions, including Type A, B, and C accident monitoring variables as defined in RG 1.97, Revision 4. These accident monitoring variables are acquired and processed by the safety-classified PS and SAS, then transmitted to the safety-classified SICS for display.~~

~~Also, the non-safety-classified PAS provides diverse processing of sensor information because the PAS obtains sensor information independently of the PS and SAS software. Type A, B, and C accident monitoring variables are acquired by the PAS from the PS or SAS via isolation devices, processed by PAS, and then transmitted to the non-safety classified PICS for display. Therefore, the PAS and the PICS provide a redundant and diverse path for display of the variables.~~

7.5.2.2.7 Independence and Compliance with IEEE Std 603-1998

Section 7.1 describes the overall I&C system architecture and how independence is achieved between safety-related and non-safety-related I&C systems. ~~Compliance to~~Compliance with Clause 5.6.3, "Independence Between Safety Systems and Other Systems," and Clause 6.3, "Interaction Between the Sense and Command Features and Other Systems," are addressed in Section 7.1. ~~Requirements of Reference 3 for safety systems meet or exceed the requirements of the endorsed version, IEEE Std 603-1991. A comparison of IEEE Std 603-1991 and IEEE 603-1998 is described in Digital Protection System Technical Report ANP-10309P (Reference 9).~~

7.5.2.2.8 ~~Self-Test Provisions~~

~~Branch Technical Position BTP 7-17 (Reference 8), provides guidance on the use of self-testing features of digital I&C system. The SICS is designed to maximize the use of automated self-testing features to minimize the burden on plant personnel. The safety-related qualified display system (QDS) within the SICS has the capability to perform automatic self-testing to verify its ability to perform the intended functions. This self-testing feature includes, but is not limited to, the availability of components such as processors, communication and link modules, power supplies, and input-output modules. The positive aspect of these self-testing features is to provide a mechanism for detecting all detectable failures. Furthermore, the positive aspects of the self-test features are not compromised by the additional complexity added to the system design. The self-test features of the SICS do not affect the ability of the system to perform its safety functions.~~

7.5.3 References

1. IEEE Std 497-2002, "Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, 2002.

2. Deleted.~~IEEE 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, 2003.~~

3. IEEE Std 603-1998, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, 1998.

4. NUREG-0696, "Functional Criteria for Emergency Response Facility," Nuclear Regulatory Commission, 1981.

5. NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Nuclear Regulatory Commission, 1980.

6. NUREG-0737, "Clarification of TMI Action Plan Requirements," Nuclear Regulatory Commission, 1980.

7. NUREG-0800, BTP 7-10, "Guidance on Application of Regulatory Guide 1.97," Nuclear Regulatory Commission, March 2007.

8. Deleted.~~NUREG-0800, BTP 7-17, "Guidance on Self Test and Surveillance Provisions," Nuclear Regulatory Commission, March 2007.~~

9. Deleted.~~ANP-10309P, Revision 0, "U.S. EPR Digital Protection System Technical Report," AREVA NP Inc., November 2009.~~

10. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Nuclear Regulatory Commission, March 2007.

11. Letter OG-1789, Tony Stallard, Chairman, B&WOG Operator Support Committee, to Chief, Reactor Systems Branch (NRC), "Transmittal of B&W Owners Group Emergency Operating Procedures Technical Bases Document, Revision 9," dated April 26, 2000 and attachments (ML003711891).

Table 7.5-1—~~Initial~~ Inventory of Post-Accident Monitoring Variables (Sheet 1 of 4)

No	Variable	Range	Minimum Channels Required	Duration	Safety Class	EQ[1] per IEEE Std 323-1974	Seismic Qualification	Type				
								A	B	C	D	E
1	Annulus Ventilation System Gamma Activity	See Table 11.5-1	2	1 year	NS-AQ	Yes	I			X		X
2	Condenser Evacuation System Exhaust Radiation	See Table 11.5-1	1	24 hours	NS	No	NSC					X
3	Containment High Range Radiation	See Table 12.3-3	2	1 year	S	Yes	I		X	X		X
4	Containment Isolation Valve Position Indications	Closed/not closed	1 / valve	1 year	S	Yes	I		X			
5	Containment Service Compartment Wide Range Pressure	-5 to 220 psig	2	1 year	S	Yes	I			X	X	
6	Core Outlet Thermocouples Wide Range Temperature	32 to 2300 °F	8 (2 / quadrant)	1 year	S	Yes	I		X	X	X	
7	Extra Borating System Flow	0 to 60 gpm	1 / train	24 hours	S	Yes	I				X	
8	Emergency Feedwater Flow to SG	0 to 545 gpm	2 / train	24 hours	S	Yes	I	X	X		X	
9	Emergency Feedwater Wide Range Pool Level	0 to 300 inches	1 / train	24 hours	S	Yes	I				X	
10	Emergency Power Supply System Voltage	0 to 8625 VAC	1 /train	1 year	S	Yes	I				X	
11	Fuel Building Fuel Pool Dose Rate	See Table 12.3-3	1	24 hours	NS	No	NSC					X

Table 7.5-1—~~Initial~~ Inventory of Post-Accident Monitoring Variables (Sheet 2 of 4)

No	Variable	Range	Minimum Channels Required	Duration	Safety Class	EQ[1] per IEEE Std 323-1974	Seismic Qualification	Type				
								A	B	C	D	E
12	Fuel Building Setdown Dose Rate	See Table 12.3-3	1	24 hours	NS	No	NSC					X
13	Hot Leg Injection Flow	0 to 3200 gpm	1 / train	1 year	S	Yes	I				X	
14	Intermediate Range Nuclear Instrumentation	5×10^{-6} to 60% NP	2	2 hours	S	Yes	I		X		X	
15	In-containment Refueling Water Storage Tank Level	0 to 20 feet	1	1 year	S	Yes	I				X	
16	Low Head Safety Injection Wide Range Flow	0 to 3800 gpm	2 / train	1 year	S	Yes	I		X		X	
17	Main Control Room Dose Rate	See Table 12.3-3	1	24 hours	NS	No	NSC					X
18	Main Steam Line Radiation	See Table 11.5-1	2 / line	2 hours	S	Yes	I	X	X			
19	Medium Head Safety Injection Wide Range Flow	0 to 1300 gpm	2 / train	24 hours	S	Yes	I		X		X	
20	Pressurizer Level	0 to 100%	2	24 hours	S	Yes	I	X	X			
21	Pressurizer Safety Relief Valve Position Indication	Closed/not closed	1 / valve	24 hours	NS-AQ	Yes	I				X	
22	Reactor Coolant System Hot Leg Pressure	0 to 3000 psig	2	1 year	S	Yes	I	X	X	X		
23	Reactor Coolant System Wide Range Cold Leg Temperature	32 to 662 °F	2	24 hours	S	Yes	I	X	X		X	

Table 7.5-1—~~Initial~~ Inventory of Post-Accident Monitoring Variables (Sheet 3 of 4)

No	Variable	Range	Minimum Channels Required	Duration	Safety Class	EQ[1] per IEEE Std 323-1974	Seismic Qualification	Type				
								A	B	C	D	E
24	Reactor Coolant System Wide Range Hot Leg Temperature	32 to 662 °F	2	24 hours	S	Yes	I	X				
25	Reactor Building Personnel Air Lock Dose Rate	See Table 12.3-3	1	24 hours	NS	No	NSC					X
26	Safeguard Building Controlled-Area Ventilation System Gamma Activity	See Table 11.5-1	1	1 year	NS-AQ	Yes	II					X
27	Safeguard Building Corridor Dose Rate	See Table 12.3-3	1 / building	24 hours	NS	No	NSC					X
28	Safeguard Building Personnel Air Lock Dose Rate	See Table 12.3-3	1	24 hours	NS	No	NSC					X
29	Steam Generator Pressure	0 to 1600 psig	2 / SG	24 hours	S	Yes	I	X	X		X	
30	Steam Generator Wide Range Level	0 to 100%	2 / SG	24 hours	S	Yes	I		X		X	
31	Safety Injection Accumulator Isolation Valve Position	Closed/not closed	1 / valve	24 hours	S	Yes	I				X	
32	Safety Injection System Suction Strainer Differential Pressure	0 to 5 psid	1 / train	1 year	S	Yes	I				X	
33	Source Range Neutron Flux	0.05 to 5 x 10^4 n/ cm^2-s	2	1 year	S	Yes	I		X			

Table 7.5-1—~~Initial~~ Inventory of Post-Accident Monitoring Variables (Sheet 4 of 4)

No	Variable	Range	Minimum Channels Required	Duration	Safety Class	EQ[1] per IEEE Std 323-1974	Seismic Qualification	Type				
								A	B	C	D	E
34	Subcooling Margin	611°F Subcooling Margin to 2088°F Superheat	2	24 hours	S	Yes	I	X	X			
35	Vent Stack Aerosol Activity	See Table 11.5-1	1	1 year	NS	No	NSC					X
36	Vent Stack Iodine Activity	See Table 11.5-1	1	1 year	NS	No	NSC					X
37	Vent Stack Gas Activity	See Table 11.5-1	1	1 year	NS	No	NSC					X

Notes:

1. Environmental Qualification.

Next File

7.7 Control Systems Not Required for Safety

The general objectives of the non-safety instrumentation and control (I&C) systems are:

- To make sure the major process variables of the nuclear steam supply system (NSSS) are kept in pre-defined and allowed ranges during normal power operation.

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- To limit the variation of process parameters during normal operation in such a way that the initial conditions for operation are met at the onset of ~~design basis events (DBE)~~ anticipated operational occurrences (AOOs) and postulated accidents (PAs) as assumed in the safety analyses.
- To minimize the need for protective actions and thus increase plant availability.
- To provide the reactor operator with monitoring instrumentation that indicates the required input and output control parameters of the systems and provide the operator with the capability of assuming manual control of the system.

7.7.1 Description

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This section provides a description ~~on the major~~of the non-safety I&C system used to implement non-safety-related functions.

The non-safety functions are categorized as follows:

- ~~Operational~~Process control I&C functions.
- Process ~~L~~limitation I&C functions.

~~Operational~~Process control I&C functions provide control of plant systems during normal operation. These functions are used to make sure the major process variables of the NSSS are kept in predefined and allowed ranges during normal power operation. These functions are described in Section 7.7.2.1 and Section 7.7.2.2.

Process ~~L~~limitation I&C functions are functions that execute one or more of the following actions:

- Prevents plant disturbances from causing normal operating limits to be exceeded.
- Alerts the operator when normal operating limits have been exceeded.
- Prevents disturbances from leading to an ~~DBE~~AOO or PA.

Process ~~L~~limitation I&C functions are described in Section 7.7.2.3.

Some ~~operational~~process control I&C functions have a minor direct influence on the process of nuclear power generation. These functions are presented in Section 7.7.2.4 and are listed in Table 7.7-1—Cross Reference of Non-Safety-Related I&C Controls.

7.7.1.1 I&C Systems Related to Core Control

The I&C systems that provide core-control functions are the reactor control, surveillance and limitation (RCSL) system, the process information and control system (PICS), and the control rod drive control system (CRDCS). The architecture of RCSL, PICS, and CRDCS ~~are~~is described in Section 7.1.

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The RCSL system receives input signals from the signal conditioning and distribution system (SCDS) and implements the automation level I&C functions related to core control. The SCDS provides the instrumentation interface to the RCSL system. PICS interfaces with the RCSL system to provide the operator with control and monitoring capability of the core control functions. The architecture of the SCDS is described in Section 7.1.

The CRDCS cabinet layout is shown in Figure 7.1-26. The CRDCS controls the actuation of the 89 rod cluster control assemblies (RCCA) in the reactor vessel. The RCSL logic transmits the direction of movement (i.e.,withdrawal or insertion), speed of movement, and drop and hold information to the rod control units of the CRDCS. Each rod control unit generates the cycling sequence input to the corresponding CRDCS coil modules in order to control the rod speed and movement for one RCCA. The coil modules control the amount of current applied to the operating coils (i.e. lift coil, movable gripper coil and stationery gripper coil) of the control rod drive mechanism (CRDM) in order to move the corresponding RCCA. A feedback signal is sent from the rod control unit to the RCSL. This feedback signal is used by the RCSL to generate a digital position indication of the RCCA and is based on the number of rod movement steps sent from the CRDCS to the operating coils of the CRDM. Figure 7.7-2 indicates the RCCA maximum speed withdrawal rate is 75 steps/minute. Because each rod movement step equals 0.393 inches/step, the maximum withdrawal rate of the RCCA is 29.48 inches/minute which is less than the maximum allowed of 30 inches/minute. A detailed description of the CRDM and its associated operating coils is provided in Section 3.9.4.

The rod position measurement system (RPMS), described in Section 7.1.1.5.14, uses analog rod position measurement coils located within the CRDM to provide an indication of RCCA position that is separate from the position signal developed by the rod control unit of the CRDCS.

~~The CRDCS controls the movement of the rod cluster control assemblies (RCCA) in the reactor pressure vessel (RPV) by providing sequenced control current to the stationary gripper, moveable gripper, and lift coils of the control rod drive mechanism (CRDM) to move its respective RCCA. The logic that generates the control current comes from the RCSL system. The CRDCS converts the demands from RCSL into rod movement current sequences supplied to the coils of the CRDM.~~

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~~The CRDCS provides the interface to the operating coils of the CRDMs. The CRDCS controls and measures the current to each CRDM coil. There is a module in the CRDCS for each CRDM coil. This controls the amount of current applied, as well as provides the correct sequencing of coil currents for control rod movement in or out of the core. A detailed description of the CRDM and its associated operating coils is provided in Section 3.9.4.~~

~~The rod control unit of the CRDCS provides the interface with the RCSL system for control of the RCCAs. The RCSL system transmits the movement direction and speed of the RCCA to the rod control unit of the CRDCS. Each rod control unit generates the cycling sequence and rod speed for one RCCA, which is used as the input to the coil modules. A feedback signal from the rod control unit to the RCSL system provides information necessary for digital position indication of the RCCA based on the number of rod movement steps sent to the RCCA.~~ Adjustments to boron concentration levels in the reactor coolant system (RCS) provide another means of core control. Boron addition and dilution demand signals are generated by RCSL and are sent to the chemical and volume control system (CVCS). Boron concentration adjustments are addressed in Section 9.3.4.

The ~~operational~~process control I&C functions related to core control are described in Section 7.7.2.1. The process limitation I&C functions related to core control are described in Section 7.7.2.3.1 through Section 7.7.2.3.10.

7.7.1.2 I&C Systems Related to Plant Control

The I&C systems that provide control of plant parameters are the process automation system (PAS) and the PICS. The PAS implements the automation level I&C functions related to the control of the plant parameters. The architecture of PAS is described in Section 7.1. The PICS interfaces with PAS to provide operator control and monitoring capability of plant parameters. The PAS receives input signals directly from sensors and I&C systems, as well as signals from the SCDS.

The ~~operational~~process control I&C functions related to plant parameters are described in Section 7.7.2.2. The process limitation I&C functions related to plant parameters are described in Section 7.7.2.3.11 through Section 7.7.2.3.14.

7.7.2 Design Basis Information

The design basis for the non-safety I&C systems ~~are~~is based on the functions described in this section.

7.7.2.1 Operational Core Control Functions

7.7.2.1.1 Principles of RCCA Control

A description of some of the principles of RCCA control is necessary to aid in the explanation of the core control functions.

Bank Descriptions

The RCCAs are divided into control banks that are used as control elements for the average coolant temperature (ACT) control function, axial offset (AO) control function, and the neutron flux control function. Shutdown banks are only used for negative reactivity insertion during a reactor trip. The 89 RCCAs have the same characteristics.

The bank composition is as follows:

- Control Bank D 9 RCCAs.
- Control Bank C 12 RCCAs.
- Control Bank B contains 12 RCCAs.
- Control Bank A contains 8 RCCAs.
- Shutdown Bank SA contains 20 RCCAs.
- Shutdown Bank SB contains 12 RCCAs.
- Shutdown Bank SC contains 16 RCCAs.

The bank name allocation is definable on a cycle-by-cycle basis. RCCA bank composition is subject to change based on the core operating limits report (COLR) requirements. The COLR is addressed in Chapter 16, Specification 5.6.3.

Bank Sequence and Overlap

The rods move in the bank configuration for all cases except in the case of the partial trip (PT). In a PT, the sub-bank of rods that are dropped is a function of rod worth and relative position in the core.

The bank insertion and withdrawal sequence and overlap are defined by the control bank insertion limits. Control rod banks are withdrawn and inserted in a prescribed sequence and overlap. For withdrawal, the sequence is shutdown SA, shutdown SB, shutdown SC, control A, control B, control C, and control D. The insertion sequence is the reverse of the withdrawal sequence. The control bank rods move in a prescribed overlap that is specified in the COLR. This means that during bank withdrawal,

control D will begin withdrawal before control C is fully withdrawn. Conversely, control C will begin insertion before control D is fully inserted.

7.7.2.1.2 Average Coolant Temperature Control

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The ACT control function is designed to maintain a programmed average RCS average temperature (T_{avg}) by regulating core power. The ACT control is the predominant function of core control. The control logic is illustrated in Figure 7.7-1—Average Coolant Temperature Control Logic. Additionally, Figure 7.7-6—Signal Flow from ~~PS~~SCDS through CU is provided ~~as an example~~ to illustrate how the logic for ACT control could be assigned to the RCSL functional units and how the RCSL system architecture can support its assigned functions. Section 7.1.1.4.5 describes the Acquisition Units (AU), Control Units (CU), and Drive Units (DU). The ACT control logic consists of the following four main elements:

- The mismatch between turbine generator load and reactor power (i.e., power imbalance feed forward).
- The formation of the ACT control setpoint based on power level.
- The difference between the measured average RCS temperature and the desired average temperature (i.e., temperature error).
- The relationship between the sum of the two error signals and the resulting rod movement actuation requests.

The ACT setpoint serves as an input to determine the temperature error. The setpoint follows the ACT versus the power relationship as shown in Figure 4.4-7.

The ACT control function consists of two main error signal channels which are summed to provide a total error input signal to the rod speed control program. The rod speed program is shown in Figure 7.7-2—Rod Speed Control Program. The rods that are used to perform this function are designated as control bank rods that move into or out of the core in a prescribed manner, referred to as sequence and overlap, that is followed during insertion or withdrawal. The signal output of the rod speed program is a digital pulse that determines both rod stepping speed and direction (i.e., insertion or withdrawal). The two error channels are:

- Average temperature error - Difference between the second highest (auctioneered) measured loop T_{avg} and the ACT setpoint.
- Power imbalance feed-forward error - Mismatch between turbine generator load and reactor power.

The power imbalance feed-forward error signal and the temperature error signal are combined additively to produce a total error signal. This total error signal is the output that determines whether the control rods need to be inserted or withdrawn and

the speed at which the movement needs to occur. If the total error is negative, rods are withdrawn. If the total error is positive, rods are inserted. The rod speed control program determines the rod movement as a function of total temperature error.

At or near full power operation, this function uses boron addition and dilution batches as final control elements.

ACT Control using Boron Addition and Dilution

Rod movements are used to respond to rapid temperature deviations in the RCS or to a rapid generator power increase or decrease. For small, long lasting temperature deviations that occur due to fuel burn-up at or near full power, this function will use boron addition and dilution batches as the method to perform the ACT control in order to avoid rod movement. ACT control using boron or dilution batches does not consider the power imbalance feed-forward signal and will only correct an average temperature error that has existed for a predetermined period of time. Boration or dilution batches will occur within the dead bands shown in Figure 7.7-2—Rod Speed Control Program. If the total error is negative, a dilution batch will be requested. If the total error is positive, a boron addition batch will be requested. Dilution or boration batches are not permitted by this function when rods are moving and for a time after rods have moved. Dilution or boration batches are not permitted by this function when the axial power distribution is not within predetermined limits. Dilution or boration batches are subordinate to the actions of the AO control function described in Section 7.7.2.1.4.

The transition between the neutron flux control function to the ACT control function occurs at 25 percent reactor power.

7.7.2.1.3 Neutron Flux Control

The neutron flux control function is designed to control reactor power (i.e., neutron flux) during startup and shutdown operations, while the secondary pressure is controlled with the turbine bypass system (TBS). This function simplifies the constant power operation and facilitates the operator tasks during the startup of the turbine and the synchronization of the generator with the grid.

In the neutron flux control mode, the control bank movements operate in the same way as under the ACT control (i.e., in sequence and overlap). This function is used below 25 percent reactor power when the secondary steam pressure is controlled with the turbine bypass valves. At higher powers the ACT control function is used instead of the neutron flux control function.

The neutron flux control setpoint can be adjusted manually by the operator using the PICS.

When the reactor is at hot shutdown with all the banks inserted, the operator begins the first stage of the startup by withdrawing the shutdown and control banks. The withdrawal sequence requires that the shutdown banks are pulled to their all rods out (ARO) position before control banks are pulled. Control banks are then withdrawn in sequence and overlap as described in Section 7.7.2.1.1.

During startup (i.e., after exceeding a low reactor power permissive P5) and shutdown operation, the reactor power (i.e., neutron flux) can be controlled in conjunction with the main steam (MS) pressure control using the turbine bypass valves. The neutron flux control function blocks turbine synchronization at power levels less than a setpoint on increasing reactor power and blocks power reductions below a setpoint until the turbine is tripped on decreasing reactor power.

The neutron flux control acts on the rod control banks in the same way as the ACT control function. The neutron flux deviation is appropriately amplified to give an output signal corresponding to that from the ACT control.

7.7.2.1.4 Axial Offset Control

The AO control function is designed to maintain core axial power within analyzed limits. AO is a measure of the axial power distribution in the core. Extreme shifts in power distributions have an adverse impact on accident analysis results.

The AO control strategy works in conjunction with the ACT control to restore AO to within prescribed limits.

Above a predetermined power level, the AO control can be activated. If the AO exceeds a power dependant positive value, a dilution batch will be requested. This effectively raises the core-wide power and average coolant temperature, which causes the ACT control to insert rods, thereby correcting the AO. If the AO exceeds a power dependant negative value, a boration batch will be requested. This effectively lowers the core-wide power and average coolant temperature, which causes the ACT control to withdraw rods and correct the AO.

7.7.2.2 Operational Plant Control Functions

7.7.2.2.1 RCS Pressure Control

The RCS pressure control maintains the RCS pressure within allowable limits during Mode 1 through Mode 5. When in the automatic control mode, the RCS pressure control maintains the primary pressure at a setpoint value in steady-state operation and within an allowable range around its setpoint (i.e., control band) during transients, including startup and cooldown operations. Figure 7.7-3—RCS Pressure Setpoints indicates the control band relative to other RCS pressure setpoints.

When the automatic heatup and cooldown mode is selected, the RCS pressure control has an automatically generated temperature dependent setpoint. The automatic heatup and cooldown mode is selected during plant Mode 2 and Mode 3. The primary pressure is required to stay in an allowable range around the automatically generated setpoint. If the pressure drifts from the limits of the setpoint, the Max2 sliding pressure limitation function described in Section 7.7.2.3.11 is actuated. If the pressure progresses further from the temperature dependent setpoint to the high pressure (HP) or low pressure (LP) locking setpoints, the automatic heatup and cooldown is interrupted, and an alarm is sent to PICS.

RCS pressure control is performed by actuating pressurizer (PZR) heaters or PZR normal spray.

A manual control mode allows manual setpoint control and manual control of the actuators.

7.7.2.2.2 Pressurizer Level Control

The PZR level control provides:

- Sufficient RCS water inventory for cooling and for proper control of RCS pressure.
- A sufficient steam volume in the PZR to accommodate in-surges in the PZR from the RCS without causing an excessive pressure increase for normal operating transients. There is also sufficient water mass to accommodate out-surges from the PZR to the RCS without causing an excessive pressure decrease.

The function of the PZR level control is to maintain the PZR level at a setpoint value in steady-state operation and within the allowable range around its setpoints during normal operational situations, including startup and cooldown. When in automatic control mode, PZR level control channel makes sure that the PZR level remains within given limits (i.e., control band) around the setpoint. Figure 7.7-4—Pressurizer Level Setpoints indicates the control band relative to other PZR level setpoints.

The PZR level control monitors the PZR level for deviations from its setpoint during Mode 1 through Mode 4, and based on mode changes, actuates different control valves at the pressure reducing stations located in the CVCS letdown lines.

A manual control mode allows manual setpoint control and manual control of the pressure reducing valve actuators.

7.7.2.2.3 RCS Loop Level Control

The RCS loop level control function provides an automatic and continuous control of the RCS water inventory during mid-loop operation. In case of primary system

inventory changes, the control function limits the resulting mid-loop operation level deviations within the specified control band.

The loop level control function provides an automatic control of RCS water inventory by continuously monitoring the RCS loop level and controlling the coolant letdown flowrate.

RCS loop level control is maintained by a closed-loop control I&C function, which is put in service manually at cold shutdown conditions.

RCS loop level control is manually activated at cold shutdown conditions. Control actions are only effective when an HP charging pump is in operation and the volume control tank (VCT) bypass line is not opened.

7.7.2.2.4 Steam Generator Level Control

The steam generator (SG) water level control automatically maintains SG level by matching feedwater flow to steam demand. The level can also be controlled manually.

This SG level control I&C function provide the following:

- Sufficient water level for heat removal from the primary to secondary side.
- Minimizes moisture carryover to the turbine.

The SG level control I&C function maintains the SG level at a setpoint value in steady-state operation during heatup and cooldown (Mode 1 through Mode 4), and within allowable limits (called the control band) during normal operational transients. Figure 7.7-5—Steam Generator Level Setpoints illustrates the control band relative to other SG level setpoints.

This function acts on the following valves in the main feedwater system (MFWS) to control SG water level:

- Full load control valve (FLCV).
- Low load control valve (LLCV).
- Very low load control valve (VLLCV).

The system can be operated in the following modes:

- Automatic control mode which controls SG water level within given limits of a setpoint. Automatic control mode is the normal mode of operation.
- Manual control mode.

7.7.2.2.5 Main Steam Pressure Control

The purpose of the MS pressure control function is to provide MS overpressure control and limitation in case of load reduction due to load steps, load ramps, or load rejection. MS pressure is controlled by automatically modulating the turbine bypass valves.

During normal power operation, this function is realized by maintaining a floating MS pressure setpoint above the measured MS pressure. As the measured pressure changes, the setpoint changes accordingly. However, a limitation is placed on the rate of change of the setpoint so that if the measured pressure increases at a rate greater than the limitation of the floating setpoint, the turbine bypass valves will be opened. The turbine bypass valves close and are prevented from opening on high condenser backpressure or high hot well level.

During plant heatup and cooldown operations, the operator can adjust a target pressure setpoint which is adapted with a limited temperature gradient. Based on the target pressure setpoint, the turbine bypass valves control MS pressure and thus reactor coolant temperature. Locking logic is provided to interrupt the automatic heatup or cooldown process when RCS parameters deviate from their setpoint thresholds.

When partial cooldown is initiated, the MS pressure setpoint follows a specific partial cooldown setpoint which has priority over all other setpoints and locking signals.

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Following a reactor trip, in order to avoid primary overcooling, the MS pressure setpoint is immediately set to a fixed maximum pressure setpoint.

7.7.2.3 Process Limitation I&C Functions

7.7.2.3.1 Loss of One Reactor Coolant Pump Limitation

This limitation function is designed to avoid the low reactor coolant system flow rate (i.e., one loop) reactor trip function described in Section 7.2.

This function initiates a PT and a turbine load reduction when two RCS loop flow values of the same loop drop below the setpoint value and the P3 permissive is validated.

7.7.2.3.2 Axial Offset Limitation

The objective of this limitation is to survey the azimuthal power imbalance and make sure that the axial power distribution is within the parameters assumed in the safety analysis to limit the consequences at high power levels of accidents or AOOs for which a top-peaked core power distribution is penalizing. The limited parameter is the AO value calculated from the self powered neutron detectors. The AO operating range is

bounded by positive and negative thresholds. This function generates alarms and the blocking of the generator power increase.

This limitation function is inhibited below a low level of power.

The calculated AO is compared with thresholds derived from reactor power. When the threshold is met an action occurs to block the increase of generator power.

7.7.2.3.3 Reactor Power Limitation with Respect to Feedwater Flow Rate

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This limitation function limits the reactor power with respect to the feedwater flowrate. The limitation function is designed to correct plant conditions before a protective action due to low SG level occurs. The loss of one or more main feedwater (MFW) pumps leads to a large imbalance between power generation in the reactor and heat transfer to the main heat sink. ~~Operational~~Process control I&C will detect the failure of one pump and start a standby pump, if available, within a few seconds, thus allowing normal operation to continue.

This limitation function can handle the following three events:

- Loss of one MFW pump (if standby pump not available).
- Loss of all MFW pumps.
- Imbalance of feedwater flowrate and reactor power during startup phase.

Loss of One MFW Pump (if standby pump not available)

This limitation function deals with the loss of one MFW pump (if standby pump not available) by initiating a PT and a turbine load reduction. An imbalance between MFW flowrate and a nominal MFW flowrate (according to feedwater temperature and reactor power) activates a PT and a generator power reduction to a power level corresponding to operation with two MFW pumps.

Loss of All MFW Pumps

A low MFW flowrate combined with a high reactor power level is the criteria for the detection of the loss of all MFW pumps. In this case the limitation function will initiate a non-safety-related reactor trip, activate turbine trip, and close all FW FLCVs. The reactor trip signal resets this actuation.

Imbalance of Feedwater Flowrate and Reactor Power During Startup Phase

Indications of a low enough feedwater flowrate and a high enough reactor power leads to blocking the withdrawal of any RCCA. This prevents an increase of the reactor power without an increase of the MFW flowrate during the startup phase.

7.7.2.3.4 Reactor Power Limitation with respect to Generator Power

This limitation function limits reactor power after loss of generator load events. The objective is to limit the energy level of the primary system in case of load rejections or turbine trip in order to avoid reaching the RT criteria. This will be done by initiating a PT. The target reactor power level is determined by:

- The maximum of generator power.
- The minimum PT target power.

In case of turbine trip or load rejection to house load, the plant is first stabilized at minimum PT target power while heat removal is performed via the turbine bypass valves. A further controlled reduction to the minimum load reactor power will then be done by ACT control.

7.7.2.3.5 Reactor Power Limitation with respect to Thermal Power

The reactor power limitation with respect to thermal power function is designed to maintain reactor power below 100 percent rated thermal power. This function provides the capability to adjust turbine power and indirectly reactor power due to cooling tower temperature changes that affect overall plant efficiencies. The reactor power signal is selected from the highest of the following:

- Continuous secondary calorimetric calculation (i.e., above 25 percent power).
- ~~Median select excore power range indication of reactor power.~~
- Median select RCS enthalpy indication of reactor power.

The continuous secondary side calorimetric uses the following sensors and parameters as input:

- Feedwater flow rate for each train of feedwater (refer to Section 10.4.7.5 and Figure 10.4.7-1).
- Feedwater temperature for each train of feedwater (refer to Section 10.4.7.5 and Figure 10.4.7-1).
- Feedwater pressure for each train of feedwater (refer to Figure 10.4.7-1).
- Steam generator blowdown flow rate for each steam generator (refer to Figure 10.4.8-1).
- Steam generator blowdown temperature for each steam generator (refer to Figure 10.4.8-1).

- Reactor coolant system charging flow rate (refer to Section 9.3.4.5 and Figure 9.3.4-1, Sheet 5 of 9).
- Reactor coolant system charging flow temperature (refer to Section 9.3.4.5 and Figure 9.3.4-1, Sheet 5 of 9).
- Reactor coolant system charging flow pressure (refer to Section 9.3.4.5 and Figure 9.3.4-1, Sheet 5 of 9).
- Reactor coolant system letdown flow rate from both high pressure reducing stations in the chemical and volume control system (refer to Section 9.3.4.5 and Figure 9.3.4-1, Sheet 1 of 9).
- Reactor coolant system letdown flow temperature (refer to Section 9.3.4.5 and Figure 9.3.4-1, Sheet 1 of 9).
- Reactor coolant system letdown flow pressure (a constant value is assumed).
- Main steam pressure for each steam generator (refer to Figure 10.3-1).
- The power losses from the reactor coolant system (including the control rod drive mechanisms) to the ambient air (a constant value is assumed).
- The reactor coolant pump power (a constant value is assumed).
- The pressurizer heater power (a constant value is assumed).
- The moisture content of the main steam (a constant value is assumed).

The enthalpy of the main steam flow, main feedwater flow, steam generator blowdown flow, charging flow, and letdown flow are calculated using the corresponding pressures and/or temperatures. The continuous secondary calorimetric calculation of reactor thermal power is performed according to methodology outlined in Reference 3, which has been accepted by the NRC, per Reference 4. As an analytical requirement, 0.48 percent uncertainty on core thermal power was assumed in the safety analysis. However, the measurement requirements for the U.S. EPR allow the secondary side calorimetric to calculate reactor thermal power within a ± 0.40 percent uncertainty. To achieve the required uncertainty in the secondary side calorimetric algorithm, the elemental uncertainties of the instrument strings and parameters, previously mentioned, are verified to comply with requirements provided in Table 7.7-2—Elemental Uncertainties for Secondary Side Calorimetric.

The control logic compares the mismatch between main turbine and generator load and the highest of the previously listed power signals and takes actions when reactor power exceeds 100 percent. There are two thresholds. The intent of the first is to alert the operator and take action to prevent further power increase. The intent of the second threshold is to reduce power to 100 percent.

A COL applicant that references the U.S. EPR design certification will, following selection of the actual plant operating instrumentation and calculation of the instrumentation uncertainties of the operating plant parameters, prior to fuel load, calculate the primary power calorimetric uncertainty. The calculations will be completed using an NRC acceptable method and confirm that the safety analysis primary power calorimetric uncertainty bounds the calculated values.

7.7.2.3.6 Rod Drop Limitation

The objective of this limitation function is to detect the spurious drop of RCCAs and to reduce the turbine generator power level to match the reactor power reduction due to the dropped RCCAs.

This limitation function is designed to avoid reactivity compensation by core control functions after the RCCAs drop and to avoid the low departure from nucleate boiling (DNBR) and high linear power density (HLPD) protective actuations after one or more RCCAs drop into the core.

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Rod drop is detected in the ~~protection system (PS)~~RCSL system based on the RCCA position measurements. In each ~~PS~~RCSL division, a quarter of the RCCAs are monitored and the~~.~~ ~~F~~four ~~(i.e., one per PS division)~~ RCCA drop detection logic signals (i.e., one per RCSL division) are ~~acquired in RCSL and~~ voted one out of four.

The other criterion indicating an RCCA drop is derived from the decrease of the reactor power level (i.e., neutron flux from power range detectors). The derivative of the four nuclear power signals are compared with a low threshold and voted one out of four.

The limitation will be actuated if both criteria coincide and no intended PT has been initiated by other limitation functions.

7.7.2.3.7 Intermediate Range High Neutron Flux Limitation

This limitation function is designed to avoid the high neutron flux (i.e., intermediate range) and low doubling time (i.e., intermediate range) reactor trips when an excessive reactivity increase occurs during reactor startup from a subcritical or a low power startup condition. At the limitation criteria the withdrawal of any RCCA is blocked.

Limitation signals are generated in RCSL, which combine the following criteria:~~Each RCSL division receives four binary limitation signals (i.e., one per PS division). Each limitation signal from the PS combines the following criteria:~~

- Low doubling time IR limitation threshold.
- High neutron flux IR limitation threshold.

- Manual inhibition above a low power level (permissive P6).

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If these criteria are met in two out of four RCSL divisions,~~a PS division, a vote signal is sent from that PS division to RCSL. If two out of four vote signals are received by RCSL from the PS,~~ the following actions are performed:

- RCCA withdrawal is blocked.
- Alarm on PICS.

7.7.2.3.8 High Linear Power Density Limitation

There are three sub-functions to the HLPD limitation function. The three sub-functions are:

- Block function.
- Reduction function.
- PT function.

The HLPD limitation sub-functions are designed to avoid a reactor trip on HLPD for each transient that leads to an uncontrolled increase of the linear power density of the reactor core. This function initiates a PT and a fast turbine load reduction.

For the block and reduction sub-functions, a calculation of the linear power density (LPD) in the lower and upper portions of the core is performed in RCSL. In each sub-function the calculated LPD values for the upper and lower portions of the core are compared to threshold levels for each portion of the core. The self-powered neutron detectors (SPND) provide input for the calculation of the LPD values in RCSL.

The threshold levels for the block sub-function are above the threshold levels for the reduction sub-function, and therefore, the block sub-function actuates before the reduction sub-function.

Violation of the block sub-function threshold levels results in the following actions:

- Block dilution signal (for lower core half threshold level violation only).
- RCCA bank withdrawal blocking signal.
- Generator power increase blocking signal.
- Block lead control bank insertion (for lower core half threshold level violation only).

Violation of the reduction sub-function threshold levels results in the following actions:

- Reduce generator power signal.
- Insert lead control bank (for upper core half threshold level violation only).

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For the PT sub-function, the actuator logic signals are generated in the ~~PS~~RCSL and are inhibited below a low power level by the P2 permissive ~~P2~~. ~~Each division of the PS provides a vote input to the two out of four voting logic in RCSL.~~ When these actuator logic signals are generated, ~~two votes for the PT sub-function are received by RCSL,~~ the following actions are performed:

- PT.
- Turbine load reduction.

7.7.2.3.9 Low Departure from Nucleate Boiling Limitation

There are three sub-functions to the low DNBR limitation function. The three sub-functions are:

- Block function.
- Reduction function.
- PT function.

These functions are designed to correct conditions to avoid the low DNBR protective functions as described in Section 7.2. The functions provide DNBR margin with respect to the DNB criterion.

For the block and reduction sub-functions, a calculation of the minimum DNBR value is performed in RCSL. The following are the inputs for the calculation of the minimum DNBR value in RCSL:

- Power density distribution of the hot channel which is derived from SPNDs.
- Average reactor inlet temperature.
- Average PZR pressure.
- Core flowrate derived from average RCP speed.

The threshold level for the block sub-function is above the threshold level for the reduction sub-function, and therefore, the block sub-function will actuate before the reduction function.

If the DNBR value drops below the threshold for the block sub-function, the following actions are performed:

- Alarm in the main control room (MCR).
- Block RCCA withdrawal.
- Block generator power increase.

If the DNBR value drops below the threshold for the reduction sub-function, the following actions are performed:

- Reduce generator power setpoint.
- Insert RCCAs.

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The actuator logic signals for the low DNBR PT limitation sub-function are generated in the ~~PS~~RCSL and are inhibited below a low power level by the P2 permissive ~~P2~~. ~~Each division of the PS provides a vote input to the two out of four voting logic in RCSL.~~ When these actuator logic signals are generated ~~two votes for the low DNBR PT limitation function are received by RCSL~~, the following actions are performed:

- PT.
- Turbine load reduction.

7.7.2.3.10 RCS Dilution (Shutdown Condition) Limitation

This limitation function is designed to avoid the actuation of the antidilution in standard shutdown states protective function as described in Section 7.3. This function contains the following sub-functions:

- Limitation in case of low RCS boron concentration.
- Prevent dilution at shutdown.

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In the first sub-function, the RCS boron concentration is calculated in the ~~PS~~boron concentration measurement system (BCMS) based on boron meter measurements and on charging flowrate measurements. This value is acquired by the RCSL, via the SCDS interface, where it is compared to the permissible shutdown state boron concentration. The low concentration limitation threshold is generated in the ~~PS~~RCSL at a higher threshold than the antidilution at a shutdown condition state protection criterion. ~~Four redundant limitation signals from the PS are transferred to RCSL.~~ When the limitation signals are generated in two out of four RCSL divisions~~the two out of four voting is fulfilled in RCSL~~, the following actions are initiated:

- Boron addition with maximum injection rate.
- Isolation of demineralized water injection lines of the reactor boron and water makeup system (RBWMS). Both demineralized water injection pumps are shut off and both control valves are closed with highest priority.

The second sub-function is activated when shutdown conditions are detected (reactor trip or no RCPs running). In this sub-function, boron concentration injected by RBWMS is measured. If the injected concentration is below the permissible value then the demineralized water injection lines will be isolated.

7.7.2.3.11 Reactor Coolant System Pressure Limitations

When the RCS pressure goes out of the normal operating range, the following RCS pressure limitation functions can be activated. These functions are designed to correct RCS pressure transients before a RT setpoint is reached, or to protect equipment. These functions have a more stringent action than the RCS pressure control function as described in Section 7.7.2.2.1. A graphical presentation of the RCS pressure limitation setpoints in relation to protective function setpoints and the control band is presented in Figure 7.7-3.

In case of post-accident operations, the operator is able to inhibit the activation of the RCS pressure limitation functions from PICS.

Max2 Pressure Function

The Max2 pressure function improves the availability of the plant by avoiding an RT on the Max2p setpoint (i.e., high PZR pressure). When the RCS pressure measurement reaches the setpoint, this function de-energizes the PZR heaters and actuates the normal spray. If the normal spray is not functional, auxiliary spray is actuated. The normal spray availability is determined based on RCP speed or the loop flowrate.

This function is operational in Mode 1 through Mode 3.

Max2 Sliding Pressure Function

The Max2 sliding pressure function improves plant availability by preventing a lock of the automatic heatup and cooldown on Max2p and limits the temperature differences between the PZR and RCS loops. The Max2 sliding pressure function is similar to the Max2 pressure function except this function has an automatically generated temperature dependent setpoint and is operational during the automatic heatup and cooldown.

When the pressure measurement reaches the setpoint, this function de-energizes the PZR heaters and actuates the normal spray. If the normal spray is not functional, the auxiliary spray is actuated. The normal spray availability is determined based on RCP speed or the loop flowrate.

Residual Heat Removal System Function

The RHRS function protects the RHRS equipment from overpressurization and prevents challenging the PZR safety relief valves (PSRV) during low temperature operation. The setpoint for the RHRS function is below the RHRS maximum pressure. This function is similar to the Max2 pressure function except the setpoint is temperature dependent and the function is operational during Mode 4 through Mode 6 when the P14 permissive has been acknowledged. The RHRS is normally connected to the RCS for decay heat removal when the P14 permissive ~~process parameter~~ setpoints are met and the P14 permissive has been acknowledged by the operator.

When primary pressure reaches the RHRS function setpoint, this function de-energizes the PZR heaters and actuates normal spray. If the normal spray is not working properly, the auxiliary spray is actuated. The unavailability of the normal spray is detected by RCP speed or the loop flowrate.

Reactor Coolant Pump Function

The reactor coolant pump (RCP) function avoids a RT on Min2p (i.e., low PZR pressure) during Mode 1. It also protects RCPs from cavitation and keeps pressure from falling below the Min3p setpoint for initiation of safety injection. The RCP function setpoint is temperature dependent and below the nominal operating pressure setpoint of the RCS pressure control function. The function is operational in Mode 1 through Mode 5.

When pressure reaches the RCP function setpoint, the function secures the PZR spray and energizes PZR heaters.

Reactor Pressure Vessel Brittle Fracture Function

At low RCS temperatures, the PSRVs opening setpoint is lowered to protect the RCS from overpressurization. The lowering of the PSRV opening setpoints is performed by the PS and is described in Section 7.3. The RPV brittle fracture function is implemented in the PAS to prevent pressure from reaching the electrically controlled PSRV open setpoints when in Mode 4 and Mode 5. The RPV brittle fracture setpoint is temperature dependent.

When the pressure measurement reaches the setpoint of the function, the RPV brittle fracture function stops the CVCS charging pumps, the medium head safety injection (MHSI) pumps, ~~the extra borating system pumps,~~ and de-energizes the PZR heaters. If the MHSI pumps are running due to a safety function actuation, the MHSI pumps will continue to run.

7.7.2.3.12 Pressurizer Level Limitations

The PZR level limitation functions are designed to backup the normal PZR level control function when the normal control function is outside of its normal control band. This process is achieved by performing actions that supplement the normal control function to return the RCS to the 100 percent power, level control band following a transient that caused the deviation. This improves the availability of the plant by correcting PZR level before reaching RT setpoints and other safety protective function setpoints. A graphical presentation of the PZR level limitation setpoints in relation to protective function setpoints and the control band is presented in Figure 7.7-4.

Max2 Level Function

The Max2 level function avoids overfilling the PZR and flooding the PZR spray nozzles. It improves the availability of the plant by correcting PZR level before reaching the Max1p RT setpoint (i.e., high PZR level) in Mode 1 and Mode 2. When the PZR level reaches the Max2 level setpoint, this function isolates the injection coming from the charging line and the auxiliary spray, thus limiting the increase of PZR level. When the PZR level returns below the Max2 level setpoint, the activating signal is withdrawn. Furthermore, the charging line isolation valve and the auxiliary spray line isolation valve remain closed; however, the operator can manually reopen the valves. An alarm on PICS indicates that the Max2 level function has been actuated.

This function is operational in Mode 1 through Mode 4.

In case of post-accident operations, the operator is able to inhibit the activation of this function from PICS.

Δ Max Function

The Δ Max function improves the availability of the plant by correcting PZR level before reaching the RT setpoint on Max1p (i.e., high PZR level). This function acts before reaching the Max2 level setpoint. The Δ Max level setpoint corresponds to a certain percentage above the PZR level control function setpoint. When the PZR level reaches the Δ Max level setpoint, this function adjusts the letdown flowrate at the HP reducing stations of CVCS to the maximum flowrate limit. An alarm on PICS indicates that the Δ Max level function has been actuated. When the PZR level returns below the Δ Max level setpoint, the activating signal is withdrawn and the HP reducing station returns to automatic PZR level control.

This function is operational in Mode 1 through Mode 5 when RCS temperature is greater than approximately 140°F and an HP reducing station of the CVCS is in operation.

In case of post-accident operations, the operator is able to inhibit the activation of this function from PICS.

Δ Min Function

The Δ Min function improves the availability of the plant by correcting PZR level before reaching the RT setpoint on Min2p (i.e., low PZR pressure). This function acts before reaching the Min2 level setpoint. The Δ Min level setpoint corresponds to a certain percentage below the PZR level control function setpoint. When the PZR level reaches the Δ Min level setpoint, this function increases the water injection into the RCS to the maximum possible flowrate by starting a second CVCS pump and adjusting the HP reducing station to the minimum flowrate limit. An alarm on PICS indicates that the Δ Min level function has been actuated. When the PZR level returns above the Δ Min level setpoint, the activating signal is withdrawn and the HP reducing station returns to automatic PZR level control. The CVCS charging pumps continue to run, however, the operator has the capability to manually stop a pump.

This function is operational in Mode 1 through Mode 5 when RCS temperature is greater than approximately 140°F and a HP reducing station of the CVCS is in operation.

In case of post-accident operations, the operator is able to inhibit the activation of this function from PICS.

Min2 Level Function

The Min2 level function avoids emptying the PZR, thus avoids activating the emergency core cooling criteria. When the PZR level reaches the Min2 function level, this function isolates the CVCS letdown lines and thus limits the decrease of RCS water inventory. An alarm on PICS indicates that the Min2 level function has been actuated. When the PZR level returns above the Min2 level setpoint, the activating signal is withdrawn, but the CVCS letdown lines remain isolated; however, the operator has the capability to manually reopen the letdown lines.

This function is operational in Mode 1 through Mode 5 when RCS temperature is greater than approximately 140°F and an HP reducing station of the CVCS is in operation.

In case of post-accident operations, the operator is able to inhibit the activation of this function from PICS.

Min3 Level Function

The Min3 level function protects the PZR heaters from being uncovered and is designed to prevent severe damage to the PZR heaters and also a potential breach of

the RCS. When the PZR level reaches the Min3 function level setpoint, the PZR heaters are de-energized. An alarm on PICS indicates that the Min3 level function has been actuated. When the PZR level returns above the Min3 level setpoint, the PZR heaters are automatically switched back to RCS pressure control.

This function is operational during all plant modes.

The Min3 level function cannot be inhibited.

7.7.2.3.13 Reactor Coolant System Loop Level Limitation

The RCS loop level limitation function continuously monitors the loop level during mid-loop operation.

The RCS loop level limitation function makes sure that the minimum and maximum admissible water levels are in the RCS loops in case of transients. This limitation function acts when an overshoot of the control band limit occurs. This function prevents the actuation of safety functions by the PS.

The RCS loop level limitation function considers the water level required to protect the low head safety injection (LHSI) pumps from cavitation during mid loop operation.

This limitation function also prevents inadvertent filling of the loops. Filling the loops interrupts the flow area for the purge gas in the loop and the necessary free water surface for removal of noble gas. This could endanger personnel working in the SG bowls, and could potentially discharge coolant to the containment via open SG manways.

The RCS loop level limitation function fully closes the LP and HP reducing station of the CVCS letdown line when the RCS water level falls below a dedicated threshold that is below the lower control band limit of the RCS loop level control function. This limitation function fully opens the LP reducing stations to increase the coolant letdown flowrate when the water level exceeds a dedicated threshold above the upper control band limit of the RCS loop level control function. Both the upper and lower thresholds of this function are constant.

The limitation function is automatically activated during the plant shutdown procedure when the operating range of the LHSI RHRS is reached.

The RCS loop level limitation function is disabled beyond its specified operating range in order to exclude the occurrence of inadvertent actuation signals.

7.7.2.3.14 Steam Generator Level Limitations

The SG level limitation functions are non-safety-related functions designed to correct SG level transients before a protective function setpoint is reached. A graphical

presentation of the SG level limitation setpoints in relation to protective function setpoints and the control band is presented in Figure 7.7-5.

In the case of post-accident operations, the operator is able to inhibit the SG level limitation functions from PICS.

High SG Level Limitation Function

The high SG level limitation function avoids RT at ~~MAX~~Max1p and returns the SG level to its normal operating range. It has higher priority over the SG level control function described in Section 7.7.2.2.4.

This function is operable in Mode 1 through Mode 4.

The high SG level limitation function receives input from the narrow range SG level and the FLCV position. Two setpoints, ~~MAX~~Max c1 and ~~MAX~~Max c2, are set between the SG level control function setpoint and the safety setpoint ~~MAX~~Max-1p, with ~~MAX~~Max c2 higher than ~~MAX~~Max c1. The LLCV and the FLCV in the MFWS are actuated depending on the SG level with respect to the ~~MAX~~Max c1 and ~~MAX~~Max c2 setpoints.

If SG level is greater than the ~~MAX~~Max c1 setpoint, a close order is sent to the FLCV first. If the FLCV position is below a certain position and the SG level is still above the ~~MAX~~Max c1 setpoint, then a close order is also sent to the LLCV. A close order to both the FLCV and the LLCV remain as long as the SG level is greater than the ~~MAX~~Max c1 setpoint.

If the SG level is greater than the ~~MAX~~Max c2 setpoint, close orders are immediately sent to both the FLCV and the LLCV regardless of their initial positions. When the water level is reduced to the intermediate region between ~~MAX~~Max c2 and ~~MAX~~Max c1, the close order is sent to the FLCV if it is not totally closed yet, while the LLCV would remain in its current position. When the FLCV is nearly closed, the LLCV will also be allowed to close.

The close orders to the FLCV and the LLCV will remain as long as the SG level is above the ~~MAX~~Max c1 setpoint.

Low SG Level Limitation Function

The low SG level limitation function avoids RT at ~~MIN~~Min1p and returns the SG level to its normal operating range. This function has higher priority over the SG level control function described in Section 7.7.2.2.4.

This function is operable in Mode 1 through Mode 4.

This function receives input from SG ~~L~~level (NR) and reactor power.

The low SG level limitation function defines a movable setpoint ~~MIN~~Min c1, set at a constant distance below the SG level control function setpoint and above the safety setpoint ~~MIN~~Min-1p. The ~~MIN~~Min c1 setpoint is designed to be movable at a constant distance from the SG level control function setpoint to prevent undesired actuation of the low SG level safety function during SG level setpoint reduction before an RCP restart.

When the SG level is less than ~~MIN~~Min c1 and reactor power is less than 20 percent, an open order is sent to the LLCV. SG level is controlled by the LLCV at this power level. The open order to the LLCV is maintained as long as the water level is less than the ~~MIN~~Min c1 setpoint. Once the level increases above than ~~MIN~~Min c1 setpoint, the control of the LLCV returns back to the automatic control mode.

When the SG level is less than ~~MIN~~Min c1 and reactor power is greater than 20 percent, an open order is sent to the FLCV and the LLCV. The open orders are maintained to both valves as long as the water level is less than the ~~MIN~~Min c1 setpoint. Once the level increases above than ~~MIN~~Min c1 setpoint, the control of the FLCV and the LLCV return back to the automatic control mode.

Very Low Flow SG Level Limitation Function

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The very low flow SG level limitation function deactivates the VLLCV ~~electronic~~signal stop and returns the SG level to the normal operating range. It has higher priority over the SG level control function described in Section 7.7.2.2.4.

This function is operable in Mode 2 and Mode 3.

The very low flow SG level limitation function deactivates the VLLCV ~~electronic~~ signal stop, which provides the minimum position limitation during the startup and shutdown phases. The FLCV and LLCV are manually closed during Mode 2 and Mode 3 and therefore the FLCV and LLCV are not controlled by this limitation function.

To prevent water hammer and thermal stratification phenomena on the SG feedwater nozzle, the VLLCV ~~electronic~~signal stop guarantees a minimum continuous feedwater flowrate by preventing the VLLCV from closing below the minimum flow position. However, this could potentially cause a high water level in the SG.

07.01-28

When the SG level is greater than the ~~MAX~~Max c1 setpoint, the VLLCV ~~electronic~~signal stop is deactivated and close orders are sent to the VLLCV. Once the SG level drops below the ~~MAX~~Max c1 setpoint, the VLLCV returns to the automatic control mode.

7.7.2.4 Non-Safety Control Systems Described in Other Sections

Table 7.7-1 provides a cross-reference to other sections of the final safety analysis report (FSAR) that contain information on I&C that support non-safety-related functions. The functions listed in Table 7.7-1 do not have direct influence on the process of nuclear power generation.

7.7.2.5 Safety Classification

07.01-28

With the exception of the SCDS, ~~T~~the I&C systems described in Section 7.7.1.1 and Section 7.7.1.2 are non-safety~~-~~-related. The functions that these systems implement provide control of important parameters, but are not necessary to ~~show~~provide protection against ~~DBEs~~AOOs and PAs. The SCDS serves only as the instrumentation interface and does not perform core control and plant control functions.

7.7.2.6 Effects of Control System Operation Upon Accidents

The effects of non-safety-related control system action and inaction on the transient response of the plant for ~~accidents and~~ AOOs and PAs are considered in the safety analysis addressed in Chapter 15.

07.01-28

The non-safety-related control functions maintain the major process variables of the NSSS in predefined and allowed ranges during normal power operation. The proper operation of the non-safety-related control functions is not necessary to provide protection against accidents.

7.7.2.7 Effects of Control System Failures

The effects of control system failures are minimized by the features described in this section.

Functions assigned to RCSL and PAS are redundant in more than one division. The failure of a function in one division is backed up by a redundant function in another division. The redundant functions and their associated equipment, including support systems are independent of each other. Independence is achieved by the following:

- Redundant functions are allocated to physically separated divisions.
- Electrical isolation between divisions.
- Erroneous signals or messages from one faulty division do not impair the functionality of the remaining divisions.

The primary source of power to the RCSL and PAS is provided by a battery backed source. The secondary power source is from a separate battery backed source fed from a different power bus. Upon loss of the primary source of power to PAS or RCSL, the secondary power source automatically and without interruption, maintains power. In

case of a total loss of power to the plant, the battery source permits continued operation of the plant controls for a two hour period.

Segregation of functions is provided by allocating functions related to core control in the RCSL and functions related to RCS parameters in the PAS. Failures of components in one non-safety-related system do not effect the functioning of the other non-safety-related system.

07.01-28

~~Data communication between the non-safety systems and other I&C systems is electrically isolated.~~

Control system failures are considered as event initiators in the safety analysis described in Chapter 15.

7.7.2.8 Environmental Control System

Environmental controls are provided to protect equipment from environmental extremes. Heating, ventilation, and air conditioning (HVAC) is provided to maintain ambient conditions in a range acceptable for proper operation of I&C equipment. Section 9.4 provides details on HVAC functions that maintain ambient temperature control where I&C equipment is located.

7.7.2.9 Independence

Electrical isolation and communication independence between safety and non-safety systems is provided by components of the safety I&C systems. Electrical isolation and communication independence is further described in Section 7.1.

7.7.2.10 Interactions between Safety-Related and Non-Safety-Related I&C Systems

Non-safety-related I&C systems use signal selection algorithms to select a process value on which to base process control actions. The controlling process value is selected from multiple input signals. The process controlling value is the result of a signal process algorithm which filters the values such that control of the process, feedwater flow for example, is based on intermediate values of the process variable so that extreme values (min or max) are ignored. Therefore, this control method reduces the possibility of the control system acting on an erroneous extreme value, as may result from instrument or other failures.

Feedwater flow is a controlled process variable that would cause significant plant transients if perturbed during 100% power operation. Without proper filtering, the process control system could be affected by a single process sensor malfunction or by an inadvertent perturbation of a sensor during maintenance. This type of process control input upset could result in a feedwater transient that would create challenges to safety systems.

7.7.2.11 Defense in Depth and Diversity

07.01-28

~~A methodology used to evaluate the adequacy of I&C design with respect to diversity and defense-in-depth is presented in the AREVA Diversity and Defense-in-Depth Methodology Assessment Technical Report (Reference 1).~~ Non-safety-related functions that are designed to provide diverse protective functions are described in Section 7.8.

7.7.2.12 Potential for Inadvertent Actuation

The non-safety-related control systems and functions are designed to limit the potential for inadvertent actuation and challenges to the safety-related systems. Many of the process limitation I&C functions described in Section 7.7.2.3 are designed to achieve optimum plant availability. These types of limitation functions act before protection functions, and thus restore normal operating conditions without challenging the protection thresholds for the most frequent accident conditions. The limitation thresholds are set before protection thresholds (as close as possible to them), but with a margin taking into account the counter-measure response time. After exceeding a limitation threshold, rapid corrective actions are automatically initiated. The typical limitation function action is the RCCA dropping called the PT which leads to a fast power decrease.

7.7.2.13 Control of Access

07.01-28

Physical access to I&C cabinets is restricted to authorized personnel. Unauthorized ~~electronic~~ access to system software via network connections is prevented by administrative controls. The loading of software and parameter changes via maintenance equipment is only possible in accordance with clearly defined procedures.

7.7.3 Analysis

The ~~control~~I&C systems described in Section 7.7.1.1 and Section 7.7.2.1 are those used for normal operation that are not relied upon to perform safety-related functions following AOOs or ~~accidents~~PAs. These systems control plant processes having an impact on plant safety.

07.01-28

The plant control systems are designed to prevent an undesirable condition in the operation of the plant that, if reached, is protected by the PS. The description and analysis of this protection is covered in Section 7.2 and Section 7.3.

How these ~~control~~I&C systems comply with the acceptance criteria and conform to guidelines set forth in NUREG-0800 (Reference 2) is described in Section 7.1.

7.7.4 References

07.01-28

1. Deleted.~~ANP-10304, Revision 1, "U.S. EPR Diversity and Defense-in-Depth Assessment Technical Report," AREVA NP Inc., December 2009.~~

2. NUREG-0800, Standard Review Plan, Section 7.7, "Control Systems," Revision 5, U.S. Nuclear Regulatory Commission, March 2007.

3. ER-157P, Topical Report, Revision 8, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System," Cameron Measurement Systems.

4. Final Safety Evaluation by the Office of Nuclear Reactor Regulation, Engineering Report ER-157P, Topical Report, Revision 8, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System," Cameron Measurement Systems," Project No. 1370.

Table 7.7-1—Cross Reference of Non-Safety-Related I&C Controls

System	Non--Safety--Related Control Function(s)	FSAR Section
Fuel Storage and Handling	Provides a means for handling fuel assemblies.	9.1
Essential Service Water System	Cooling of the Severe Accident Heat Removal System.	9.2.1
Component Cooling Water System	• Cooling of non-safety-related components and heat exchangers. • Cooling of the Severe Accident Heat Removal System.	9.2.2
Operational Chilled Water System	Cooling source for non-safety-related loads of HVAC.	N/A
Compressed Air System	Provides compressed air to non-safety-related components.	9.3.1
Chemical Volume and Control System	• Reactor coolant water purification and clean up. • -RCP sealing water supply. • Provide auxiliary spray to PZR.	9.3.4
Air Conditioning, Heating, Cooling and Ventilation Systems	Provide ambient air cooling for non-safety-related systems and components.	9.4
Fire Protection Systems	Detects and suppresses fires.	9.5.1
Turbine Generator	Converts the thermal energy supplied by the main steam supply system into electrical energy.	10.2
Condensate and Feedwater System	Provides feedwater to the steam generators (SG) at the required temperature, pressure and flow rate	10.4.7
Liquid Waste Management Systems	Receive and process radioactive liquid wastes from various systems.	11.2
Gaseous Waste Management Systems	Receive and process radioactive gaseous wastes from various systems.	11.3
Solid Waste Management Systems	Receive and process radioactive solid wastes from various systems.	11.4

Table 7.7-2—Elemental Uncertainties for Secondary Side Calorimetric

Input	Maximum Allowable Uncertainty at 100%NP
Feedwater Flow Rate	0.28% of the Actual Value
Feedwater Temperature	± 0.6°F of the Actual Value
Feedwater Pressure	± 25 psia of the Actual Value
Steam Pressure	± 25.4 psia of the Actual Value
Blowdown Flow Rate	± 5% of the Actual Value
Blowdown Temperature	± 3.0°F of the Actual Value
Charging Flow Rate	± 4% of the Actual Value
Charging Temperature	± 3% of the Actual Value
Charging Pressure	± 3% of the Actual Value
Letdown Flow Rate	± 4% of the Actual Value
Letdown Temperature	± 3% of the Actual Value
Letdown Pressure	± 3% of the Actual Value
Reactor Coolant Pump Power	± 20% of the Actual Value
Power Losses from the Reactor Coolant System	± 20% of the Actual Value
Pressurizer Heater Power	± 20% of the Actual Value
Steam Moisture Content	± 0.25% of the Actual Value

Figure 7.7-1—Average Coolant Temperature Control Logic

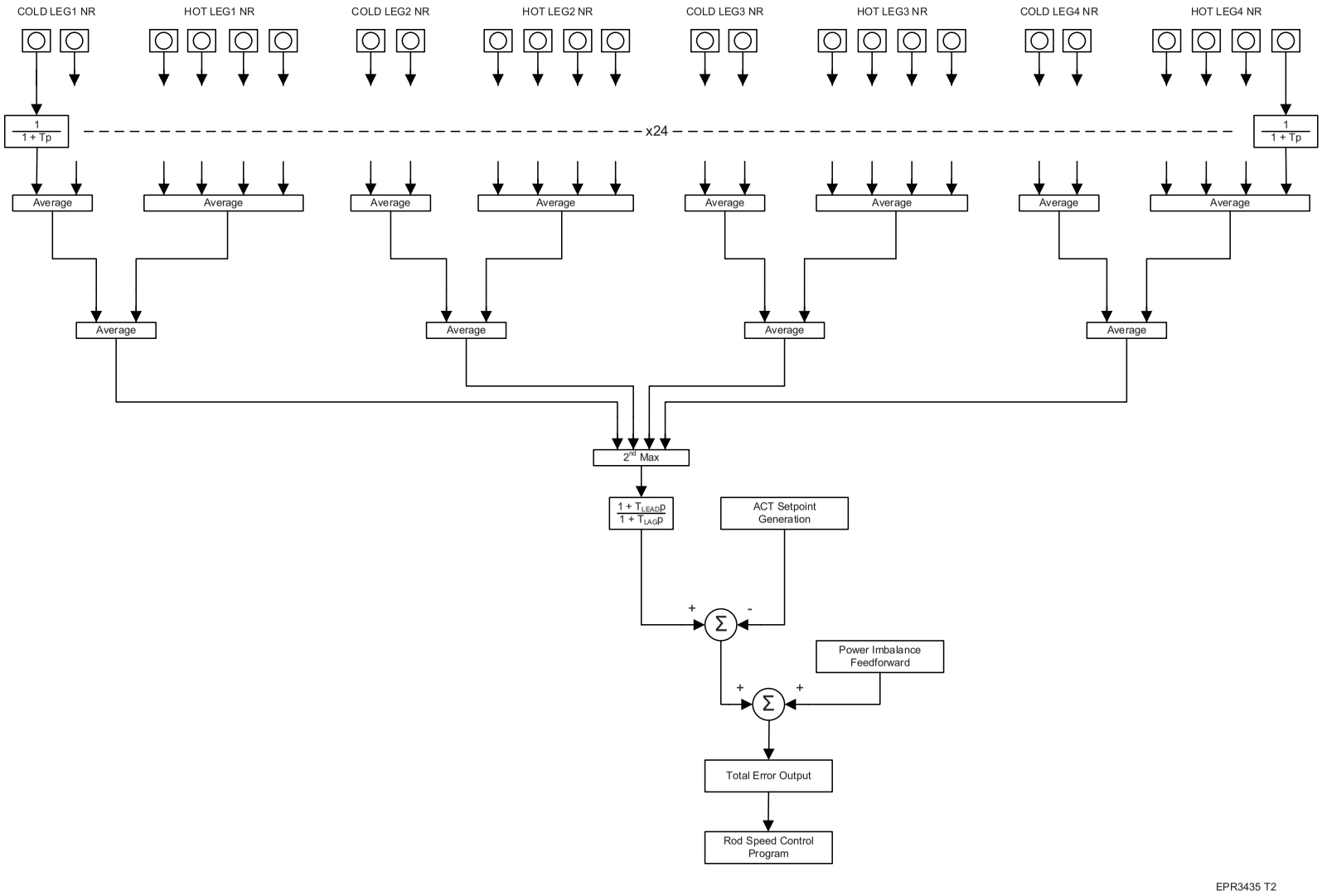

Figure 7.7-2—Rod Speed Control Program

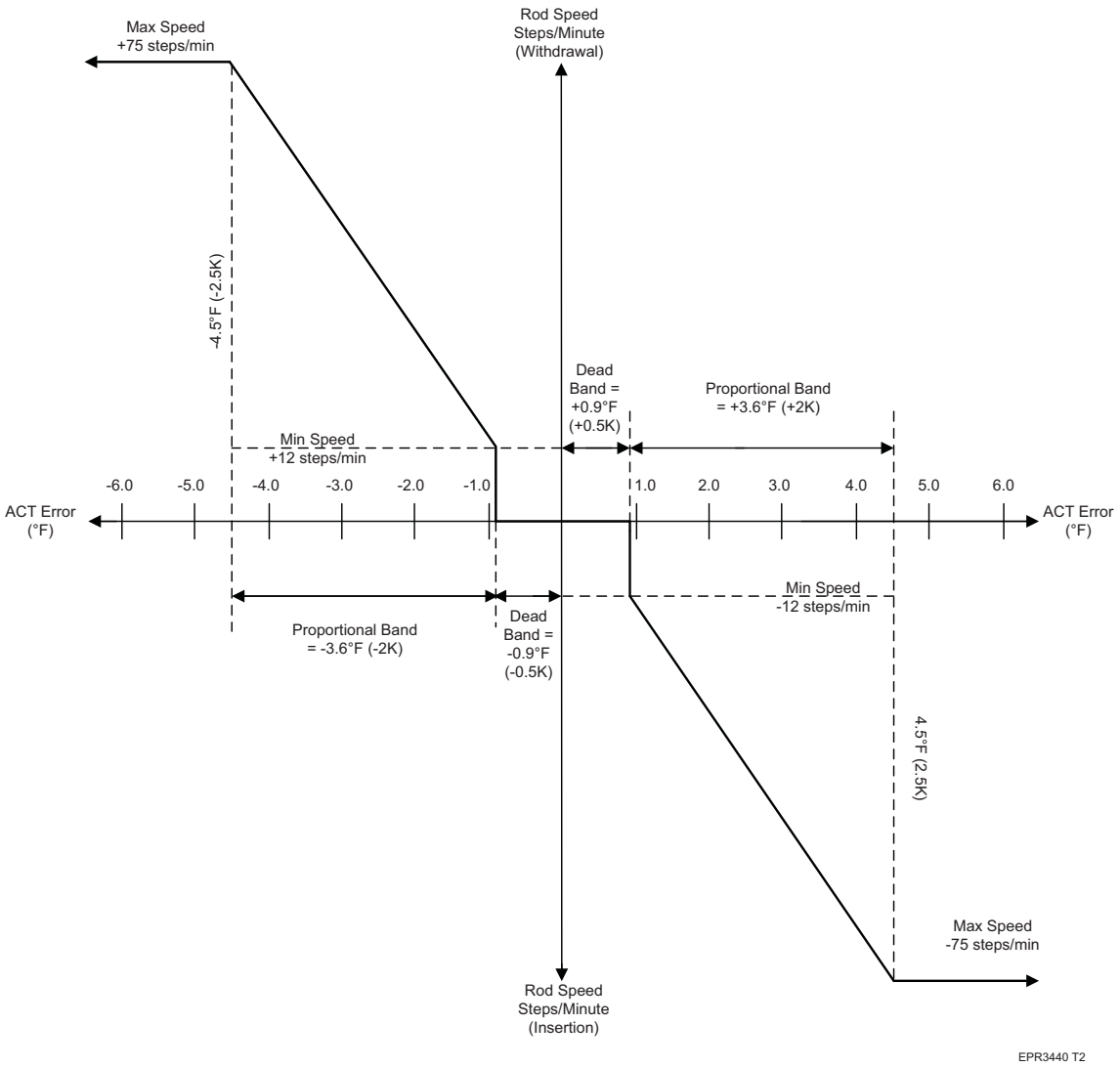

Figure 7.7-3—RCS Pressure Setpoints

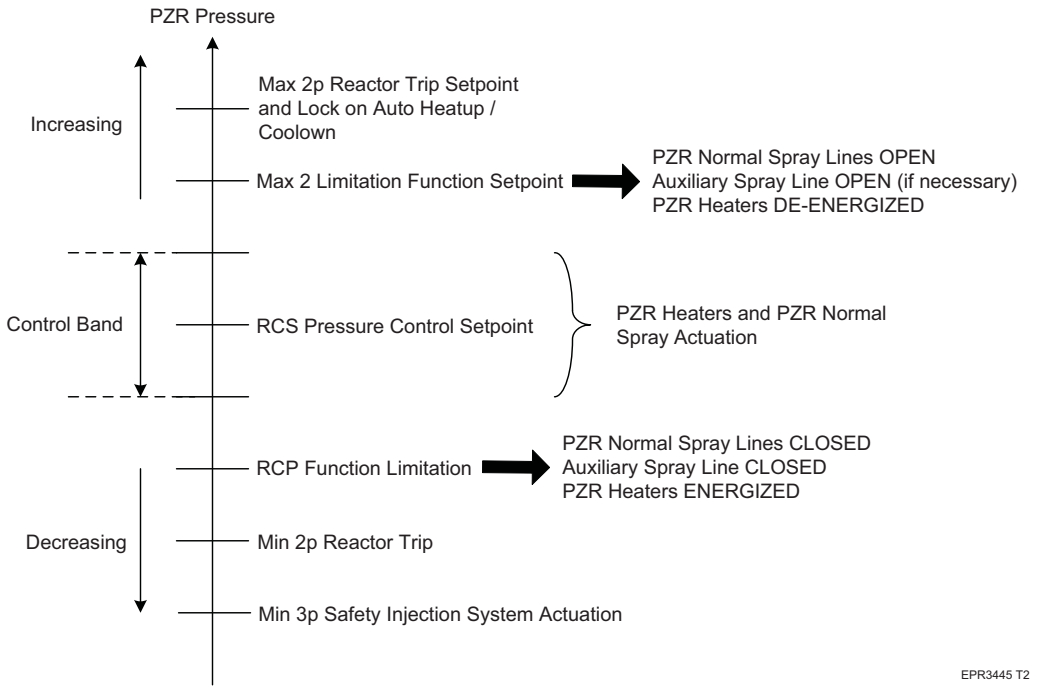

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Figure 7.7-4—Pressurizer Level Setpoints

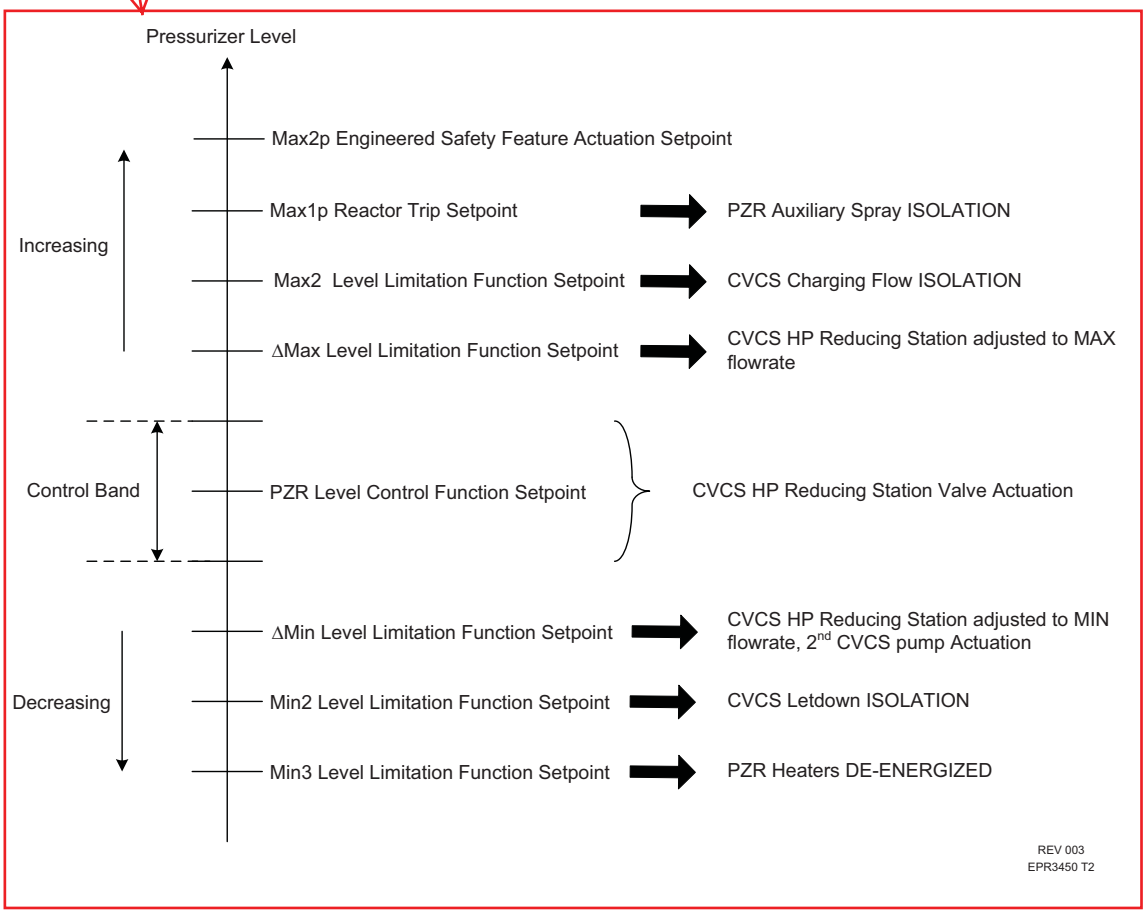

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Figure 7.7-5—Steam Generator Level Setpoints

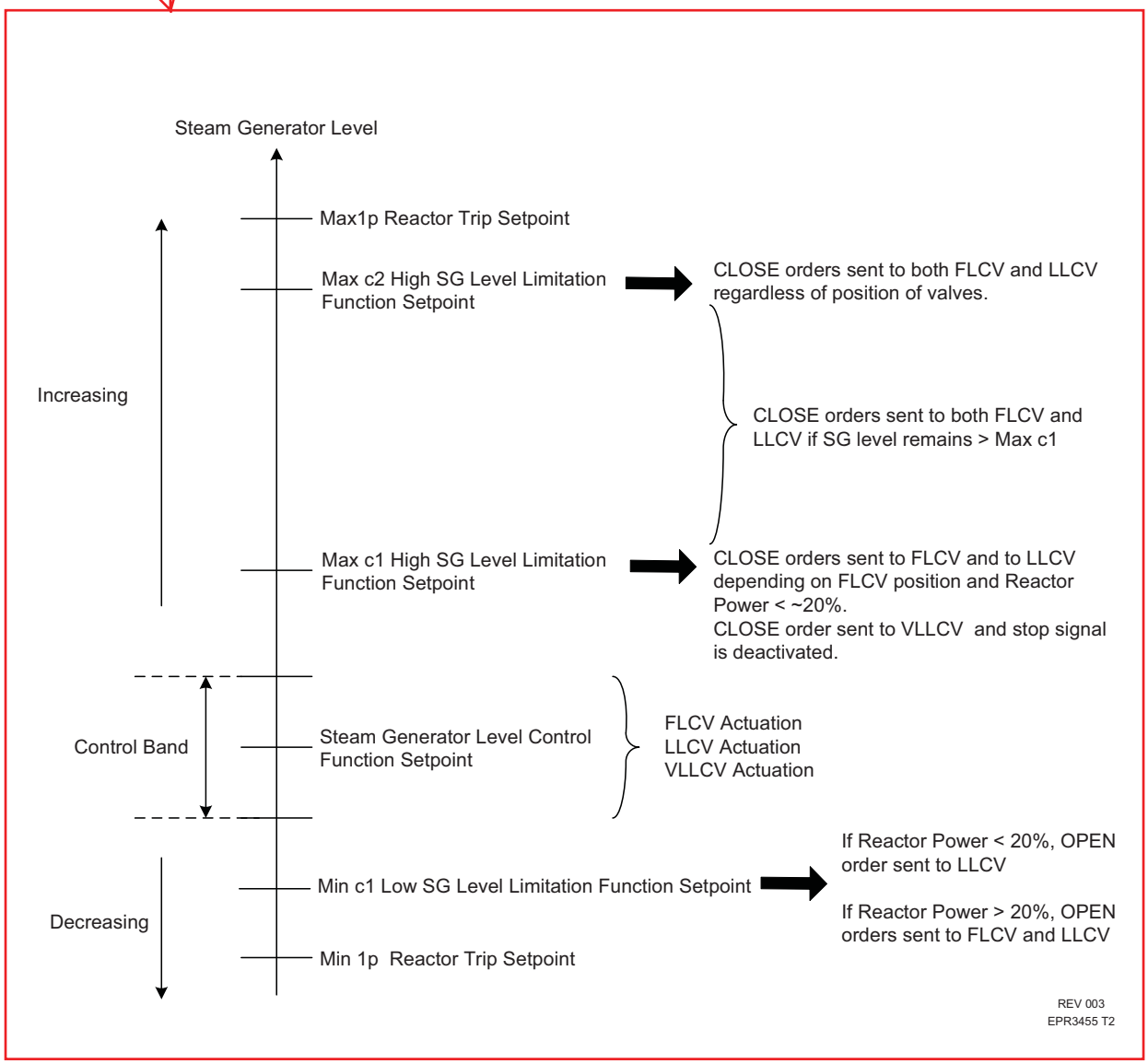

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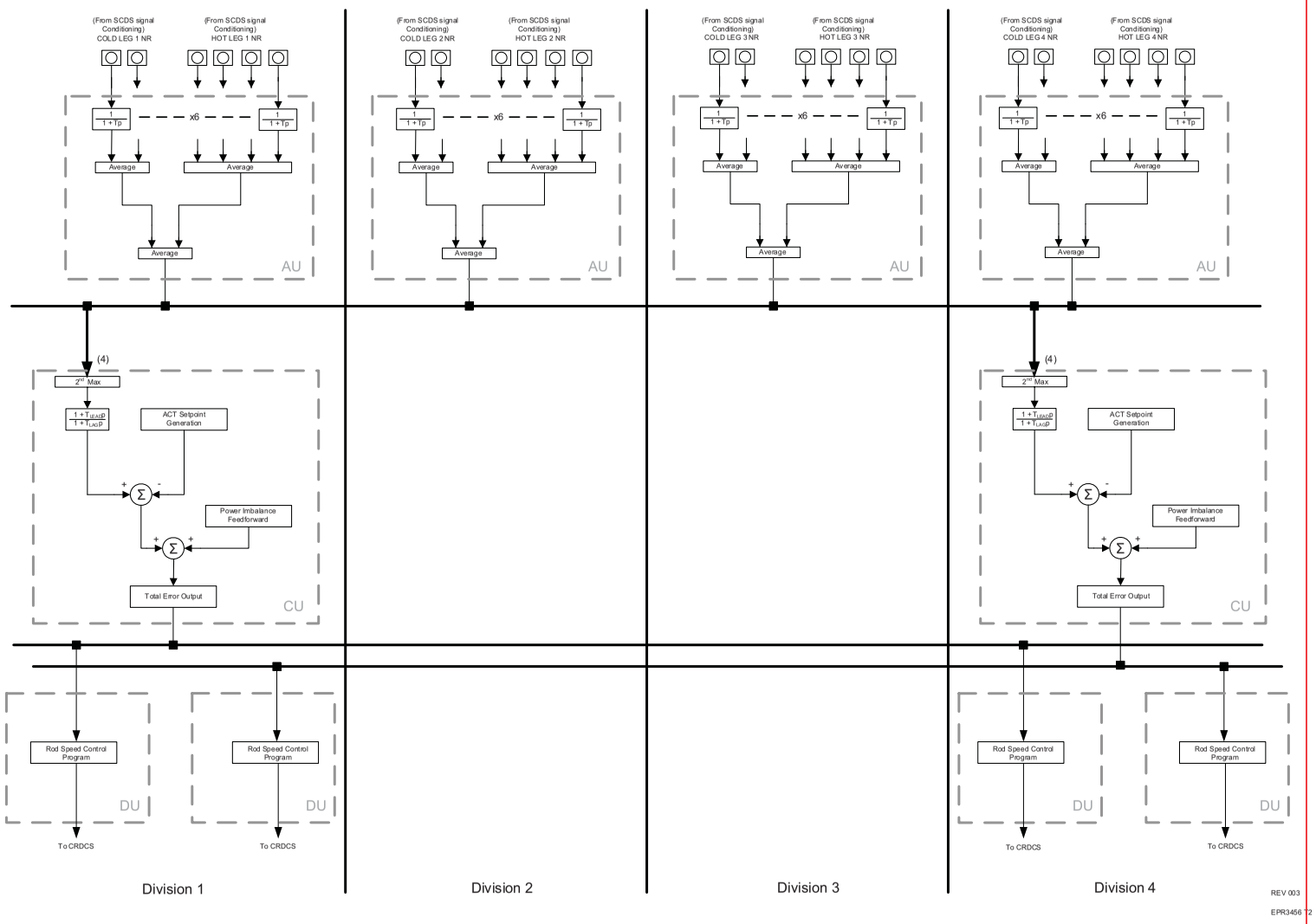
Figure 7.7-6—Signal Flow from ~~PS~~SCDS through CU



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