ArevaEPRDCPEm Resource

From: WILLIFORD Dennis (AREVA) [Dennis.Williford@areva.com]
Sent: Wednesday, August 10, 2011 4:36 PM
To: Tesfaye, Getachew
Cc: BENNETT Kathy (AREVA); DELANO Karen (AREVA); ROMINE Judy (AREVA); RYAN Tom (AREVA); NOXON David (AREVA)
Subject: Response to U.S. EPR Design Certification Application RAI No. 273, FSAR Ch 11, Supplement 16
Attachments: RAI 273 Supplement 16 Response US EPR DC.pdf

Getachew,

AREVA NP Inc. (AREVA NP) provided technically correct and complete responses to 10 of the 42 questions of RAI No. 273 on October 14, 2009. Supplement 1 to RAI No. 273 was sent on November 6, 2009 which responded to 17 of the 32 remaining questions and provided partial responses to 4 of the remaining 32. AREVA NP provided Supplement 2 on November 25, 2009 and Supplement 3 on December 8, 2009 to revise the commitment date for 11 of the remaining questions. AREVA NP provided Supplement 4 on December 10, 2009 to provide responses to 11 of the 15 remaining questions with a schedule to provide an FSAR markup for 9 of the questions. AREVA NP provided Supplement 5 on March 31, 2010 to provide a technically correct and complete response to 1 of the 4 remaining questions and provide FSAR markups for 3 of the 9 FSAR Markups. Supplement 6 to RAI No. 273 was sent on May 19, 2009 which responded to 1 of the 3 remaining questions. RAI 273 Supplement 7 sent on July 29, 2010, Supplement 8 sent on October 1, 2010, Supplement 9 sent on October 28, 2010, Supplement 10 sent on December 17, 2011, Supplement 11 on January 21, 2011 and Supplement 12 on March 3, 2011 provided a revised schedule for the remaining FSAR markups and responses. Supplement 13, sent on April 11, 2011 provided a technically correct and complete response to 2 of the 2 remaining questions and a revised schedule for the remaining 6 FSAR Markups. Supplement 14 sent on June 10, 2011, and Supplement 15 sent on August 9, 2011 provided a revised schedule for the remaining FSAR markups. The attached file, “RAI 273 Supplement 16 Response US EPR DC.pdf” provides technically correct and complete FINAL FSAR markups to the six remaining questions, 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 273 Questions 11.05-2, 11.05-5, 11.05-7, 11.05- 8, 11.05-9, and 11.05-10.

The following table indicates the respective pages in the response document, “RAI 273 Supplement 16 Response US EPR DC.pdf,” that contain AREVA NP’s response to the subject questions.

Question #	Start Page	End Page
RAI 273 — 11.05-2	2	3
RAI 273 — 11.05-5	4	5
RAI 273 — 11.05-7	6	6
RAI 273 — 11.05-8	7	7
RAI 273 — 11.05-9	8	9
RAI 273 — 11.05-10	10	10

This concludes the formal AREVA NP response to RAI 273, and there are no questions from this RAI for which AREVA NP has not provided responses.

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: Tuesday, August 09, 2011 6:35 PM
To: Getachew.Tesfaye@nrc.gov
Cc: BENNETT Kathy (RS/NB); DELANO Karen (RS/NB); ROMINE Judy (RS/NB); RYAN Tom (RS/NB); NOXON David (RS/NB)
Subject: Response to U.S. EPR Design Certification Application RAI No. 273, FSAR Ch 11, Supplement 15

Getachew,

AREVA NP Inc. (AREVA NP) provided technically correct and complete responses to 10 of the 42 questions of RAI No. 273 on October 14, 2009. Supplement 1 to RAI No. 273 was sent on November 6, 2009 which responded to 17 of the 32 remaining questions and provided partial responses to 4 of the remaining 32. AREVA NP provided Supplement 2 on November 25, 2009 and Supplement 3 on December 8, 2009 to revise the commitment date for 11 of the remaining questions. AREVA NP provided Supplement 4 on December 10, 2009 to provide responses to 11 of the 15 remaining questions with a schedule to provide an FSAR markup for 9 of the questions. AREVA NP provided Supplement 5 on March 31, 2010 to provide a technically correct and complete response to 1 of the 4 remaining questions and provide FSAR markups for 3 of the 9 FSAR Markups. Supplement 6 to RAI No. 273 was sent on May 19, 2009 which responded to 1 of the 3 remaining questions. RAI 273 Supplement 7 sent on July 29, 2010, Supplement 8 sent on October 1, 2010, Supplement 9 sent on October 28, 2010, Supplement 10 sent on December 17, 2011, Supplement 11 on January 21, 2011 and Supplement 12 on March 3, 2011provided a revised schedule for the remaining FSAR markups and responses. Supplement 13, sent on April 11, 2011 provided a technically correct and complete response to 2 of the 2 remaining questions and a revised schedule for the remaining 6 FSAR Markups. Supplement 14 sent on June 10, 2011 provided a revised schedule for the remaining FSAR markups.

The schedule for the FSAR markups for the 6 remaining questions is being revised as provided below:

Question #	Supplement Date (providing FSAR Markup)
RAI 273 — 11.05-2	**August 16, 2011**
RAI 273 — 11.05-5	**August 16, 2011**
RAI 273 — 11.05-7	**August 16, 2011**
RAI 273 — 11.05-8	**August 16, 2011**
RAI 273 — 11.05-9	**August 16, 2011**
RAI 273 — 11.05-10	**August 16, 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, June 10, 2011 7:53 AM
To: 'Tesfaye, Getachew'
Cc: BENNETT Kathy (RS/NB); DELANO Karen (RS/NB); ROMINE Judy (RS/NB); RYAN Tom (RS/NB); NOXON David (RS/NB)
Subject: Response to U.S. EPR Design Certification Application RAI No. 273, FSAR Ch 11, Supplement 14

Getachew,

AREVA NP Inc. (AREVA NP) provided technically correct and complete responses to 10 of the 42 questions of RAI No. 273 on October 14, 2009. Supplement 1 to RAI No. 273 was sent on November 6, 2009 which responded to 17 of the 32 remaining questions and provided partial responses to 4 of the remaining 32. AREVA NP provided Supplement 2 on November 25, 2009 and Supplement 3 on December 8, 2009 to revise the commitment date for 11 of the remaining questions. AREVA NP provided Supplement 4 on December 10, 2009 to provide responses to 11 of the 15 remaining questions with a schedule to provide an FSAR markup for 9 of the questions. AREVA NP provided Supplement 5 on March 31, 2010 to provide a technically correct and complete response to 1 of the 4 remaining questions and provide FSAR markups for 3 of the 9 FSAR Markups. Supplement 6 to RAI No. 273 was sent on May 19, 2010 which responded to 1 of the 3 remaining questions. RAI 273 Supplement 7 sent on July 29, 2010, Supplement 8 sent on October 1, 2010, Supplement 9 sent on October 28, 2010, Supplement 10 sent on December 17, 2011, Supplement 11 sent on January 21, 2011 and Supplement 12 sent on March 3, 2011 provided a revised schedule for the remaining FSAR markups and responses. Supplement 13 sent on April 11, 2011 provided a technically correct and complete response to the 2 remaining questions and a revised schedule for the remaining 6 FSAR Markups.

The schedule for the FSAR markups for the 6 remaining questions is being revised as provided below:

Question #	Supplement Date (providing FSAR Markup)
RAI 273 — 11.05-2	August 9, 2011
RAI 273 — 11.05-5	August 9, 2011
RAI 273 — 11.05-7	August 9, 2011
RAI 273 — 11.05-8	August 9, 2011
RAI 273 — 11.05-9	August 9, 2011
RAI 273 — 11.05-10	August 9, 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 11, 2011 2:08 PM
To: 'Getachew Tesfaye'
Cc: Michael. Miernicki@nrc. gov (Michael.Miernicki@nrc.gov); NOXON David (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. 273, FSAR Ch 11, Supplement 13

Getachew,

AREVA NP Inc. (AREVA NP) provided technically correct and complete responses to 10 of the 42 questions of RAI No. 273 on October 14, 2009. Supplement 1 to RAI No. 273 was sent on November 6, 2009 which responded to 17 of the 32 remaining questions and provided partial responses to 4 of the remaining 32. AREVA NP provided Supplement 2 on November 25, 2009 and Supplement 3 on December 8, 2009 to revise the commitment date for 11 of the remaining questions. AREVA NP provided Supplement 4 on December 10, 2009 to provide responses to 11 of the 15 remaining questions with a schedule to provide an FSAR markup for 9 of the questions. AREVA NP provided Supplement 5 on March 31, 2010 to provide a technically correct and complete response to 1 of the 4 remaining questions and provide FSAR markups for 3 of the 9 FSAR Markups. Supplement 6 to RAI No. 273 was sent on May 19, 2009 which responded to 1 of the 3 remaining questions. RAI 273 Supplement 7 sent on July 29, 2010, Supplement 8 sent on October 1, 2010, Supplement 9 sent on October 28, 2010, Supplement 10 sent on December 17, 2011, Supplement 11 on January 21, 2011 and Supplement 12 on March 3, 2011 provided a revised schedule for the remaining FSAR markups and responses. The attached file, "RAI 273 Supplement 13 Response US EPR DC.pdf," provides a technically correct and complete response to 2 of the 2 remaining questions. A revised schedule is provided for the remaining 6 FSAR Markups.

The following table indicates the respective pages in the response document, "RAI 273 Supplement 13 Response US EPR DC.pdf," that contain AREVA NP's responses to the subject questions.

Question #	Start Page	End Page
RAI 273 — 11.02-14	2	2
RAI 273 — 11.03-12	3	3

The schedule for the FSAR markups for the 6 remaining questions is being revised to allow more time to interact with the NRC. The revised schedule for the remaining 6 FSAR markups is provided below:

Question #	Supplement Date (providing FSAR Markup)
RAI 273 — 11.05-2	June 10, 2011
RAI 273 — 11.05-5	June 10, 2011
RAI 273 — 11.05-7	June 10, 2011
RAI 273 — 11.05-8	June 10, 2011
RAI 273 — 11.05-9	June 10, 2011
RAI 273 — 11.05-10	June 10, 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: Thursday, March 03, 2011 9:11 AM
To: 'Tesfaye, Getachew'
Cc: DELANO Karen (RS/NB); ROMINE Judy (RS/NB); BENNETT Kathy (RS/NB); NOXON David (RS/NB)
Subject: Response to U.S. EPR Design Certification Application RAI No. 273, FSAR Ch 11, Supplement 12

Getachew,

AREVA NP Inc. (AREVA NP) provided technically correct and complete responses to 10 of the 42 questions of RAI No. 273 on October 14, 2009. Supplement 1 to RAI No. 273 was sent on November 6, 2009 which responded to 17 of the 32 remaining questions and provided partial responses to 4 of the remaining 32. AREVA NP provided Supplement 2 on November 25, 2009 and Supplement 3 on December 8, 2009 to revise the commitment date for 11 of the remaining questions. AREVA NP provided Supplement 4 on December 10, 2009 to provide responses to 11 of the 15 remaining questions with a schedule to provide an FSAR markup for 9 of the questions. AREVA NP provided Supplement 5 on March 31, 2010 to provide a technically correct and complete response to 1 of the 4 remaining questions and provide FSAR markups for 3 of the 9 FSAR Markups. Supplement 6 to RAI No. 273 was sent on May 19, 2009 which responded to 1 of the 3 remaining questions. RAI 273 Supplement 7 sent on July 29, 2010, Supplement 8 sent on October 1, 2010, Supplement 9 sent on October 28, 2010, Supplement 10 sent on December 17, 2011 and Supplement 11 on January 21, 2011provided a revised schedule for the remaining FSAR markups and responses.

The schedule for the FSAR markups for the 6 remaining questions is being revised to allow more time to interact with the NRC. The revised schedule for the remaining 6 FSAR markups is provided below:

Question #	Supplement Date (providing FSAR Markup)
RAI 273 — 11.05-2	April 14, 2011
RAI 273 — 11.05-5	April 14, 2011
RAI 273 — 11.05-7	April 14, 2011
RAI 273 — 11.05-8	April 14, 2011
RAI 273 — 11.05-9	April 14, 2011
RAI 273 — 11.05-10	April 14, 2011

The schedule for a technically correct and complete response to the remaining 2 questions is being revised to allow more time to interact with the NRC. The revised schedule for the responses to the remaining 2 questions is provided below:

Question #	Response Date
RAI 273 — 11.02-14	April 14, 2011
RAI 273 — 11.03-12	April 14, 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, January 21, 2011 8:23 AM
To: 'Tesfaye, Getachew'
Cc: DELANO Karen (RS/NB); ROMINE Judy (RS/NB); BENNETT Kathy (RS/NB); NOXON David (RS/NB)
Subject: Response to U.S. EPR Design Certification Application RAI No. 273, FSAR Ch 11, Supplement 11

Getachew,

AREVA NP Inc. (AREVA NP) provided technically correct and complete responses to 10 of the 42 questions of RAI No. 273 on October 14, 2009. Supplement 1 to RAI No. 273 was sent on November 6, 2009 which responded to 17 of the 32 remaining questions and provided partial responses to 4 of the remaining 32. AREVA NP provided Supplement 2 on November 25, 2009 and Supplement 3 on December 8, 2009 to revise the commitment date for 11 of the remaining questions. AREVA NP provided Supplement 4 on December 10, 2009 to provide responses to 11 of the 15 remaining questions with a schedule to provide an FSAR markup for 9 of the questions. AREVA NP provided Supplement 5 on March 31, 2010 to provide a technically correct and complete response to 1 of the 4 remaining questions and provide FSAR markups for 3 of the 9 FSAR Markups. Supplement 6 to RAI No. 273 was sent on May 19, 2009 which responded to 1 of the 3 remaining questions. RAI 273 Supplement 7 sent on July 29, 2010, Supplement 8 sent on October 1, 2010, Supplement 9 sent on October 28, 2010 and Supplement 10 sent on December 17, 2011 provided a revised schedule for the remaining FSAR markups and responses.

The schedule for the FSAR markups for the 6 remaining questions is being revised to allow more time to interact with the NRC. The revised schedule for the remaining 6 FSAR markups is provided below:

Question #	Supplement Date (providing FSAR Markup)
RAI 273 — 11.05-2	March 3, 2011
RAI 273 — 11.05-5	March 3, 2011
RAI 273 — 11.05-7	March 3, 2011
RAI 273 — 11.05-8	March 3, 2011
RAI 273 — 11.05-9	March 3, 2011
RAI 273 — 11.05-10	March 3, 2011

The schedule for a technically correct and complete response to the remaining 2 questions is being revised to allow more time to interact with the NRC. The revised schedule for the responses to the remaining 2 questions is provided below:

Question #	Response Date
RAI 273 — 11.02-14	March 3, 2011
RAI 273 — 11.03-12	March 3, 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, December 17, 2010 9:51 AM
To: Tesfaye, Getachew
Cc: DELANO Karen (RS/NB); ROMINE Judy (RS/NB); BENNETT Kathy (RS/NB); NOXON David (RS/NB); 'Carneal, Jason'
Subject: Response to U.S. EPR Design Certification Application RAI No. 273, FSAR Ch 11, Supplement 10

Getachew,

AREVA NP Inc. (AREVA NP) provided technically correct and complete responses to 10 of the 42 questions of RAI No. 273 on October 14, 2009. Supplement 1 to RAI No. 273 was sent on November 6, 2009 which responded to 17 of the 32 remaining questions and provided partial responses to 4 of the remaining 32. AREVA NP provided Supplement 2 on November 25, 2009 and Supplement 3 on December 8, 2009 to revise the commitment date for 11 of the remaining questions. AREVA NP provided Supplement 4 on December 10, 2009 to provide responses to 11 of the 15 remaining questions with a schedule to provide an FSAR markup for 9 of the questions. AREVA NP provided Supplement 5 on March 31, 2010 to provide a technically correct and complete response to 1 of the 4 remaining questions and provide FSAR markups for 3 of the 9 FSAR Markups. Supplement 6 to RAI No. 273 was sent on May 19, 2009 which responded to 1 of the 3 remaining questions. RAI 273 Supplement 7 sent on July 29, 2010, Supplement 8 sent on October 1, 2010 and Supplement 9 sent on October 28, 2010 provided a revised schedule for the remaining FSAR markups and responses.

The schedule for the FSAR markups for the 6 remaining questions is being revised to allow more time to interact with the NRC. The revised schedule for the remaining 6 FSAR markups is provided below:

Question #	Supplement Date (providing FSAR Markup)
RAI 273 — 11.05-2	January 21, 2011
RAI 273 — 11.05-5	January 21, 2011
RAI 273 — 11.05-7	January 21, 2011
RAI 273 — 11.05-8	January 21, 2011
RAI 273 — 11.05-9	January 21, 2011
RAI 273 — 11.05-10	January 21, 2011

The schedule for a technically correct and complete response to the remaining 2 questions is being revised to allow more time to interact with the NRC. The revised schedule for the responses to the remaining 2 questions is provided below:

Question #	Response Date
RAI 273 — 11.02-14	January 21, 2011
RAI 273 — 11.03-12	January 21, 2011

Sincerely,

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

From: BRYAN Martin (External RS/NB)
Sent: Thursday, October 28, 2010 2:23 PM
To: 'Tesfaye, Getachew'
Cc: DELANO Karen (RS/NB); ROMINE Judy (RS/NB); BENNETT Kathy (RS/NB); NOXON David (RS/NB)
Subject: Response to U.S. EPR Design Certification Application RAI No. 273, FSAR Ch 11, Supplement 9

Getachew,

AREVA NP Inc. (AREVA NP) provided technically correct and complete responses to 10 of the 42 questions of RAI No. 273 on October 14, 2009. Supplement 1 to RAI No. 273 was sent on November 6, 2009 which responded to 17 of the 32 remaining questions and provided partial responses to 4 of the remaining 32. AREVA NP provided Supplement 2 on November 25, 2009 and Supplement 3 on December 8, 2009 to revise the commitment date for 11 of the remaining questions. AREVA NP provided Supplement 4 on December 10, 2009 to provide responses to 11 of the 15 remaining questions with a schedule to provide an FSAR markup for 9 of the questions. AREVA NP provided Supplement 5 on March 31, 2010 to provide a technically correct and complete response to 1 of the 4 remaining questions and provide FSAR markups for 3 of the 9 FSAR Markups. Supplement 6 to RAI No. 273 was sent on May 19, 2009 which responded to 1 of the 3 remaining questions. RAI 273 Supplement 7 sent on July 29, 2010 and Supplement 8 sent on October 1, 2010 provided a revised schedule for the remaining FSAR markups and responses.

The schedule for the FSAR markups for the 6 remaining questions is being revised to allow more time to interact with the NRC. The revised schedule for the remaining 6 FSAR markups is provided below:

Question #	Supplement Date (providing FSAR Markup)
RAI 273 — 11.05-2	December 17, 2010
RAI 273 — 11.05-5	December 17, 2010
RAI 273 — 11.05-7	December 17, 2010
RAI 273 — 11.05-8	December 17, 2010
RAI 273 — 11.05-9	December 17, 2010
RAI 273 — 11.05-10	December 17, 2010

The schedule for a technically correct and complete response to the remaining 2 questions is being revised to allow more time to interact with the NRC. The revised schedule for the responses to the remaining 2 questions is provided below:

Question #	Response Date
RAI 273 — 11.02-14	December 17, 2010
RAI 273 — 11.03-12	December 17, 2010

Sincerely,

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

From: BRYAN Martin (External RS/NB)
Sent: Friday, October 01, 2010 3:20 PM
To: 'Tesfaye, Getachew'
Cc: DELANO Karen (RS/NB); ROMINE Judy (RS/NB); BENNETT Kathy (RS/NB); NOXON David (RS/NB)
Subject: Response to U.S. EPR Design Certification Application RAI No. 273, FSAR Ch 11, Supplement 8

Getachew,

AREVA NP Inc. (AREVA NP) provided technically correct and complete responses to 10 of the 42 questions of RAI No. 273 on October 14, 2009. Supplement 1 to RAI No. 273 was sent on November 6, 2009 which responded to 17 of the 32 remaining questions and provided partial responses to 4 of the remaining 32. AREVA NP provided Supplement 2 on November 25, 2009 and Supplement 3 on December 8, 2009 to revise the commitment date for 11 of the remaining questions. AREVA NP provided Supplement 4 on December 10, 2009 to provide responses to 11 of the 15 remaining questions with a schedule to provide an FSAR markup for 9 of the questions. AREVA NP provided Supplement 5 on March 31, 2010 to provide a technically correct and complete response to 1 of the 4 remaining questions and provide FSAR markups for 3 of the 9 FSAR Markups. Supplement 6 to RAI No. 273 was sent on May 19, 2009 which responded to 1 of the 3 remaining questions. Supplement 7 to RAI 273 was sent on July 29, 2010 to provide a revised schedule for the remaining FSAR markups and responses.

The schedule for the FSAR markups for the 6 remaining questions is being revised to allow more time to interact with the NRC. The revised schedule for the remaining 6 FSAR markups is provided below:

Question #	Supplement Date (providing FSAR Markup)
RAI 273 — 11.05-2	November 1, 2010
RAI 273 — 11.05-5	November 1, 2010
RAI 273 — 11.05-7	November 1, 2010
RAI 273 — 11.05-8	November 1, 2010
RAI 273 — 11.05-9	November 1, 2010
RAI 273 — 11.05-10	November 1, 2010

The schedule for a technically correct and complete response to the remaining 2 questions is being revised to allow more time to interact with the NRC. The revised schedule for the responses to the remaining 2 questions is provided below:

Question #	Response Date
RAI 273 — 11.02-14	November 1, 2010
RAI 273 — 11.03-12	November 1, 2010

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

From: BRYAN Martin (EXT)
Sent: Thursday, July 29, 2010 3:45 PM
To: 'Tesfaye, Getachew'
Cc: DELANO Karen V (AREVA NP INC); ROMINE Judy (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); NOXON David B (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 273, FSAR Ch 11, Supplement 7

Getachew,

AREVA NP Inc. (AREVA NP) provided technically correct and complete responses to 10 of the 42 questions of RAI No. 273 on October 14, 2009. Supplement 1 to RAI No. 273 was sent on November 6, 2009 which responded to 17 of the 32 remaining questions and provided partial responses to 4 of the remaining 32. AREVA NP provided Supplement 2 on November 25, 2009 and Supplement 3 on December 8, 2009 to revise the commitment date for 11 of the remaining questions. AREVA NP provided Supplement 4 on December 10, 2009 to provide responses to 11 of the 15 remaining questions with a schedule to provide an FSAR markup for 9 of the questions. AREVA NP provided Supplement 5 on March 31, 2010 to provide a technically correct and complete response to 1 of the 4 remaining questions and provide FSAR markups for 3 of the 9 FSAR Markups. Supplement 6 to RAI No. 273 was sent on May 19, 2009 which responded to 1 of the 3 remaining questions.

Based on the need for more time to incorporate NRC comments, a complete FSAR markup is not provided for 6 of the questions scheduled for July 29, 2010. The revised schedule for the remaining 6 FSAR markups is provided below:

Question #	Supplement Date (providing FSAR Markup)
RAI 273 — 11.05-2	October 1, 2010
RAI 273 — 11.05-5	October 1, 2010
RAI 273 — 11.05-7	October 1, 2010
RAI 273 — 11.05-8	October 1, 2010
RAI 273 — 11.05-9	October 1, 2010
RAI 273 — 11.05-10	October 1, 2010

The schedule for a technically correct and complete response to the remaining 2 questions is being revised based on the need for more time to incorporate NRC comments. The revised schedule for the responses to the remaining 2 questions is provided below:

Question #	Response Date
RAI 273 — 11.02-14	October 1, 2010
RAI 273 — 11.03-12	October 1, 2010

Sincerely,

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

From: BRYAN Martin (EXT)
Sent: Wednesday, May 19, 2010 3:08 PM
To: Tesfaye, Getachew
Cc: DELANO Karen V (AREVA NP INC); ROMINE Judy (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); NOXON David B (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 273, FSAR Ch 11, Supplement 6

Getachew,

AREVA NP Inc. (AREVA NP) provided technically correct and complete responses to 10 of the 42 questions of RAI No. 273 on October 14, 2009. Supplement 1 to RAI No. 273 was sent on November 6, 2009 which responded to 17 of the 32 remaining questions and provided partial responses to 4 of the remaining 32. AREVA NP provided Supplement 2 on November 25, 2009 and Supplement 3 on December 8, 2009 to revise the commitment date for 11 of the remaining questions. AREVA NP provided Supplement 4 on December 10, 2009 to provide responses to 11 of the 15 remaining questions with a schedule to provide an FSAR markup for 9 of the questions. AREVA NP provided Supplement 5 on March 31, 2010 to provide a technically correct and complete response to 1 of the 4 remaining questions and provide FSAR markups for 3 of the 9 FSAR Markups. The attached file, "RAI 273 Supplement 6 Response US EPR DC.pdf," provides a technically correct and complete response to 1 of the 3 remaining questions. A revised schedule is provided for the remaining 6 FSAR Markups.

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 273 Question 11.04 -15.

The following table indicates the respective pages in the response document, "RAI 273 Supplement 6 Response US EPR DC.pdf," that contain AREVA NP's responses to the subject questions.

Question #	Start Page	End Page
RAI 273 — 11.04-15	1	3

Based on the April 22 and 23, 2010 Chapter 11 audit, a complete FSAR markup is not provided for 6 of the questions as originally scheduled for May 19, 2010. The revised schedule for the remaining 6 FSAR markups is provided below:

Question #	Supplement Date (providing FSAR Markup)
RAI 273 — 11.05-2	July 29, 2010
RAI 273 — 11.05-5	July 29, 2010
RAI 273 — 11.05-7	July 29, 2010
RAI 273 — 11.05-8	July 29, 2010
RAI 273 — 11.05-9	July 29, 2010
RAI 273 — 11.05-10	July 29, 2010

The schedule for a technically correct and complete response to the remaining 2 questions is revised based on the April 22 and 23, 2010 Chapter 11 audit and is provided below.

Question #	Response Date
RAI 273 — 11.02-14	July 29, 2010
RAI 273 — 11.03-12	July 29, 2010

Sincerely,

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

From: BRYAN Martin (EXT)
Sent: Wednesday, March 31, 2010 4:36 PM
To: 'Tesfaye, Getachew'
Cc: DELANO Karen V (AREVA NP INC); ROMINE Judy (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); NOXON David B (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 273, FSAR Ch 11, Supplement 5

Getachew,

AREVA NP Inc. (AREVA NP) provided technically correct and complete responses to 10 of the 42 questions of RAI No. 273 on October 14, 2009. Supplement 1 to RAI No. 273 was sent on November 6, 2009 which responded to 17 of the 32 remaining questions and provided partial responses to 4 of the remaining 32. AREVA NP provided Supplement 2 on November 25, 2009 and Supplement 3 on December 8, 2009 to revise the commitment date for 11 of the remaining questions. AREVA NP provided Supplement 4 on December 10, 2009 to provide response to 11 of the 15 remaining questions with a schedule to provide an FSAR markup for 9 of the questions. The attached file, "RAI 273 Supplement 5 Response US EPR DC.pdf," provides a technically correct and complete response to 1 of the 4 remaining questions and provides FSAR markups for 3 of the 9 questions.

Appended to this file are affected pages of the U.S. EPR Final Safety Analysis Report in redline-strikeout format which support the responses to RAI 273 Questions 11.03 -13, 11.05-01, 11.05 -04, and 11.05-6.

The following table indicates the respective pages in the response document, "RAI 273 Supplement 5 Response US EPR DC.pdf," that contain AREVA NP's responses to the subject questions.

Question #	Start Page	End Page
RAI 273 — 11.03-13	2	3
RAI 273 — 11.05-1	4	4
RAI 273 — 11.05-4	5	5
RAI 273 — 11.05-6	6	6

Based on additional time needed to update engineering source documents, a complete FSAR markup is not provided for 6 of the questions as originally scheduled for March 31, 2010. The revised schedule for the remaining 6 FSAR markups is provided below:

Question #	Supplement Date (providing FSAR Markup)
RAI 273 — 11.05-2	May 19, 2010
RAI 273 — 11.05-5	May 19, 2010
RAI 273 — 11.05-7	May 19, 2010
RAI 273 — 11.05-8	May 19, 2010
RAI 273 — 11.05-9	May 19, 2010
RAI 273 — 11.05-10	May 19, 2010

The schedule for a technically correct and complete response to the remaining 3 questions is revised based on the March 24, 2010 Chapter 11 audit and is provided below.

Question #	Response Date
RAI 273 — 11.02-14	May 19, 2010
RAI 273 — 11.03-12	May 19, 2010
RAI 273 — 11.04-15	May 19, 2010

Sincerely,

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

From: Pederson Ronda M (AREVA NP INC)
Sent: Thursday, December 10, 2009 6:29 PM
To: 'Tesfaye, Getachew'
Cc: BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC); NOXON David B (AREVA NP INC); WILLIFORD Dennis C (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 273, FSAR Ch 11, Supplement 4

Getachew,

AREVA NP Inc. (AREVA NP) provided technically correct and complete responses to 10 of the 42 questions of RAI No. 273 on October 14, 2009. Supplement 1 to RAI No. 273 was sent on November 6, 2009 which responded to 17 of the 32 remaining questions and provided partial responses to 4 of the remaining 32. Supplements 2 and Supplement 3 on revised the commitment date for 11 of the remaining questions. The attached file, “RAI 273 Supplement 4 Response US EPR DC.pdf,” provides a technically correct and complete response to 11 of the 15 remaining questions, 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 273 Questions 11.05 -06, 11.06-07, 11.05 -08, 11.06-09, 11.05 -10, 11.06-11, and 11.06-12.

A complete FSAR markup is not provided for the RAI 273 questions. As agreed by NRC staff during an FSAR Chapter 11 audit on October 7, 2009, FSAR markups may be submitted after Phase 2 completion to support Staff review to close confirmatory items. Therefore, a complete FSAR markup for the RAI 273 questions will be provided as indicated in the following table:

Question #	Supplement Date (providing FSAR Markup)
RAI 273 — 11.05-1	March 31, 2010
RAI 273 — 11.05-2	March 31, 2010
RAI 273 — 11.05-4	March 31, 2010
RAI 273 — 11.05-5	March 31, 2010
RAI 273 — 11.05-6	March 31, 2010
RAI 273 — 11.05-7	March 31, 2010
RAI 273 — 11.05-8	March 31, 2010
RAI 273 — 11.05-9	March 31, 2010
RAI 273 — 11.05-10	March 31, 2010

The following table indicates the respective pages in the response document, “RAI 273 Supplement 4 Response US EPR DC.pdf,” that contain AREVA NP’s response to the subject question.

Question #	Start Page	End Page
RAI 273 — 11.05-1	2	2
RAI 273 — 11.05-2	3	6
RAI 273 — 11.05-4	7	8
RAI 273 — 11.05-5	9	16

RAI 273 — 11.05-6	17	19
RAI 273 — 11.05-7	20	21
RAI 273 — 11.05-8	22	23
RAI 273 — 11.05-9	24	26
RAI 273 — 11.05-10	27	27
RAI 273 — 11.05-11	28	30
RAI 273 — 11.05-12	31	31

The schedule for a technically correct and complete response to the remaining 4 questions is unchanged and provided below.

Question #	Response Date
RAI 273 — 11.02-14	March 31, 2010
RAI 273 — 11.03-12	March 31, 2010
RAI 273 — 11.03-13	March 31, 2010
RAI 273 — 11.04-15	March 31, 2010

Sincerely,

Ronda Pederson

ronda.pederson@areva.com
Licensing Manager, U.S. EPR Design Certification
AREVA NP Inc.
An AREVA and Siemens company
3315 Old Forest Road
Lynchburg, VA 24506-0935
Phone: 434-832-3694
Cell: 434-841-8788

From: Pederson Ronda M (AREVA NP INC)
Sent: Tuesday, December 08, 2009 6:22 PM
To: 'Tesfaye, Getachew'
Cc: BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC); NOXON David B (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 273, FSAR Ch 11, Supplement 3

Getachew,

AREVA NP is unable to provide a technically correct and complete response to the 11 remaining questions for RAI 273 today, as committed, and a revised schedule for a technically correct and complete response to the remaining questions is provided below.

Question #	Response Date
RAI 273 — 11.05-1	**December 11, 2009**
RAI 273 — 11.05-2	**December 11, 2009**
RAI 273 — 11.05-4	**December 11, 2009**
RAI 273 — 11.05-5	**December 11, 2009**
RAI 273 — 11.05-6	**December 11, 2009**
RAI 273 — 11.05-7	**December 11, 2009**

RAI 273 — 11.05-8	**December 11, 2009**
RAI 273 — 11.05-9	**December 11, 2009**
RAI 273 — 11.05-10	**December 11, 2009**
RAI 273 — 11.05-11	**December 11, 2009**
RAI 273 — 11.05-12	**December 11, 2009**
RAI 273 — 11.02-14	March 31, 2010
RAI 273 — 11.03-12	March 31, 2010
RAI 273 — 11.03-13	March 31, 2010
RAI 273 — 11.04-15	March 31, 2010

Sincerely,

Ronda Pederson

ronda.pederson@areva.com
Licensing Manager, U.S. EPR Design Certification
AREVA NP Inc.
An AREVA and Siemens company
3315 Old Forest Road
Lynchburg, VA 24506-0935
Phone: 434-832-3694
Cell: 434-841-8788

From: WELLS Russell D (AREVA NP INC)
Sent: Wednesday, November 25, 2009 2:38 PM
To: 'Getachew Tesfaye'; 'Michael Miernicki'
Cc: Pederson Ronda M (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 273, FSAR Ch 11, Supplement 2

Getachew,

AREVA NP Inc. (AREVA NP) provided technically correct and complete responses to 10 of the 42 questions of RAI No. 273 on October 14, 2009. AREVA NP submitted Supplement 1 on November 6, 2009 which provided a technically correct and complete response to 17 of the remaining questions and indicated that a response to 11 of the remaining 15 questions would be provided by November 25, 2009. However, AREVA NP is unable to provide a technically correct and complete response to 11 of the 15 remaining questions for RAI 273 as committed and a revised schedule for a technically correct and complete response to the remaining 15 questions is provided below.

Question #	Response Date
RAI 273 — 11.05-1	**December 8, 2009**
RAI 273 — 11.05-2	**December 8, 2009**
RAI 273 — 11.05-4	**December 8, 2009**
RAI 273 — 11.05-5	**December 8, 2009**
RAI 273 — 11.05-6	**December 8, 2009**
RAI 273 — 11.05-7	**December 8, 2009**
RAI 273 — 11.05-8	**December 8, 2009**
RAI 273 — 11.05-9	**December 8, 2009**
RAI 273 — 11.05-10	**December 8, 2009**
RAI 273 — 11.05-11	**December 8, 2009**
RAI 273 — 11.05-12	**December 8, 2009**
RAI 273 — 11.02-14	March 31, 2010

RAI 273 — 11.03-12	March 31, 2010
RAI 273 — 11.03-13	March 31, 2010
RAI 273 — 11.04-15	March 31, 2010

Sincerely,

(Russ Wells on behalf of)

Ronda Pederson

ronda.pederson@areva.com

Licensing Manager, U.S. EPR Design Certification

New Plants Deployment

AREVA NP, Inc.

An AREVA and Siemens company

3315 Old Forest Road

Lynchburg, VA 24506-0935

Phone: 434-832-3694

Cell: 434-841-8788

From: Pederson Ronda M (AREVA NP INC)
Sent: Friday, November 06, 2009 9:58 PM
To: 'Tesfaye, Getachew'
Cc: MCINTYRE Brian (AREVA NP INC); DELANO Karen V (AREVA NP INC); SLIVA Dana (AREVA NP INC); WILLIFORD Dennis C (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 273, FSAR Ch. 11, Supplement 1

Getachew,

AREVA NP Inc. (AREVA NP) provided technically correct and complete responses to 10 of the 42 questions of RAI No. 273 on October 14, 2009. The attached file, “RAI 273 Supplement 1 Response US EPR DC.pdf,” provides a technically correct and complete response to 17 of the remaining questions and partial responses to 4 of the remaining questions.

Appended to this file are the affected pages of the U.S. EPR Final Safety Analysis Report (FSAR) in redline-strikeout format which support the response to RAI 273 Question 11.02-4, 11.02-5, 11.02-6, 11.02-7, 11.02-8, 11.02-9, 11.02-12, 11.02-13, 11.02-15, 11.03-4, 11.03-5 , 11.03-8, 11.04-7, 11.04-8, 11.04-10, 11.04-14, 11.04-15, and 11.05-3.

Also included are related markups to AREVA NP's document, ANP-10292, Revision 1, "U.S. EPR Conformance with Standard Review Plan (NUREG-0800) Technical Report."

A complete FSAR markup is not provided for four of the answered questions. As agreed by NRC staff during an FSAR Chapter 11 audit on October 7, 2009, FSAR markups may be submitted after Phase 2 completion to support Staff review to close confirmatory items. Therefore, a complete FSAR markup for the four questions will be provided as indicated in the following table:

Question #	Supplement Date (providing FSAR Markup)
RAI 273 — 11.02-14	March 31, 2010
RAI 273 — 11.03-12	March 31, 2010
RAI 273 — 11.03-13	March 31, 2010
RAI 273 — 11.04-15	March 31, 2010

The following table indicates the respective page(s) in the response document, “RAI 273 Supplement 1 Response US EPR DC.pdf,” that contain AREVA NP’s response to the subject question.

Question #	Start Page	End Page
RAI 273 — 11.02-4	2	2
RAI 273 — 11.02-5	3	4
RAI 273 — 11.02-6	5	5
RAI 273 — 11.02-7	6	6
RAI 273 — 11.02-8	7	7
RAI 273 — 11.02-9	8	9
RAI 273 — 11.02-12	10	10
RAI 273 — 11.02-13	11	11
RAI 273 — 11.02-14	12	12
RAI 273 — 11.02-15	13	14
RAI 273 — 11.03-4	15	15
RAI 273 — 11.03-5	16	16
RAI 273 — 11.03-8	17	18
RAI 273 — 11.03-12	19	19
RAI 273 — 11.03-13	20	20
RAI 273 — 11.04-7	21	22
RAI 273 — 11.04-8	23	24
RAI 273 — 11.04-10	25	25
RAI 273 — 11.04-14	26	26
RAI 273 — 11.04-15	27	27
RAI 273 — 11.05-3	28	28

A complete answer is not provided for 15 of the 42 questions. The schedule for a technically correct and complete response to these questions is provided below.

Question #	Response Date
RAI 273 — 11.02-14	March 31, 2010
RAI 273 — 11.03-12	March 31, 2010
RAI 273 — 11.03-13	March 31, 2010
RAI 273 — 11.04-15	March 31, 2010
RAI 273 — 11.05-1	November 25, 2009
RAI 273 — 11.05-2	November 25, 2009
RAI 273 — 11.05-4	November 25, 2009
RAI 273 — 11.05-5	November 25, 2009
RAI 273 — 11.05-6	November 25, 2009
RAI 273 — 11.05-7	November 25, 2009
RAI 273 — 11.05-8	November 25, 2009
RAI 273 — 11.05-9	November 25, 2009
RAI 273 — 11.05-10	November 25, 2009
RAI 273 — 11.05-11	November 25, 2009
RAI 273 — 11.05-12	November 25, 2009

Sincerely,

Ronda Pederson
ronda.pederson@areva.com
Licensing Manager, U.S. EPR Design Certification

AREVA NP Inc.
An AREVA and Siemens company
3315 Old Forest Road
Lynchburg, VA 24506-0935
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Cell: 434-841-8788

From: Pederson Ronda M (AREVA NP INC)
Sent: Wednesday, October 14, 2009 5:45 PM
To: 'Tesfaye, Getachew'
Cc: BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC); WILLIFORD Dennis C (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 273, FSAR Ch. 11

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 273 Response US EPR DC.pdf" provides technically correct and complete responses to 10 of the 42 questions.

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 273, Questions 11.02-11, 11.03-7, 11.03-9, 11.03-10, 11.04-10, 11.04-11 and 11.04-12.

The following table indicates the respective pages in the response document, "RAI 273 Response US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 273 — 11.02-4	2	2
RAI 273 — 11.02-5	3	3
RAI 273 — 11.02-6	4	4
RAI 273 — 11.02-7	5	5
RAI 273 — 11.02-8	6	6
RAI 273 — 11.02-9	7	7
RAI 273 — 11.02-10	8	8
RAI 273 — 11.02-11	9	9
RAI 273 — 11.02-12	10	10
RAI 273 — 11.02-13	11	11
RAI 273 — 11.02-14	12	12
RAI 273 — 11.02-15	13	13
RAI 273 — 11.03-4	14	14
RAI 273 — 11.03-5	15	15
RAI 273 — 11.03-6	16	16
RAI 273 — 11.03-7	17	17
RAI 273 — 11.03-8	18	18
RAI 273 — 11.03-9	19	19
RAI 273 — 11.03-10	20	21
RAI 273 — 11.03-11	22	22
RAI 273 — 11.03-12	23	23

RAI 273 — 11.03-13	24	24
RAI 273 — 11.04-7	25	25
RAI 273 — 11.04-8	26	26
RAI 273 — 11.04-10	27	28
RAI 273 — 11.04-11	29	29
RAI 273 — 11.04-12	30	30
RAI 273 — 11.04-13	31	31
RAI 273 — 11.04-14	32	32
RAI 273 — 11.04-15	33	33
RAI 273 — 11.05-1	34	34
RAI 273 — 11.05-2	35	36
RAI 273 — 11.05-3	37	37
RAI 273 — 11.05-4	38	38
RAI 273 — 11.05-5	39	40
RAI 273 — 11.05-6	41	41
RAI 273 — 11.05-7	42	42
RAI 273 — 11.05-8	43	43
RAI 273 — 11.05-9	44	44
RAI 273 — 11.05-10	45	45
RAI 273 — 11.05-11	46	47
RAI 273 — 11.05-12	48	48

A complete answer is not provided for 32 of the 42 questions. The schedule for a technically correct and complete response to these questions is provided below.

Question #	Response Date
RAI 273 — 11.02-4	November 6, 2009
RAI 273 — 11.02-5	November 6, 2009
RAI 273 — 11.02-6	November 6, 2009
RAI 273 — 11.02-7	November 6, 2009
RAI 273 — 11.02-8	November 6, 2009
RAI 273 — 11.02-9	November 6, 2009
RAI 273 — 11.02-12	November 6, 2009
RAI 273 — 11.02-13	November 6, 2009
RAI 273 — 11.02-14	November 6, 2009
RAI 273 — 11.02-15	November 6, 2009
RAI 273 — 11.03-4	November 6, 2009
RAI 273 — 11.03-5	November 6, 2009
RAI 273 — 11.03-8	November 6, 2009
RAI 273 — 11.03-12	November 6, 2009
RAI 273 — 11.03-13	November 6, 2009
RAI 273 — 11.04-7	November 6, 2009
RAI 273 — 11.04-8	November 6, 2009
RAI 273 — 11.04-10 (Part 3)	November 6, 2009
RAI 273 — 11.04-14	November 6, 2009
RAI 273 — 11.04-15	November 6, 2009
RAI 273 — 11.05-1	November 6, 2009
RAI 273 — 11.05-2	November 6, 2009

RAI 273 — 11.05-3	November 6, 2009
RAI 273 — 11.05-4	November 6, 2009
RAI 273 — 11.05-5	November 6, 2009
RAI 273 — 11.05-6	November 6, 2009
RAI 273 — 11.05-7	November 6, 2009
RAI 273 — 11.05-8	November 6, 2009
RAI 273 — 11.05-9	November 6, 2009
RAI 273 — 11.05-10	November 6, 2009
RAI 273 — 11.05-11	November 6, 2009
RAI 273 — 11.05-12	November 6, 2009

Sincerely,

Ronda Pederson

ronda.pederson@areva.com
Licensing Manager, U.S. EPR Design Certification
AREVA NP Inc.
An AREVA and Siemens company
3315 Old Forest Road
Lynchburg, VA 24506-0935
Phone: 434-832-3694
Cell: 434-841-8788

From: Tesfaye, Getachew [mailto:Getachew.Tesfaye@nrc.gov]
Sent: Monday, September 14, 2009 3:12 PM
To: ZZ-DL-A-USEPR-DL
Cc: Dehmel, Jean-Claude; Frye, Timothy; Jennings, Jason; Colaccino, Joseph; ArevaEPRDCPEm Resource
Subject: U.S. EPR Design Certification Application RAI No. 273 (3450, 3459,3460, 3462), FSAR Ch. 11

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on August 11, 2009, and discussed with your staff on August 25, 2009. Draft RAI Question 11.04-9 was deleted, and Draft RAI Questions 11.02-4, 11.02-14, 11.03-4, 11.03-12, 11.03-13, 11.04-7, 11.04-12, 11.05-1, 11.05-4, and 11.05-5 were modified as a result of that discussion. The schedule we have established for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs. For any RAIs that cannot be answered within 30 days, it is expected that a date for receipt of this information will be provided to the staff within the 30 day period so that the staff can assess how this information will impact the published schedule.

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

Hearing Identifier: AREVA_EPR_DC_RAIs
Email Number: 3315

Mail Envelope Properties (2FBE1051AEB2E748A0F98DF9EEE5A5D482D624)

Subject: Response to U.S. EPR Design Certification Application RAI No. 273, FSAR Ch 11, Supplement 16
Sent Date: 8/10/2011 4:36:27 PM
Received Date: 8/10/2011 4:37:00 PM
From: WILLIFORD Dennis (AREVA)

Created By: Dennis.Williford@areva.com

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

Post Office: auscharmx02.adom.ad.corp

Files	**Size**	**Date & Time**
MESSAGE	48680	8/10/2011 4:37:00 PM
RAI 273 Supplement 16 Response US EPR DC.pdf		1749119

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

Response to

Request for Additional Information No. 273 (3450, 3459, 3460, 3462), Revision 0, Supplement 16

9/14/2009

U.S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 11.02 - Liquid Waste Management System
SRP Section: 11.03 - Gaseous Waste Management System
SRP Section: 11.04 - Solid Waste Management System
SRP Section: 11.05 - Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems

Application Section: Chapter 11

QUESTIONS for Health Physics Branch (CHPB)

Question 11.05-2:

In Sections 11.5.2, 11.5.3 and 11.5.4, the FSAR describes the various functions and components of PERMSS subsystems. A review of the description indicates that the information is presented inconsistently. Several components described in the text and Table 11.5-1 and Figure 11.5-1 are discussed in the text and not shown in a corresponding figure, or are not discussed in the text but shown in a figure. For example:

1) The descriptions do not address subsystem features to perform functional channel checks, source checks, and whether subsystems need to be brought off line to perform the surveillance and functional checks required by standard radiological effluent controls (see GL 89-01 and NUREG-1301).

2) The design features do not describe provisions and methods for offline radiation detectors to be purged or flushed with clean air or water, which portions of the LWMS and GWMS systems will contaminated purge or flush fluids be returned to, and provisions to prevent the cross-contamination of purge and flush supply systems.

3) The design features do not describe the calibration of subsystems under different operational conditions (e.g., routine, AOOs, and accident) given expected differences in radionuclide distributions; how raw instrument output data will be converted to meaningful radiological units in determining compliance with Part 20 Appendix B and Appendix I to Part 50; calibration frequencies of process and effluent of radiation monitoring subsystems and associated sampling equipment; and the QA/QC process that will be used to verify and validate computer codes and algorithm purchased through vendors or supplied with PERM instrumentation.

4) Subsystem figures and diagrams do not show sampling locations for liquid and gaseous process and effluent streams, and sample conditioning for specific systems to minimize sample loss and distortion of chemical and physical compositions. Also, the relationship with FSAR Section 9.3.2 is not well established in understanding shared used of equipment and sampling functions among LWMS and GWMS subsystems.

5) The design features do not address requirements for the derivation of lower limits of detection for liquid and gaseous effluent monitors and detection sensitivities for liquid and gaseous process monitors.

6) The design features do not address the placement of isolation or diversion valves and radiation detectors on process and effluent piping/ductwork in order to ensure the timely closure of such valves upon the detection of elevated radioactivity levels in liquid and gaseous effluent streams in terminating releases and isolating process streams from further contamination.

7) The design features of the liquid and gaseous monitoring systems do not consider how the failure of a radiation detector or its corresponding channel is indicated in local and control room panels. For example, does a detector or channel failure result in a high or zero instrumentation reading? Are there specific warnings differentiating various types of failures, such as loss of detector only, loss of an entire channel, loss of sampling flow from a process or effluent stream?

Collectively, these observations apply to nearly all PERMSS subsystem descriptions and should be reviewed against Section 11.5 of the SRP and Regulatory Guide 1.206 and corrected accordingly in FSAR subsystem descriptions and tables and figures.

Response to Question 11.05-2:

This question was addressed in the Response to RAI 273, Supplement 4.

Based on the Chapter 11 audit held April 6, 2011 and additional NRC staff comments provided via e-mail, additional changes to the U.S. EPR FSAR were identified for consistency with US EPR FSAR Tier 2 Table 11.5-1.

FSAR Impact:

U.S. EPR FSAR Tier 1 Section 2.6, Section 2.8, Section 2.9, Tier 2, Section 6.2, Section 6.4, Section 6.5, Section 7.5, Section 9.2, Section 9.3, Section 9.4, Section 10.3, Section 10.4, Section 11.2, Section 11.3, Section 11.4, Section 11.5, Section 12.3, Section 14.2, and Table 11.5-1 will be revised as described in this response and the response to this question provided in RAI 273, Supplement 4 and indicated on the enclosed markup.

Question 11.05-5:

FSAR Sections 11.5.2 to 11.5.4 present the descriptions of PERMSS subsystems and Table 11.5-1 lists radiation monitoring instrumentation used to monitor process and effluent streams. However, a review of subsystems listed in Table 11.5-1 and Figure 11.5-1 indicates that the descriptions are inconsistent and incomplete. For example:

1) Several PERMSS subsystems listed in Table 11.5-1 are not described in FSAR Sections 11.5.2 to 11.5.4. Section 11.5.4 provides descriptions for six subsystems with in-process radiation detectors while Table 11.5-1 lists 15. Similarly, there are differences between the text and tables in nomenclatures used to describe subsystems, e.g., condenser air removal RMS (Section 11.5.4.2) vs condenser evacuation system (Table 11.5-1).
2) FSAR Table 11.5-1 does not differentiate between safety and non safety-related PERMSS subsystems.
3) FSAR Table 11.5-1 does not indicate which radiation detectors will be equipped with built-in radioactive check sources for the purpose of performing channel checks.
4) The operational range noted for the upstream radiation monitor of the GWMS delay beds is presented in a meaningless radiological unit. The range should be expressed in μCi/cc instead of counts per second (cps).
5) FSAR Section 10.4.3.2.2.states that the exhaust from the turbine gland seal exhausters is routed to the "turbine building air removal system" where they are monitored for radioactivity. Table 11.5-1 does not list a radiation monitor for the turbine air removal system. FSAR Section 9.4.4 states that the Turbine Building does not include the means to control radioactive effluents. Accordingly, the design descriptions of Section 10.4.3 and 9.4.4 should be reviewed and corrected to indicate that there is no turbine building air removal system and that instead the exhaust from the turbine gland seal exhausters is a system exhaust that is directed and discharged via the Nuclear Auxiliary Building Exhaust.
6) For the Nuclear Auxiliary Building exhaust, Figure 9.3.4-2 shows a radiation detector on the ductwork going into Cell 1 of the NABVS but it is not clear if this radiation detector is an extra one or is it the same as that shown in Figure 11.3-1 given its location.
7) Figure 9.3.4-2 shows a radiation detector in Cell 3 of the NABVS but it is not clear if this radiation detector is part of an existing PERMSS subsystem and not described in FSAR Sections 11.5 and 9.4.3, or is part of another system, such as used for monitoring ambient airborne radioactivity levels in radiological controlled areas.
8) Figure 9.4.3-3 does not include a radiation detector in the Reactor Building Exhaust (last input on lower left side of drawing).
9) FSAR Table 11.5-1 implies that there are two continuous noble gas monitors for the containment building ventilation purge subsystem, but FSAR Figure 9.4.7-2 shows one monitor on the low flow purge exhaust and none for the full flow purge exhaust.
10) FSAR Figure 11.5-1 shows a radiation monitor on the Turbine Building Plant drainage line, but this monitor is not listed in FSAR Table 11.5-1.

11) FSAR Table 11.5-1 does not indicate any automatic control features for GWMS, but FSAR Section 11.3.3.1 states that discharge requirements consider gaseous waste activity and "automatic isolation settings."

12) FSAR Table 11.5-1 does not indicate any automatic control features for the NABVS, but FSAR Section 9.4.3.2.1 states that the exhaust from the NABVS is diverted to an iodine filtration system upon receiving a radiation alarm.

13) FSAR Table 11.5-1 does not indicate any automatic control features (ACF) for the SBVS, but FSAR Section 9.4.5.1 states that the exhaust from the SBVS is diverted to charcoal filtration beds upon receiving a radiation alarm.

14) FSAR Table 11.5-1 presents operational ranges for gaseous and liquid process and effluent radiation monitors. A review of this information indicates that it is incomplete. For gaseous process and effluent monitors, the supporting text to this table does not present the technical basis for the stated operational ranges for those that are included and does not provide corresponding sets of values for the plant vent radiation monitor used in confirming compliance with Part 20, Appendix B effluent concentration limits. For particulate, iodine, and tritium expected in gaseous streams, Table 11.5-1 does not include operational ranges and surrogate radionuclides for the corresponding radiation monitoring systems. The above comments also apply to radiation monitors assigned to liquid process and effluent streams.

15) FSAR Table 11.5-1 presents operational ranges for gaseous process and effluent radiation monitors. A review of this information indicates that it is incomplete. The description does not indicate whether the plant vent radiation monitoring system will be used to monitor radioactivity levels during normal operations and accident conditions with a sufficient range to encompass the entire range of effluent concentration levels listed in Regulatory Guide (RG) 1.97 (Rev. 3) and BTP 7-10 (SRP, NUREG-0800) for Type E variables, as they relate to compliance with Part 50.34(f)(2)(xvii) and (xxvii). If multiple radiation monitoring components are part of the design of the plant vent monitor to comply with the operational ranges of RG 1.97, the FSAR should describe the additional radiation monitoring components and address overall system accuracies in the overlapping ranges of the components of both systems when operating in the upper end of expected radioactivity levels.

Collectively, the above observations should be reviewed and, if confirmed, should be corrected in the FSAR.

Response to Question 11.05-5:

This question was addressed in the Response to RAI 273, Supplement 4. There are no changes to U.S. EPR FSAR Tier 2 Tables 3.10-1 and 3.11-1 required for consistency with the Chapter 11 changes.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 9.4.3.2.1, Section 10.4.3.2.2, Section 11.5.3, Section 11.5.4, Table 3.2.2-1, and Table 11.5-1 will be revised as described in this response and the response to RAI 273, Supplement 4 and indicated on the enclosed markup.

Question 11.05-7:

FSAR Sections 11.5.2 to 11.5.4 present the descriptions of PERMSS subsystems and Table 11.5-1 lists radiation monitoring instrumentation used to monitor airborne effluent streams from the Nuclear Auxiliary Building Ventilation System (NABVS), as described in FSAR Section 9.4.3. A review of subsystems listed in Table 11.5-1 and Figures 11.5-1 and 9.4.3-2 to 9.4.3-4 indicates that the descriptions are inconsistent and incomplete. Specifically:

a. Table 11.5-1 does not identify the Automatic Control Features (ACF) of the radiation monitor for the NABVS radiation monitoring system. A review of FSAR Section 9.4.3 indicates that if elevated radiation levels are detected, the NABS exhaust flow is diverted to the iodine filtration train prior to discharge via the plant vent. Accordingly, the ACF provisions of Table 11.5-1 for the NABVS should be reviewed and revised to note whether the isolation of NABS Cells 1, 2 and 3 is part of the ACF design features for that radiation monitor.

b. FSAR Section 9.4.3.5 provides information on instrumentation requirements. However, Section 9.4.3.5 does not refer to FSAR Section 11.5 and Table 11.5-1 for specific details on the associated radiation monitoring equipment. Accordingly, FSAR Sections 11.5 and 9.4.3 should be reviewed and revised to ensure a consistent use of internal references on radiation instrumentation design features in controlling airborne radioactivity releases via the plant vent.

c. In support to FSAR Table 11.5-1, FSAR Section 11.5.3.1 should list all radiological exhaust ventilation systems supported by radiation monitoring instrumentation and sampling systems. Accordingly, FSAR Sections 9.3, 9.4, 11.2, 11.3, 11.4, and 11.5 should be reviewed and revised to ensure a complete and consistent presentation of all systems serviced by radiation instrumentation in controlling and monitoring airborne radioactivity releases via the plant vent.

Response to Question 11.05-7:

This question was addressed in the Response to RAI 273, Supplement 4.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 9.3, Section 9.4, Section 11.2, Section 11.3, Section 11.4, Section 11.5, Table 11.5-1, and Figure 9.4.3-3 will be revised as described in the response to this question provided in RAI 273, Supplement 4 and indicated on the enclosed markup.

Question 11.05-8:

FSAR Sections 11.5.2 to 11.5.4 present the descriptions of PERMSS subsystems and Table 11.5-1 lists radiation monitoring instrumentation used to monitor airborne effluent streams from the Safeguard Building Controlled-Area Ventilation System (SBVS), as described in FSAR Section 9.4.5. A review of subsystems listed in Table 11.5-1 and Figures 11.5-1 and 9.4.5-2 indicates that the descriptions are inconsistent and incomplete. Specifically:

a. Table 11.5-1 does not identify the Automatic Control Features (ACF) of the radiation monitor for the SBVS radiation monitoring system. A review of FSAR Section 9.4.5 indicates that if elevated radiation levels are detected, the SBVS exhaust flow is diverted to the iodine filtration train located in the Fuel Building prior to discharge via the plant vent. Sections 9.4.5.1 and 9.4.5.2.3 also state that in the event of an accident in the Fuel Building or Reactor Building, the exhaust flows from these systems are diverted to the iodine train of the SBVS. However, it is not clear if all stated isolation and diversion functions of exhaust flows from these systems are automatic or whether some require manual operation for the described abnormal operating conditions and accidents. Accordingly, the ACF provisions of Table 11.5-1 for the SBVS should be reviewed and revised to distinguish the automatic isolation of SBVS dampers in directing exhaust to the SBVS iodine filtration train, and isolation features (automatic or manual as the case may be) for abnormal operations and accident conditions occurring in the FB and RB but which depend on the design features of the SBVS.

b. Although FSAR Table 11.5-1 identifies the use of a multi-function process radiation monitor, its location could not be readily determined in FSAR Figure 9.4.5-2. Accordingly, Figure 9.4.5-2 should be reviewed to confirm whether its location is indicated and, if not, it should be added to the figure to ensure a complete representation of the system.

c. FSAR Section 9.4.5.5 provides information on instrumentation requirements. However, Section 9.4.5.5 does not refer to FSAR Section 11.5 and Table 11.5-1 for specific details on the associated radiation monitoring equipment. Accordingly, FSAR Sections 11.5 and 9.4.5 should be reviewed and revised to ensure a consistent use of internal references on radiation instrumentation design features in controlling airborne radioactivity releases via the plant vent.

Response to Question 11.05-8:

This question was addressed in the Response to RAI 273, Supplement 4.

FSAR Impact:

U.S. EPR FSAR Tier 2, Table 11.5-1 will be revised as described in the response to this question provided in RAI 273, Supplement 4 and indicated on the enclosed markup.

Question 11.05-9:

FSAR Sections 11.5.2 to 11.5.4 present the descriptions of PERMSS subsystems and Table 11.5-1 lists radiation monitoring instrumentation used to monitor airborne effluent streams from the Radioactive Waste Building Ventilation System (RWBVS), as described in FSAR Section 9.4.8. A review of subsystems listed in Table 11.5-1 and Figures 11.5-1 and 9.4.8-1 and 9.4.8-2 indicates that the descriptions are inconsistent and incomplete. Specifically:

a. In FSAR Table 11.5-1, the entry on Automatic Control Features (ACF) for the RWBVS radiation monitoring system should state "n/a" or "none" as opposed to the ambiguous entry shown as "--- ". In addition, there is a need to ensure a consistent use of system nomenclature between figures and descriptions. For example, FSAR Sections 11.5 and 9.4.8 do not refer to RWBVS Cells 1 and 2 exhausts, while Figure 9.4.8-2 makes a distinction in differentiating sources with different radioactivity levels. Accordingly, FSAR Sections 11.5 and 9.4.8 should be reviewed and revised to ensure a consistent presentation of design features and nomenclatures used in demonstrating compliance with effluent concentration limits of Appendix B to Part 20.

b. Based on a review of FSAR Sections 11.5 and 9.4.8 figures and descriptions, it is not clear if a radiation detector is missing or one needs to be relocated in Figure 9.4.8-2 on the line coming from the RWBVS Processing Rooms (Line C from Sheet 1 to Sheet 2). In contrast, the corresponding lines from RWBVS Cells 1 and 2 show a radiation detector for Lines A and B from Sheet 1 to Sheet 2 before the filter trains. Accordingly, FSAR Sections 11.5 and 9.4.8 should be reviewed and revised to ensure a consistent presentation of design and operational features in understanding how RWBVS radiation monitors function and alert operators when effluent releases could exceed the effluent concentration limits of Appendix B to Part 20.

c. In FSAR Table 11.5-1, the entry for the RWBVS radiation monitoring system indicates that there are two iodine radiation monitors and four aerosol radiation monitors. A review of FSAR Figures 11.5-1 and 9.4.8-2 indicates that the placements of these monitors is not clear given that there are three input flows to the RWBVS exhaust, one for Cell 1 and one for Cell 2 (Lines A and B from Sheet 1 to Sheet 2), and one for the RWBVS Processing Rooms (Line C from Sheet 1 to Sheet 2). There is no rationale provided for having one set of radiation monitors before each particulate/charcoal train for Cells 1 and 2 vent exhausts and none after it, and no explanations for the placement of the Processing Rooms monitor after the particulate/charcoal train with none before it. Accordingly, FSAR Sections 11.5 and 9.4.8 should be reviewed and revised to provide the technical rationale for the placement of radiation monitoring instrumentation in ensuring that effluent releases do not exceed the effluent concentration limits of Appendix B to Part 20.

d. FSAR Section 9.4.8.5 provides information on instrumentation requirements. However, Section 9.4.8.5 does not refer to FSAR Section 11.5 and Table 11.5-1 for specific details on the associated radiation monitoring equipment. Accordingly, FSAR Sections 11.5 and 9.4.8 should be reviewed and revised to ensure a consistent use of internal references on radiation instrumentation design features in controlling airborne radioactivity releases via the plant vent.

Response to Question 11.05-9:

This question was addressed in the Response to RAI 273, Supplement 4.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 9.4.8, Table 11.5-1, and Figure 9.4.8-2 will be revised as described in the response to this question provided in RAI 273, Supplement 4 and indicated on the enclosed markup.

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Question 11.05-10:

FSAR Sections 11.5.2 to 11.5.4 present the descriptions of PERMSS subsystems and Table 11.5-1 lists radiation monitoring instrumentation used to monitor airborne effluent streams from the Access Building Ventilation System (ABVS), as described in FSAR Section 9.4.14. A review of subsystems listed in Table 11.5-1 and Figures 11.5-1 and 9.4.14-2 indicates that the descriptions are inconsistent and incomplete.

a. FSAR Table 11.5-1 identifies only a sampling system for the ABVS, while FSAR Figure 9.4.14-2 identifies a radiation monitor instead before the filtration train. Accordingly, FSAR Sections 11.5 and 9.4.14 should be reviewed and revised to ensure a consistent description of radiation instrumentation and design features used in controlling airborne radioactivity releases via the plant vent.

b. FSAR Section 9.4.14.6 provides information on instrumentation requirements for the ABVS. However, Section 9.4.14.6 does not refer to FSAR Section 11.5 and Table 11.5-1 for specific details on the associated radiation monitoring equipment for the ABVS. Accordingly, FSAR Sections 11.5 and 9.4.14 should be reviewed and revised to ensure a consistent use of internal references on radiation instrumentation design features in controlling airborne radioactivity releases via the plant vent.

Response to Question 11.05-10:

This question was addressed in the Response to RAI 273, Supplement 4.

FSAR Impact:

U.S. EPR FSAR Tier 2, Figure 9.4.14-2 and Table 11.5-1 will be revised as described in the response to this question provided in RAI 273, Supplement 4 and indicated on the enclosed markup.

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2.2 The location of the CRACS equipment is as listed in Table 2.6.1-1—Main Control Room Air Conditioning System Equipment Mechanical Design.

2.3 Physical separation exists between the CRACS air intake, iodine filtration, air recirculation, and air conditioning trains.

3.0 Mechanical Design Features

3.1 Deleted.

3.2 Equipment listed in Table 2.6.1-1 can perform the function listed in Table 2.6.1-1 under system operating conditions.

3.3 Components identified as Seismic Category I in Table 2.6.1-1 can withstand seismic design basis loads without a loss of the function listed in Table 2.6.1-1.

3.4 Components listed in Table 2.6.1-1 as ASME AG-1 Code are designed in accordance with ASME AG-1 Code requirements.

3.5 Components listed in Table 2.6.1-1 as ASME AG-1 Code are fabricated in accordance with ASME AG-1 Code requirements, including welding requirements.

3.6 Components listed in Table 2.6.1-1 as ASME AG-1 Code are inspected and tested in accordance with ASME AG-1 Code requirements.

4.0 Displays and Controls

4.1 Displays listed in Table 2.6.1-2—Main Control Room Air Conditioning System Equipment I&C and Electrical Design, are retrievable in the main control room (MCR) and the remote shutdown station (RSS) as listed in Table 2.6.1-2.

4.2 The CRACS equipment controls are provided in the MCR and RSS as listed in Table 2.6.1-2.

4.3 Equipment listed as being controlled by a priority and actuator control system (PACS) module in Table 2.6.1-2 responds to the state requested by a test signal.

5.0 Electrical Power Design Features

5.1 The equipment designated as Class 1E in Table 2.6.1-2 are powered from the Class 1E division as listed in Table 2.6.1-2 in a normal or alternate feed condition.

5.2 Deleted.

6.0 Equipment and System Performance

6.1 The CRACS maintains a positive pressure in the CRE area~~s~~ relative to the outside environment and adjacent areas, while operating in a design basis accident alignment.

6.2 Upon receipt of a containment isolation signal (CIS), ~~or high radiation alarm signal in the air intake duct,~~ the iodine filtration train will start automatically, outside air supply to the CRE area is diverted through the iodine filtration train, ~~and~~ a minimum recirculation

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Table 2.6.1-3—Main Control Room Air Conditioning System ITAAC (6 Sheets)

Commitment Wording		Inspections, Tests, Analyses	Acceptance Criteria
6.2	Upon receipt of a containment isolation signal (CIS), or high radiation alarm signal in the air intake duct, the iodine filtration train will start automatically, outside air supply to the CRE area is diverted through the iodine filtration train, and a minimum recirculation flowrate is established from the CRE area to the iodine filtration train and a positive pressure is maintained in the CRE area relative to the adjacent areas.	a. A test will be performed to verify, that upon receipt of a containment isolation CIS test signal, that the iodine filtration train will start automatically; and the outside air supply to the CRE area is diverted through the iodine filtration train. A Ttest iswill be performed separately for each iodine filtration train. b. A test will be performed to verify that upon receipt of high radiation alarm signal in the air intake duct, the iodine filtration train will start automatically; and the outside air supply is diverted through the iodine filtration train. Test is performed separately for each iodine filtration train. cb. A test will be performed to verify, upon receipt of a containment isolation test signal, that a minimum recirculation flowrate is established from the CRE area to for each the iodine filtration train is achieved. A test will be performed separately for each iodine filtration train. c. A test will be performed to verify, upon receipt of a containment isolation test signal, that the CRACS maintains a positive pressure in the CRE area relative to the adjacent areas.	a. A separate test for each iodine filtration train confirms, that upon receipt of a containment isolation CIS test signal, that the iodine filtration train will start automatically within 60 seconds; and the outside air supply to the CRE area is diverted through the iodine filtration train. b. A separate test for each iodine filtration train confirms that upon receipt of high radiation alarm signal in the air intake duct, the iodine filtration train will start automatically within 60 seconds; and the outside air supply is diverted through the iodine filtration train. cb. A separate test for each iodine filtration train confirms, upon receipt of a containment isolation test signal, that a CRE recirculation flowrate of greater than or equal to 3000 scfm. is established from the CRE area to the iodine filtration train. c. A test confirms, upon receipt of a containment isolation test signal, that the CRACS maintains the pressure greater than or equal to 0.125 inches water gauge in the CRE area relative to the adjacent areas.

Table 2.6.1-3—Main Control Room Air Conditioning System ITAAC (6 Sheets)

	Commitment Wording	Inspections, Tests, Analyses	Acceptance Criteria
6.7	Upon receipt of a high radiation alarm signal in the air intake duct, the iodine filtration train will start automatically, the outside air supply to the CRE area is diverted through the iodine filtration train, a minimum CRE recirculation flowrate is established from the CRE area to the iodine filtration train, and a positive pressure is maintained in the CRE area relative to the adjacent areas.	a. A test will be performed to verify, upon receipt of high radiation alarm test signal in the air intake duct, that the iodine filtration train will start automatically; and the outside air supply to the CRE area is diverted through the iodine filtration train. A test will be performed separately for each iodine filtration train. b. A test will be performed to verify, upon receipt of high radiation alarm test signal in the air intake duct, that a minimum CRE recirculation flowrate for each iodine filtration train is achieved. A test will be performed separately for each iodine filtration train. c. A test will be performed to verify, upon receipt of high radiation alarm test signal in the air intake duct, that a positive pressure is maintained in the CRE area relative to the adjacent areas.	a. A separate test for each iodine filtration train confirms, upon receipt of high radiation alarm test signal in the air intake duct (KLK65CR001/002 and KLK66CR001/002), that the iodine filtration train will start automatically within 60 seconds, and the outside air supply is diverted through the iodine filtration train. b. A separate test for each iodine filtration train confirms, upon receipt of high radiation alarm test signal in the air intake duct, that a CRE recirculation flowrate of greater than or equal to 3,000 scfm is established from the CRE area to the iodine filtration train. c. A test confirms, upon receipt of high radiation alarm test signal in the air intake duct, that a positive pressure of greater than or equal to 0.125 inches water gauge is maintained in the CRE area relative to the adjacent areas.

2.6.4 Fuel Building Ventilation System

1.0 Description

The fuel building ventilation system (FBVS) receives the conditioned air supply from the nuclear auxiliary building ventilation system (NABVS). The exhaust from the FBVS is processed by the NABVS through a filtration train, and the exhaust air is directed to the vent stack.

The FBVS controls the Fuel Building temperature, humidity and air change rate for personnel comfort, personnel safety, and equipment protection during normal plant operation. The FBVS provides cooling, heating, and ventilation for the Fuel Building (FB) to remove equipment heat and heat generated from other sources. The FBVS also provides heat to maintain a minimum temperature in the building. The FBVS provides a minimal air change rate for the building and controls the building pressurization to reduce spreading of contamination.

The FBVS provides the following safety-related functions:

- Isolation of the supply and exhaust airflow of the fuel handling hall.
- Isolation of the supply and exhaust airflow of the hall in front of equipment hatch.
- Isolation of the supply and exhaust airflow to the room in front of the emergency air lock.

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- Isolation of the FB from NABVS supply and exhaust on receipt of containment isolation signal ~~or high radiation signal in the Reactor Building~~. The FB atmosphere is then processed through iodine filtration trains of the safeguard building controlled-area ventilation system (~~SBCAVS~~SBVS).
- Heating of the rooms which have safety-related systems, structures, or components containing borated fluid and the rooms surrounding the extra borating system tanks to maintain minimum ambient room temperatures.
- Cooling of rooms which have the extra borating system pumps and the fuel pool cooling system pumps to maintain ambient conditions.

The FBVS provides the following non-safety related functions:

- Maintains the room ambient conditions for operation of equipment and to allow personnel access during normal operation.
- Reduces spread of contamination from the contaminated rooms to less contaminated rooms during normal operation.
- Reduces concentration of aerosols and radioactive gases from the room air.
- Maintains a negative pressure within the Fuel Building with respect to outside atmosphere.

6.0 Environmental Qualifications

6.1 Components in Table 2.6.4-2, that are designated as harsh environment, will perform the function listed in Table 2.6.4-1 in the environments that exist during and following design basis events.

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7.0 Equipment and System Performance

7.1 Upon receipt of a containment isolation signal (CIS), Tthe FBVS maintains a negative pressure relative to the outside environment in the Fuel Building upon receipt of a containment isolation signal.during normal operation.

7.2 Upon receipt of a containment isolation signal (CIS), the FBVS isolation dampers identified in Table 2.6.4-1 realign to exhaust air to the SBVS iodine filtration exhaust to the plant vent stack within the design basis closure time. or high radiation alarm signal in the Reactor Building, the FB is isolated from the NABVS by automatically closing the air supply and exhaust isolation dampers listed in Table 2.6.4-1 for Fuel Building Isolation.

7.3 The FBVS provides cooling to maintain design temperatures in the Fuel Building pump rooms, while operating in a design basis accident alignment.

7.4 The FBVS provides heating to maintain design temperatures in the Fuel Building rooms for systems containing borated fluid, while operating in a design basis accident alignment.

8.0 Inspections, Tests, Analyses and Acceptance Criteria

Table 2.6.4-3 lists the FBVS ITAAC.

Table 2.6.4-3—Fuel Building Ventilation System ITAAC (4 Sheets)

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Commitment Wording		Inspections, Tests, Analyses	Acceptance Criteria
		b. Components listed as harsh environment in Table 2.6.4-2 will be inspected to verify installation in accordance with the construction drawings including the associated wiring, cables and terminations. Deviations to the construction drawings will be reconciled to the EQDP.	b. Inspection reports exists and conclude that the components listed in Table 2.6.4-2 as harsh environment has been installed per the construction drawings and any deviations have been reconciled to the EQDP.
7.1	Upon receipt of a containment isolation signal, ~~T~~the FBVS maintains a negative pressure relative to the outside environment in the Fuel Building~~during normal operation~~.	A ~~T~~test~~s~~ will be performed to verify, upon receipt of a containment isolation test signal, ~~on the capability of~~ that the FBVS ~~to~~ maintains a negative pressure relative to the outside environment in the Fuel Building~~during normal operation~~.	The test confirms, upon receipt of a containment isolation test signal, that the FBVS maintains ~~a~~ the pressure ~~in the Fuel Building of~~ less than or equal to -0.25 inches water gauge ~~negative pressure of at least 0.25 inches water gauge~~ relative to the outside environment in the Fuel Building~~during normal operation~~.
7.2	Upon receipt of a containment isolation signal, the FBVS isolation dampers identified in Table 2.6.4-1 realign to exhaust air to the SBVS iodine filtration exhaust to the plant vent stack within the design basis closure time.~~or high radiation alarm signal in the Reactor Building, the FB is isolated from the NABVS by automatically closing the air supply and exhaust isolation dampers listed in Table 2.6.4-1 for Fuel Building Isolation.~~	A test will be performed to verify, ~~that~~ upon receipt of a containment isolation test signal~~or high radiation alarm signal in the Reactor Building~~, that the FBVS ~~is~~ isolat~~ed~~ion dampers identified ~~from the NABVS by automatically closing the air supply and exhaust isolation dampers listed~~ in Table 2.6.4-1 realign to exhaust air to the SBVS iodine filtration exhaust to the plant vent stack within the design basis closure time.~~for Fuel Building Isolation.~~	A test confirms, ~~that~~ upon receipt of a containment isolation test signal~~or high radiation alarm signal in the Reactor Building~~, that the FBVS ~~is~~ isolat~~ed~~ion ~~from the NABVS by automatically closing the air supply and exhaust isolation~~ dampers identified~~listed~~ in Table 2.6.4-1 realign to exhaust air to the SBVS iodine filtration exhaust to the plant vent stack ~~for Fuel Building Isolation~~ within 60 seconds.

2.6.6 Safeguard Building Controlled-Area Ventilation System

1.0 Description

The safeguard building controlled-area ventilation system (SBVS) provides cooling, heating, and ventilation for the hot areas of the four divisions of the Safeguard Buildings to remove equipment heat and heat generated from other sources. The SBVS also provides heat to maintain a minimum temperature in areas of the Safeguard Buildings. The SBVS provides a minimal air change rate for the buildings and controls the building pressurization to reduce spreading of contamination.

The SBVS provides the following safety-related functions:

- Isolates the volume of the hot mechanical area of the Safeguard Buildings and confines this volume by maintaining a negative pressure and removing the iodine that might be released due to post-accident operation of the safety injection system (SIS).
- Removes heat generated by equipment of the safety injection / residual heat removal systems in the hot mechanical rooms to maintain ambient temperatures during accident conditions.
- Removes heat generated by piping and equipment of the component cooling water and emergency feedwater systems in the valve rooms to maintain ambient temperatures during accident conditions.
- Removes heat generated by equipment of the hydrogen monitoring and post accident atmosphere sampling systems to maintain ambient temperatures during accident conditions.

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- Maintains a negative pressure in the Fuel Building (FB) to direct the air from the FB to the SBVS iodine filtration trains when the FB is isolated from the nuclear auxiliary building ventilation system (NABVS) on receipt of a containment isolation signal ~~or high radiation signal in the Reactor Building~~.

The SBVS provides the following non-safety-related functions:

- Ventilates the hot mechanical areas of the Safeguard Buildings and provides a minimum required air change rate during normal operation.
- Maintains acceptable ambient conditions in the hot mechanical areas of the Safeguard Buildings during normal operation.
- Maintains negative pressure and direction of flow with the supply air from the electrical division of safeguard building ventilation system (SBVSE), and exhaust air to the NABVS during normal operation.
- Confines the volume of the fuel pool hall by maintaining negative pressure and removing iodine released in the event of a fuel handling accident in the Fuel Building.

5.0 Electrical Power Design Features

5.1 The equipment designated as Class 1E in Table 2.6.6-2 are powered from the Class 1E division as listed in Table 2.6.6-2 in a normal or alternate feed condition.

5.2 Deleted.

6.0 Environmental Qualifications

6.1 Components in Table 2.6.6-2, that are designated as harsh environment, will perform the function listed in Table 2.6.6-1 in the environments that exist during and following design basis events.

6.2 Deleted.The SBVS provides recirculation cooling s,and has the capability to remove design heat load from Safeguard Building hot mechanical rooms.

7.0 Equipment and System Performance

7.1 Upon receipt of a containment isolation signal, The the SBVS maintains a negative pressure relative to the outside environment in the hot mechanical areasrooms of the Safeguard Buildings relative to the adjacent areas during normal operation.

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7.2 Deleted.Upon receipt of a high radiation signal in the hot mechanical area rooms of a Safeguard Building division during normal operation, supply and exhaust air flow is configured such that the SBVS exhaust is automatically directed to the NAVBS iodine exhaust filters.

7.3 Upon receipt of a high radiation signal in the FB, or the Reactor Building, both SBVS iodine filtration trains start automatically, the isolation dampers open to the building where the high radiation signal is initiated (either the FB or the Reactor Building), and the accident air is directed through the SBVS iodine filtration trains.

7.4 Upon receipt of a containment isolation signal or high radiation signal in the Reactor Building, the SBVS is isolated from the SBVSE and NAVBS by automatically closing the air supply and exhaust isolation dampers, both SBVS iodine filtration trains start automatically, and the FB and SB exhaust air is directed through the iodine filtration trains to maintain a negative pressure inside the FB and hot mechanical area of the SB.

7.5 The SBVS provides recirculation cooling to maintain design temperatures in the hot mechanical rooms in the Safeguard Buildings, while operating in a design basis accident alignment.

8.0 Inspections, Tests, Analyses and Acceptance Criteria

Table 2.6.6-3 lists the SBVS ITAAC.

Table 2.6.6-3—Safeguard Building Controlled-Area Ventilation System ITAAC (8 Sheets)

	Commitment Wording	Inspections, Tests, Analyses	Acceptance Criteria
7.2	~~Upon receipt of a high radiation signal in the hot mechanical area rooms of a Safeguard Building division during normal operation, supply and exhaust air flow is configured such that the SBVS exhaust is automatically directed to the NAVBS iodine exhaust filters~~Deleted.	~~A test will be performed to verify that upon receipt of a high radiation signal in the hot mechanical area of a Safeguard Building division during normal operation, the supply air control dampers (30KLC11/12/13/14 AA003 on Figure 2.6.6-1) and exhaust air control dampers (30KLC21/22/23/24 AA006 on Figure 2.6.6-2) applicable to each division reposition automatically, and the NABVS dampers reposition automatically to exhaust through the iodine exhaust filters. Test is performed separately for each Safeguard Building division~~Deleted.	~~A separate test for each Safeguard Building division confirms that upon receipt of a high radiation signal in the hot mechanical area of a Safeguard Building division, the supply air control dampers (30KLC11/12/13/14 AA003 on Figure 2.6.6-1) and exhaust air control dampers (30KLC21/22/23/24 AA006 on Figure 2.6.6-2) applicable to each division reposition automatically, and the NABVS dampers reposition automatically to exhaust through the iodine exhaust filters. All the above dampers reposition automatically within 60 seconds~~Deleted.

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Table 2.6.6-3—Safeguard Building Controlled-Area Ventilation System ITAAC (8 Sheets)

	Commitment Wording	Inspections, Tests, Analyses	Acceptance Criteria
7.3	Upon receipt of a high radiation signal in the Fuel Building, ~~or the Reactor Building,~~ both SBVS iodine filtration trains start automatically, the isolation dampers open ~~to the building where the high radiation signal is initiated (either the Fuel Building or the Reactor Building)~~, and the accident air is directed through the SBVS iodine filtration trains.	A test will be performed to verify that upon receipt of a high radiation signal in the Fuel Building ~~or the Reactor Building~~, both SBVS iodine filtration trains start automatically, the isolation dampers ~~open to the building where the high radiation signal is initiated~~ (~~either~~ the Fuel Building dampers ~~30~~KLC45 AA003/AA004 ~~or the Reactor Building dampers 30KLC45 AA005/AA006~~), the SBVS isolation dampers (~~30~~KLC45 AA001/AA002) close, and the accident air is directed through the SBVS iodine filtration trains by aligning the iodine filtration banks isolation dampers (~~30~~KLC41/42 AA001/AA002) to the open position (see Figure 2.6.6-2 for the above components). A test is performed using a simulated high radiation signal from the Fuel Building~~, and a test is performed using a simulated high radiation signal from the Reactor Building~~.	A separate test for a radiation signal in the Fuel Building ~~or Reactor Building~~ (KLK38CR001/002) confirms that upon receipt of a high radiation signal in the Fuel Building or Reactor Building, both SBVS iodine filtration trains start automatically, the isolation dampers open ~~to the building where the high radiation signal is initiated~~(~~either~~ the Fuel Building dampers ~~30~~KLC45 AA003/AA004 ~~or the Reactor Building dampers 30KLC45 AA005/AA006~~), the SBVS isolation dampers (~~30~~KLC45 AA001/AA002) close, and the accident air is directed through the SBVS iodine filtration trains by aligning the iodine filtration banks isolation dampers (30KLC41/42 AA001/AA002) to the open position (see Figure 2.6.6-2 for the above components). Above dampers close or open within 60 seconds.

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Table 2.6.6-3—Safeguard Building Controlled-Area Ventilation System ITAAC (8 Sheets)

	Commitment Wording	Inspections, Tests, Analyses	Acceptance Criteria
7.4	Upon receipt of a containment isolation signal or high radiation signal in the Reactor Building, the SBVS is isolated from the SBVSE and NABVS by automatically closing the air supply and exhaust isolation dampers, both SBVS iodine filtration trains start automatically, and the FB and SB exhaust air is directed through the iodine filtration trains to maintain a negative pressure inside the FB and SB.	A test will be performed to verify that upon receipt of a containment isolation signal or high radiation signal in the Reactor Building, the SBVS is isolated automatically by closing the SBVSE air supply isolation dampers (30KLC11/12/13/14 AA004/AA005 on Figure 2.6.6-1) and the NABVS exhaust air isolation dampers (30KLC21/22/23/24 AA007/AA008 on Figure 2.6.6-2). Both SBVS trains (shown on Figure 2.6.6-2) start automatically aligning the filter bank isolation dampers (30KLC41/42 AA001/AA002), the SB Division 1-4 exhaust trains isolation dampers (30KLC31/32/33/34AA 001), and the isolation dampers from the SB (30KLC45 AA001/AA002) and the FB (30KLC45 AA003/AA004) to the open position, and maintaining a negative pressure inside the FB and SB.	A test confirms that upon receipt of a containment isolation signal or high radiation signal in the Reactor Building (JYK15CR101, JYK15CR102, JYK15CR103, JYK28CR101), the SBVS is isolated automatically within 60 seconds by closing the SBVSE air supply isolation dampers (30KLC11/12/13/14 AA004/AA005 on Figure 2.6.6-1) and the NABVS exhaust air isolation dampers (30KLC21/22/23/24 AA007/AA008 on Figure 2.6.6-2). Both SBVS trains (shown on Figure 2.6.6-2) start automatically aligning the filter bank isolation dampers (30KLC41/42 AA001/AA002) (30KLC21/24 AA010) (30KLC31/32/33/34 AA003) to the open position, aligning the SB Division 1-4 exhaust trains isolation dampers (30KLC31/32/33/34AA 001) to the open position, and aligning the isolation dampers from the SB (30KLC45 AA001/AA002) and the FB (30KLC45 AA003/AA004) to the open position, and maintaining a minimum negative pressure of 0.25 inches water gauge inside the FB and SB. Above dampers close or open within 60 seconds.

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Table 3.2.2-1—Classification Summary
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KKS System or Component Code	SSC Description	Safety Classification (Note 15)	Quality Group Classification	Seismic Category (Note 16)	10 CFR 50 Appendix B Program (Note 5)	Location (Note 17)	Comments/ Commercial Code
JYK, LBA, CLE20, CLF20, CLG20, CLH20	**Radiation Monitoring System**						JYK takes input from sensors in other systems
JYK15CR101/102/ 103 JYK28CR101 JYK00GH111/112 JYK00GH007/008	High Range Dose Rate Monitor	S	N/A	I	Yes	UJA, UFA	
JYK00GH061/062 JYK00GH109/110	BBX Main Control Room Vent Duct Gamma	S	N/A	I	Yes	UJK	RAI 273, Q. 11.05-5(1)
LBA10CR811 LBA10CR821 LBA10CR831 LBA10CR841	Main Steam Line Radiation Monitors Division 1	S	N/A	I	Yes	1UJE	
LBA20CR811 LBA20CR821 LBA20CR831 LBA20CR841	Main Steam Line Radiation Monitors Division 2	S	N/A	I	Yes	2UJE	
LBA30CR811 LBA30CR821 LBA30CR831 LBA30CR841	Main Steam Line Radiation Monitors Division 3	S	N/A	I	Yes	3UJE	

Table 3.2.2-1—Classification Summary
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KKS System or Component Code	SSC Description	Safety Classification (Note 15)	Quality Group Classification	Seismic Category (Note 16)	10 CFR 50 Appendix B Program (Note 5)	Location (Note 17)	Comments/ Commercial Code
LBA40CR811 LBA40CR821 LBA40CR831 LBA40CR841	Main Steam Line Radiation Monitors Division 4	S	N/A	I	Yes	4UJE	
JYK	Balance of JYK System	NS	N/A	NSC	No	UJA, UFA, UJH, UJK, UKS, UKA	
JYK	Balance of JYK System	NS-AQ	N/A	NSC	No	UFA	
CLE20	Radiation Monitoring Cabinet Division 1	S	N/A	I	Yes	1UJK	RAI 273, Q. 11.05-5(1)
CLF20	Radiation Monitoring Cabinet Division 2	S	N/A	I	Yes	2UJK	
CLG20	Radiation Monitoring Cabinet Division 3	S	N/A	I	Yes	3UJK	
CLH20	Radiation Monitoring Cabinet Division 4	S	N/A	I	Yes	4UJK	
~~JYK15/28 CR101~~	~~High Range Dose Rate Monitors~~	~~S~~	~~N/A~~	~~I~~	~~Yes~~	~~UJA~~	

Table 3.2.2-1—Classification Summary
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KKS System or Component Code	SSC Description	Safety Classification (Note 15)	Quality Group Classification	Seismic Category (Note 16)	10 CFR 50 Appendix B Program (Note 5)	Location (Note 17)	Comments/ Commercial Code
~~JYK~~	~~Balance of JYK System~~	~~NS~~	~~N/A~~	~~NSC~~	~~No~~	~~UJA, UFA, UJH, UJK, UKS, UKA~~	RAI 273, Q. 11.05-5(1)
BUA	**Control Rod Drive Control System**						
BUA	Contactors only	S	N/A	I	Yes	1UJK, 4UJK	
BUA	Balance of Control Rod Drive Control System	NS-AQ	N/A	II	Yes	1UJK, 4UJK	
CPE	**Seismic Monitoring System**	NS-AQ	N/A	I	Yes	Site Specific	Exception to Criteria based on the seismic monitoring required to be functional during and after a seismic event
CR	**Process Automation System**						
CR	~~Components used for Accident Monitoring Functions~~Process Automation System	NS~~-AQ~~	N/A	NSC	No	UJK, UBP, UBA, UAA	
CR	~~Components used for Accident Monitoring Functions~~Process Automation System (adjacent to Safety Related Equipment)	NS-AQ	N/A	II	Yes	UJK, UBP	Located in close proximity to safety-related equipment

ICECON model. However, the volume impact from this additional heat sink is not considered in the combined containment free volume or the maximum containment free volume calculations. An additional assumption increases the nominal heat transfer surface areas by 10 percent to increase the energy removed from the containment atmosphere, which is consistent with a conservative PCT calculation.

6.2.1.6 Tests and Inspections

Refer to Section 3.8.1.7 and Section 3.8.2.7 for testing and inspection requirements for the containment structure. Refer to Section 6.2.6 for the containment leakage rate testing program, and Section 6.6 for inservice inspection of ASME Class 2 and 3 components. Containment testing and inspections are also included in the Technical Specifications (Chapter 16).

6.2.1.7 Instrumentation Requirements

Refer to Section 7.3 for engineered safety features instrumentation. Refer to Section 11.5 and Section 12.3.4 for radiation monitoring instrumentation.

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normal operation train is not able to maintain the subatmospheric pressure in the annulus.

The motor-operated air-tight dampers—located on the normal operation filtration train supply and exhaust ducts—isolate the secondary containment in case of a postulated accident. The redundant dampers in the supply and exhaust trains are powered by different electrical divisions backed by separate emergency diesel generators. The dampers can be operated automatically or manually from the main control room (MCR). In the event of a station blackout (SBO), these dampers are automatically closed by batteries.

The fire dampers on both supply and exhaust trains are located at the wall penetration between the Fuel Building and the annulus. These dampers are equipped with thermal sensors for automatic closing, and can be closed or re-opened remotely if not released by the thermal sensor.

6.2.3.2.2.2 AVS Accident Trains

The AVS accident filtration trains are shown on Figure 6.2.3-2. The filtration trains are engineered safety feature (ESF) filters and are used during postulated accidents to contain leakage from the primary containment by maintaining a subatmospheric pressure in the annulus. The exhaust air from the annulus is filtered before release to the environment via the vent stack.

There are two full capacity ESF trains, each consists of an air-tight motor-controlled damper, an electrical heater, a pre-filter, an upstream HEPA filter, an iodine absorber, a downstream HEPA filter, an air-tight motor controlled damper, a fan, and a back-draft damper. The filter system components are designed in accordance with Regulatory Guide 1.52, and are described in Section 6.5.1.

During a postulated accident, the ESF filtration trains collect the containment leakage from the annulus, remove airborne radioactivity through the filtration train, and release the filtered air to the vent stack. The AVS accident trains reduce the pressure in the annulus to at least -0.25 inches water gauge or less and maintain the lower subatmospheric pressure. The system is capable of maintaining a uniform negative pressure throughout the secondary containment structure following the design basis loss of coolant accident (LOCA).

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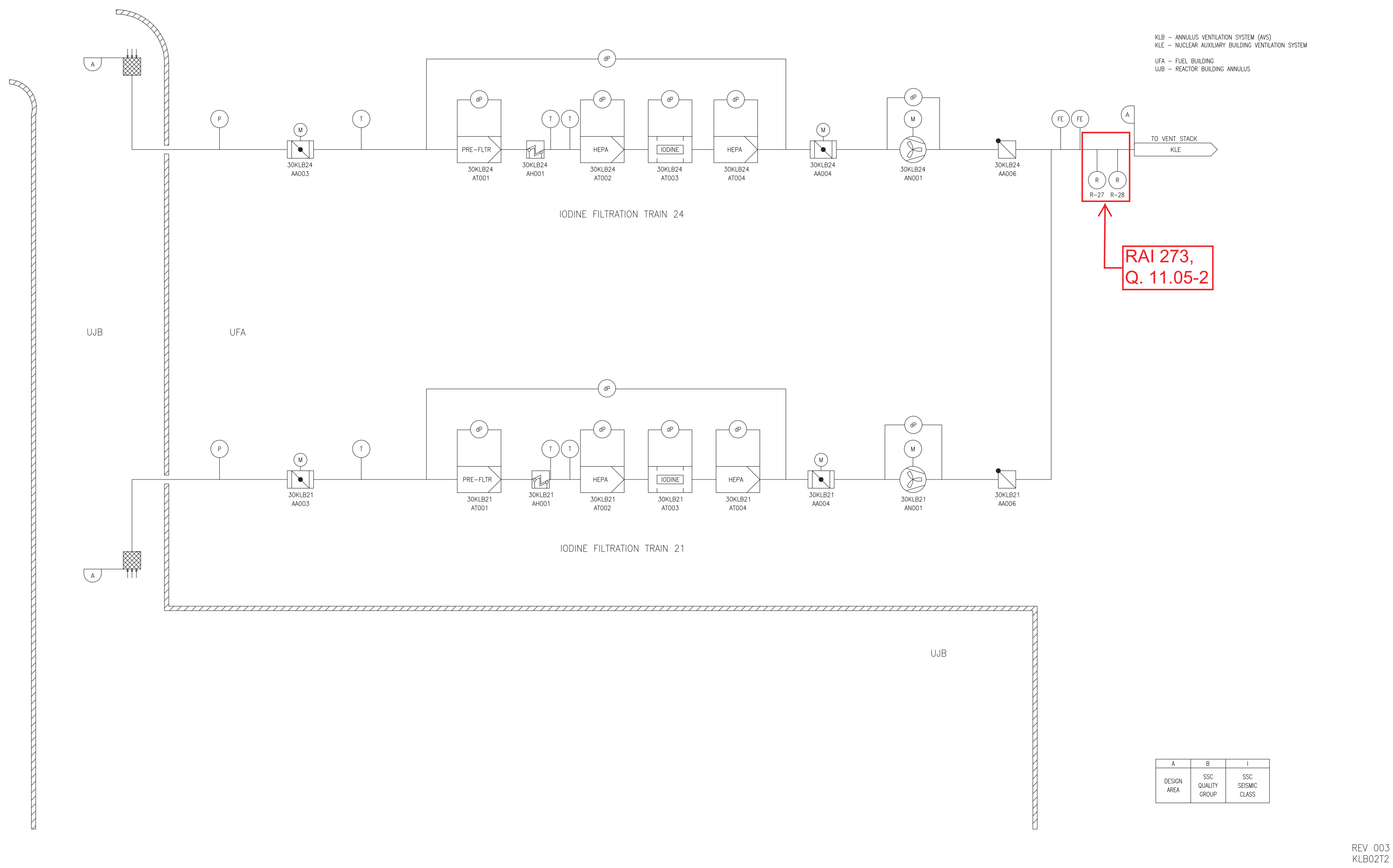
The exhaust air is monitored and sampled for radiation levels before release to the vent stack, as described in Section 11.5.3.1.10~~Section 12.3.4~~ and Table 11.5-1, Monitors R-27 and R-28.

The two ESF trains are physically separated by being installed in separate rooms of the Fuel Building, which are also in separate fire areas. The two ESF trains are powered by different electrical divisions backed by separate emergency diesel generators.

Figure 6.2.3-2—AVS Accident Trains



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- One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve is not used as the automatic isolation valve outside containment.

Certain containment isolation valves in systems required for accident mitigation do not receive a containment isolation actuation signal because their systems are required for accident mitigation. However, the containment isolation function may be required postaccident; therefore, the valves can be closed remotely from the main control room (MCR).

The four safety injection system trains draw suction from the in-containment water storage tank (IRWST). These suction lines each contain only one remotely operated isolation valve located outside the containment. The isolation valves do not receive an automatic containment isolation signal. Rather, the valves remain open during a postulated accident to support the safety injection function. The section of piping located between the IRWST and each isolation valve is contained in a guard pipe, thus providing a double leak-tight penetration barrier. This arrangement provides a higher level of safety than the standard isolation valve arrangement. The SIS outside containment is protected from missiles and seismic events, is constructed to Quality Group B and Seismic Category I standards, and has a design temperature and pressure rating equal to that of the containment. The IRWST is provided with sump screens to prevent debris entrainment into the SIS, as described in Section 6.3.2.5.

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The containment building ventilation system (CBVS) includes purge lines that connect directly to the containment atmosphere, as described in Section 9.4.7. Area radiation monitors provide a containment isolation signal on high radiation (see Section 12.3 and Table 12.3-4).

The severe accident heat removal system (SAHRS) draws suction from the IRWST. The piping inside containment is embedded in concrete and the penetration piping is protected by a guard pipe. Two remotely operated isolation valves outside containment receive an automatic containment isolation signal to close. An identical arrangement exists for the suction line from the IRWST to the chemical and volume control (CVCS) charging pumps.

The positions of the individual containment isolation valves depend on the plant's operating mode, and on the specific fluid system's functional requirements for that mode. The positions of each containment isolation valve under normal and accident conditions are listed in Table 6.2.4-1.

6.2.4.2.3 Closed System Isolation Valves

Lines that penetrate the containment and are neither part of the RCPB nor connected directly to the containment atmosphere have at least one isolation valve that is located outside the containment. The isolation valve is either automatic, or locked closed, or post-accident filtration. Section 6.5.1 describes the engineered safety features (ESF) filter systems and fission product removal capability for the CRACS.

Section 3.8.4 contains elevation and plan views of the Safeguard Buildings. Figure 2.3-1 provides the relative locations of potential radiological release points and the CRACS air intakes. ~~The evaluation of potential toxic chemical accidents is addressed in Section 2.2.3.~~ Figure 6.4-1 through Figure 6.4-3 illustrate the CRE layout, including surrounding corridors, doors, stairwells and shielded walls.

One outside air intake for the CRACS is located in Safeguard Building 2 and the other is located at a separate location on Safeguard Building 3, to prevent intrusion of ~~toxic gases (includes CO and CO_2) or~~ radiological contamination.

The CRACS intakes are located on the roof of Safeguard Buildings 2 and 3~~, to prevent intrusion of toxic gases or radiological contamination~~. The two intakes are physically separated and are removed from potential radiological release points, including the main steam relief exhaust, the Safeguard Building depressurization shafts, and the vent stack, in both lateral and vertical directions. Section 15.0.3 identifies the bounding atmospheric release point used in the radiological analyses.

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Radiation monitors (R-29 and R-30, Table 11.5-1) in the CRACS supply air duct continuously measure the concentration of radioactive materials in the supply air. The control room airborne radioactivity monitoring system is addressed in Section 12.3.4 and Section 11.5.3.1.11.

The main features related to control room habitability of the CRACS design are:

- Under normal operating conditions:
  - The ventilation system operates in the recycling mode with fresh air makeup.
  - The air makeup rate corresponds to the exhausts from the kitchen and restrooms ~~rooms~~ and leakage out of the area ~~due to the controlled overpressure~~.
- The ventilation system maintains an ambient condition for comfort and safety of control room occupants and to support operability of the MCR components during normal operation, anticipated operational occurrences (AOO), and design bases accidents (DBA).
- The ventilation system maintains a positive pressure of 0.125 inches water gauge as a minimum within the CRE areas with respect to adjacent environmental zones to prevent uncontrolled, unfiltered in-leakage during normal and accident conditions. The filtered outside air supply rate during accident conditions corresponds to 0.3 volume changes per hour.

6.4.4 Design Evaluations

Section 9.4.1 contains the design evaluation of the CRACS. Fire protection inside and outside the CRE boundary is addressed in Section 9.5.1.

The total effective dose equivalent (TEDE) for the MCR occupants throughout the duration of any postulated DBA does not exceed the limits of GDC 19. The evaluation of radiological exposure to control room operators and the dose calculation model for the MCR is described in Section 15.0.3.

The CRE is designed, maintained and tested in accordance with RG 1.196 and RG 1.197. Habitability systems provide the capability to detect and protect personnel within the CRE boundaries from external fires, smoke, ~~toxic gases~~ and airborne radioactivity.

A COL applicant that references the U.S. EPR design certification will confirm that the radiation exposure of MCR occupants resulting from a DBA at a nearby unit on a multi-unit site is bounded by the radiation exposure from the postulated design basis accidents analyzed for the U.S. EPR; or confirm that the limits of GDC 19 are met.

~~The evaluation of potential toxic chemical accidents is addressed by the COL applicant in Section 2.2.3 and includes the identification of toxic chemicals. A COL applicant that references the U.S. EPR design certification will evaluate the results of the toxic chemical accidents from Section 2.2.3 and address their impact on control room habitability in accordance with RG 1.78.~~

6.4.5 Testing and Inspection

Testing and inspection of the CRACS are described in Section 9.4.1. Refer to Section 14.2 (test abstract #082) for initial plant testing.

Periodic testing to confirm CRE integrity is performed using testing methods and at testing frequencies consistent with RG 1.197. The air in-leakage test (tracer gas test) of the CRE boundary is performed in accordance with ASTM E741 (Reference 3). Air quality testing is performed in accordance with ANSI/ASHRAE 52.2 (Reference 4) and ASME N510 (Reference 5).

The control room envelope habitability program in Technical Specifications Section 5.5.17 defines testing requirements.

6.4.6 Instrumentation Requirements

The instrumentation and control features of the CRACS are described in Section 9.4.1. Radiation monitoring equipment for the CRE is described in Section 12.3.4, Section 11.5.3.1.11 and Table 11.5-1, Monitors R-29 and R-30.

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6.5 Fission Product Removal and Control Systems

Several U.S. EPR systems are designed to prevent or limit the release of fission products following a postulated design basis accident (DBA) or fuel handling accident. These systems include the engineered safety features (ESF) filter systems, the primary and secondary containment structures and systems, and the containment isolation system. This section describes the systems and how they mitigate fission product release. Section 15.0.3 presents the DBA radiological evaluations that demonstrate the effectiveness of these fission product removal and control systems in maintaining radioactivity releases within regulatory limits. The sequence of events assumed in the dose analyses for DBAs are also presented in Section 15.0.3.

6.5.1 ESF Filter Systems

ESF filter systems consist of filters, heaters, fans, dampers, and ductwork that remove particulate and gaseous radioactive material from the atmosphere. Four ESF filter systems work in conjunction with the five ventilation systems listed below:

- Main control room (MCR) air conditioning system (CRACS), described in Sections 6.4 and 9.4.1.
- Annulus ventilation system (AVS), described in Section 6.2.3.
- Safeguard building (SB) controlled area ventilation system (SBVS), described in Section 9.4.5.
- Fuel building (FB) ventilation system (FBVS), described in Section 9.4.2.
- Containment building ventilation system (CBVS) for the low-flow purge exhaust subsystem, described in Section 9.4.7.

The sections identified for the ventilation systems provide the descriptions and piping and instrumentation diagrams of these ventilation systems, along with design bases and safety evaluations.

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Ventilation systems are aligned to ESF filter systems to support plant operations and accident mitigation. ESF filters in the CBVS low-flow purge exhaust subsystem are aligned during purging operations. The FBVS aligns to the SBVS ESF filters in response to ~~high radiation in the Reactor Building (RB) or~~ a containment isolation signal. ~~Portions of the FBVS and CBVS are aligned to the ESF filters of the SBVS during the movement of irradiated fuel assemblies.~~

6.5.1.1 Design Bases

The ESF filter systems mitigate the consequences of postulated accidents by removing particulate and gaseous radioactive material from the atmosphere that could be released to the environment (GDC 41). The ESF filter systems are designed to permit periodic inspection and periodic pressure and functional testing (GDC 42, GDC 43).

The ESF filter systems remove radioactive material from the atmosphere to maintain the MCR in a safe condition under accident conditions, including loss of coolant accidents (LOCA), in accordance with GDC 19. These systems, although not credited in the radiological analyses, also provide protection during fuel handling in accordance with GDC 61.

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Design bases for radiation monitoring are presented in Section 12.3.4 and Section 11.5.1.

The ESF filter systems are designed to meet the design and performance requirements of RG 1.52, Revision 3, ASME N509 (Reference 1), and ASME N510 (Reference 2).

6.5.1.2 System Design

6.5.1.2.1 General System Design

The ESF filter systems described in this section are designed to limit the release of fission products to the environment and to limit radiation dose to the personnel in the MCR. Regardless of the application, each ESF filter system consists of two independent trains. Each train has an activated charcoal carbon adsorber with motorized dampers, an electric heater, a prefilter, and inlet and outlet high efficiency particulate air (HEPA) filters. A booster fan and isolation dampers are included to provide the flow to the vent stack for the discharge of filtered air.

~~Refer to Section 3.2 for the seismic and system quality group classification of the~~Table 3.2.2-1 provides the seismic and other design classifications for the components of the ESF filter systems.

6.5.1.2.2 Component Design

Filter Air Heaters

The iodine adsorption efficiency of the filters decreases at high humidity. Therefore, the radiological filter air heaters limit the relative humidity to a maximum of 70 percent so that the carbon adsorber can remove iodine from the exhaust air. The filter air heaters are located upstream of the iodine filtration units to prevent excessive moisture accumulation in the charcoal filter beds. The heaters meet the requirements of ASME AG-1 (Reference 3). The heater sizes are as follows:

- AVS: 6 kW nominal heater rating.
- SBVS: 11 kW nominal heater rating.

- CBVS: ≥ 2700 cfm and ≤ 3300 cfm (nominal 3000 cfm), face velocity 375 fpm, configuration 2 High x 1 Wide.

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The ESF filter systems in the CRACS, AVS, and SBVS are aligned automatically with their associated ventilation systems upon receipt of an ESF actuation signal, including safety injection, or detection of high radiation levels (refer to Table 11.5-1, Monitors R-29 and R-30 (CRACs) and R-25 (SBVS)). The ESF filter systems may also be manually aligned. The ESF filter systems can also be aligned to the FB and the containment area during fuel handling of irradiated fuel assemblies. The systems are placed in line with the FBVS and CBVS in case of a fuel handling accident. With this ESF filter system alignment, the offsite release of radioactive material from a fuel handling accident does not exceed regulatory limits. During containment purging, the ESF filters in the low-flow purge exhaust subsystem of the CBVS are aligned to reduce radioactive releases in case of a rod ejection accident occurring during purging operations.

Each ESF filter system is sized to accommodate the required ventilation flow and to remove greater than 99 percent of the fission products that could be entrained in the air. The ESF filter systems conform to the requirements of RG 1.52.

Performance evaluations of the ventilation systems that operate in conjunction with the ESF filter systems to limit fission product release to the environment or the MCR are presented in the sections corresponding to the ventilation systems.

6.5.1.4 Tests and Inspections

Refer to Section 14.2 (test abstracts #076, #077, #082, and #083) for initial plant testing of the ESF filter systems. Routine testing and inspection of ESF filter systems are conducted under the ventilation filter testing program in Technical Specifications Section 5.5.10. Laboratory testing of samples of activated carbon adsorber material is performed in accordance with ASTM D3803 (Reference 5) and RG 1.52.

6.5.1.5 Instrumentation Requirements

Instrumentation and controls provide automatic operation and remote control of the ESF filter systems and continuous indication of system parameters. ~~Instrumentation and controls related to the ESF filters are described with the associated ventilation systems in the corresponding sections. Instrumentation and controls for ESF filter systems are provided in Table 9.4.1-1—Minimum Instrumentation, Indication, and Alarm Features for CREF (Iodine Filtration) Train Subsystem.~~

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.

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- Containment radiation ~~intensity~~ activity (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~~ is 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.

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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	Type B	Type C	Type D	Type E
1	Annulus Ventilation System Gamma Activity	See Table 11.5-1, R-27	2	1 year	NS-AQ	Yes	I			X		X
2	~~Condenser Evacuation System Exhaust~~ Vent System for Air Removal Radiation	See Table 11.5-1, R-3	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	

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
11	Fuel Building Fuel Pool Dose Rate	See Table 12.3-3	1	24 hours	NS	No	NSC					X
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 x 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, R-55 through R-58	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		

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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
23	Reactor Coolant System Wide Range Cold Leg Temperature	32 to 662 °F	2	24 hours	S	Yes	I	X	X		X	
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, R-25	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	

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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
33	Source Range Neutron Flux	0.05 to 5 x 10^4 n/ cm^2-s	2	1 year	S	Yes	I		X			
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, R-4	1	1 year	NS	No	NSC					X
36	Vent Stack Iodine Activity	See Table 11.5-1, R-4	1	1 year	NS	No	NSC					X
37	Vent Stack Gas Activity	See Table 11.5-1, R-4	1	1 year	NS	No	NSC					X

Notes:

1. Environmental Qualification.

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ESWS are powered by Class 1E electrical buses and are emergency powered by the EDGs.

The non-safety-related dedicated division contains a dedicated ESWS pump, debris filter, piping, valves, controls, and instrumentation. The non-safety related ESWS pumps cooling water from the division four UHS cooling tower basin to the dedicated CCWS HX and back to the division four UHS cooling tower during severe accidents (SA). The dedicated ESWS ~~pump~~train is powered by Class 1E electrical ~~buses~~Division 4 and is capable of being supplied by an EDG or a station blackout diesel generator (SBODG).

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Refer to Section 12.3.6.5.7 for ~~essential service water system~~ESWS design features which demonstrate compliance with the requirements of 10 CFR 20.1406. Refer to Section 11.5.4 and Table 11.5-1, radiation monitors R-66 through R-70 for process and effluent radiological monitoring and sampling within the ESWS.

9.2.1.3 Component Description

9.2.1.3.1 Safety-Related Essential Service Water Pumps

Each of the four safety-related cooling divisions contains one 100 percent capacity pump. During normal operating conditions, two of the four divisions are operating. The required flow rate of each ESWS pump is defined by the heat to be removed from the system loads. Design parameters are listed in Table 9.2.1-1. The pumps are designed to fulfill the corresponding minimal required design mass flow rate plus an additional minimum flow margin of approximately 12.5 percent, under the following conditions:

- Minimal water level without cavitation.
- Head losses in the cooling water inlet piping according to full power plant operation.
- Fluctuations in the supplied electrical frequency.
- Increased pipe roughness due to aging and fouling.
- Fouled debris filters.
- Maximum pressure drop through the system HXs.
- Minimum water level in cooling tower basin considers minimum submergence requirements to prevent vortex effects, and net positive suction head to prevent cavitation of the ESWS pump.
- Maximum allowable water level differential across the coarse and fine screens.

Figure 9.2.1-1—Essential Service Water System Piping & Instrumentation Diagram Sheet 2 of 4

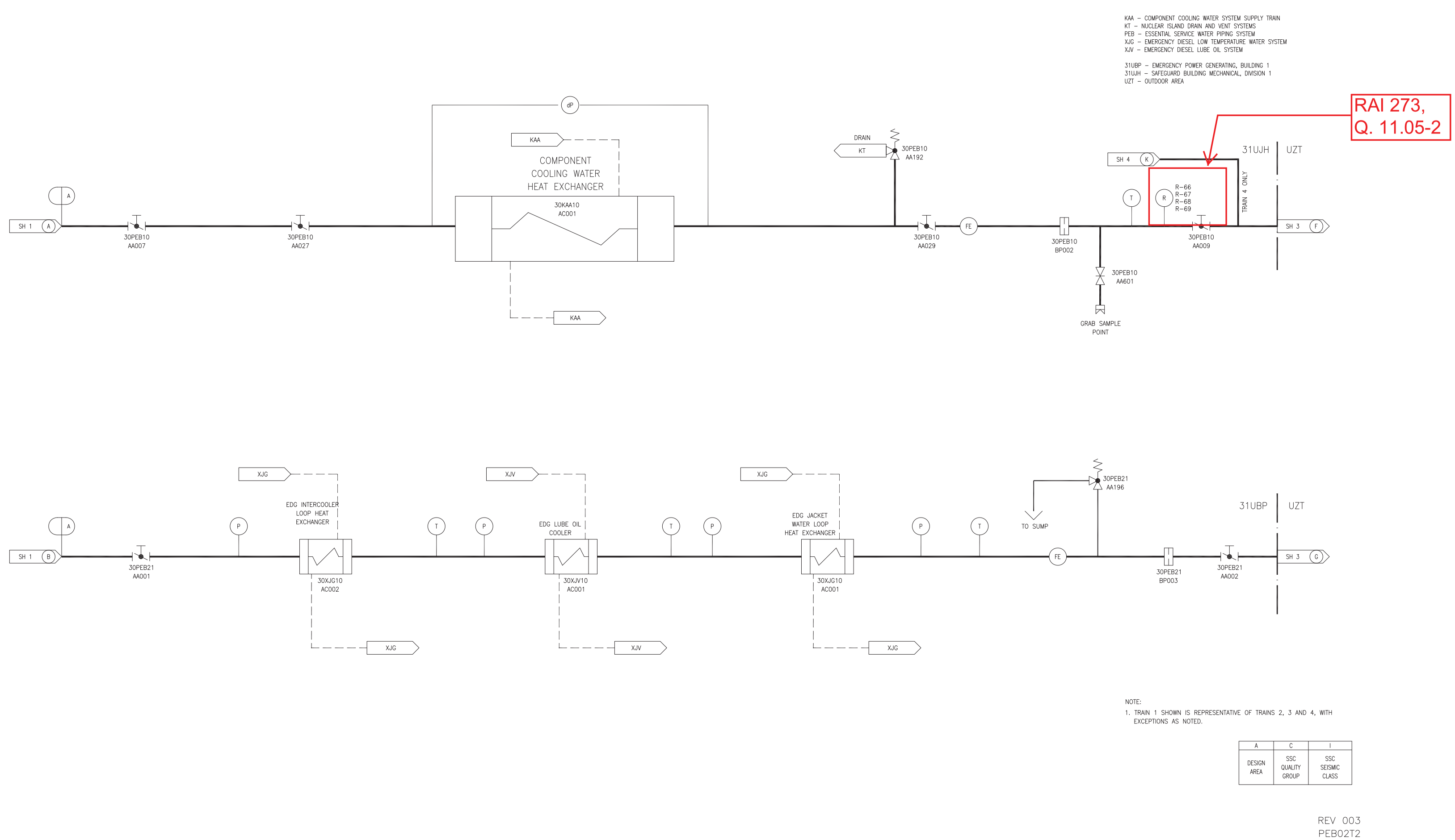
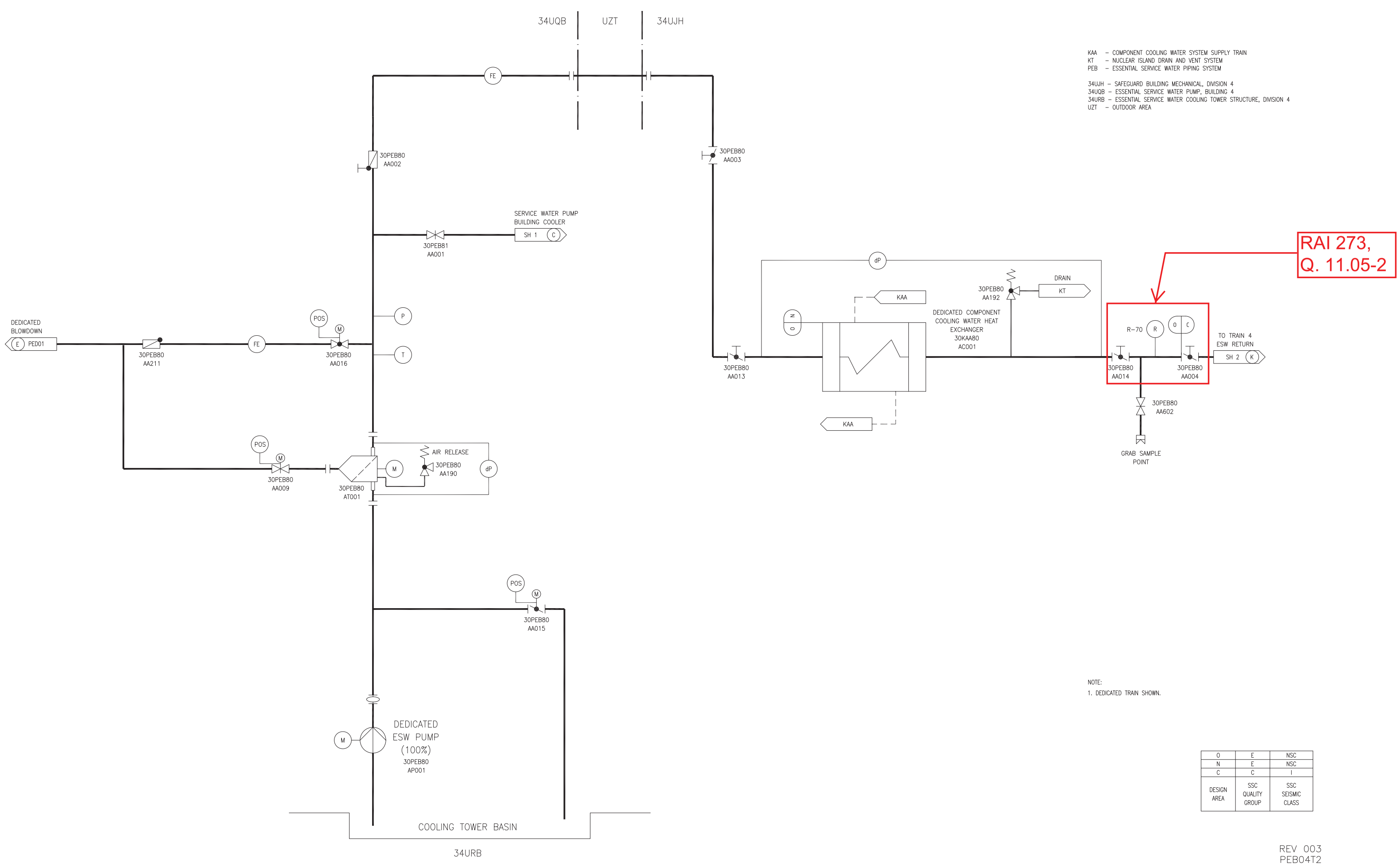


Figure 9.2.1-1—Essential Service Water System Piping & Instrumentation Diagram
Sheet 4 of 4



Next File

9.2.2.3 System Operation

9.2.2.3.1 Normal System Operation

The safety-related CCWS is a four train concept which allows sharing of operational and safety users during normal operation and to separate them in case of design and beyond design based accidents. Each physically separated train consists of a main pump, motor cooler, an HX, surge tank, sample piping with permanently installed radiation monitor (refer to Section 11.5.4.4 and Table 11.5-1, Monitors R-35 through R-38), a chemical addition tank and pairs of common header isolation valves. Each train also supplies cooling to the associated MHSI pump motor cooler and the LHSI HX. The CCWS trains 2 and 3 also provide cooling to the LHSI pump motor and seal water coolers.

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During normal operations, one or two trains can be in operation in each pair of associated trains (trains 1 and 2 or trains 3 and 4) to cool the two common sets of users (common 1.b (2.b), with or without common 1.a (2.a)). Each of the common headers may be split so that one of the two associated trains is supplying the common 1.a (2.a) header and the other is supplying the common 1.b (2.b) header to enhance cooling efficacy of the cooling chain.

The common 1.a header provides cooling to the first FPCS train and the common 2.a header provides cooling to the second FPCS train. These are separated from the other operational loads (1.b and 2.b) that the CCWS supplies to maintain FPCS cooling capacity during maintenance or plant outages.

Each train in operation:

- Cools the fluid in a closed loop through the CCWS exchanger.
- Provides recirculation in the surge tank for CCW mixing.
- Cools the main auxiliary pumps coolers (SIS and CCWS pumps).
- Is continuously sampled for radioactivity leakage into the CCWS using the permanently installed radiation monitor (refer to Section 11.5.4.4 and Table 11.5-1, Monitors R-35 through R-38).

For each of common headers (1 and 2), the associated safety-related trains are isolated from each other by the four switchover isolation valves located on the supply and return side of each common header sub-loop (a and b).

The permissive for the switchover valve of the on-coming CCWS train is by limit switch of the switchover valve of the off-going CCWS train. The switchover valve of the off-going train must signal "Closed" prior to the switchover valve of the on-coming train being allowed to open. Train separation of the redundant CCWS divisions is can sample directly the content of the water boxes of the HX via a dedicated sampling line.

- Before connecting an LHSI train for the first time to the RCS in the RHR mode, the relevant LHSI pump is started on its minimum flow line through the in-containment refuelling water storage tank. This second line of defense permits detection of the failed HX.

- When a LHSI pump is started on its minimum flow line or on the closed loop, a faulty HX causes leakage into the CCWS. This leakage is detected by the train radiation monitor or an uncontrolled level rise in the CCWS surge tank (refer to Section 11.5.4.4 and Table 11.5-1, Monitors R-35 through R-38).

- Provisions are required to minimize the risk of CCWS leakage during maintenance on the LHSI trains. The component boundary can be verified via a pressure test on the CCWS side or a pressure test of the RHR/LHSI.

Failure of a LHSI Pump Seal Fluid Cooler

When a LHSI pump is not in operation, the isolation valve upstream and a check valve downstream of the seal cooler prevent any leakage from the CCWS to the RHRS.

Tube Rupture Inside Thermal Barrier

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The possibility of diluting the RCS via a faulty RCP thermal barrier exists only when the RCS is in a low pressure state.

After a predetermined time delay (≈15 minutes), which allows for RCP coast down and when the RCS pressure is low, the CCWS will be automatically isolated from the RCP thermal barrier via the CCW inlet and outlet isolation valves.

Tube Rupture Inside a CVCS HP Cooler

A leak from the CVCS into the CCWS will be detected by radiation monitors (refer to Table 11.5-1, Monitors R-51 through R-54) on the CCWS return lines from each of the CVCS HP coolers. A high radiation alarm from either of these monitors will trigger automatic isolation of the affected CVCS HP cooler via motor-operated valves (KBA 11/12 AA001/003) in the CVCS (See Figure 9.3.4-1 and Section 11.5.4.17 and Monitors R-51 and R-52 (CCWS Common Loop 1) and R-53 and R-54 (CCWS Common Loop 2)).

RCS Cooldown with Less Than 3 Trains

If less than three trains of the plant cooling chain (RHR, CCW, ESW) are available, stabilization of the RCS temperature is achievable. If the RCS must be cooled to cold shutdown conditions, it may be necessary to remove non-essential CCWS user loads from operation.

for pump coast down. This action prevents potential leakage of the CCWS into the SIS train.

- In the event of a pump low flow condition, the associated LHSI HX isolation valve automatically opens to provide a minimum flow path for CCWS pump protection. In the event of a pump high flow condition, the FPCS HX outlet flow control valve is closed to its minimum opening mechanical stop position to reduce the CCWS flow rate and to maintain normal pump operation.

- The CCWS surge tanks are instrumented with level indication and graduated level control and equipment protection set points designated from lowest to highest level (MIN4, MIN3, MIN2, MIN1, MAX1, MAX2, MAX3 and MAX4). A CCWS train can operate continuously so long as the water level in its surge tank is maintained between MIN1 and MAX1.

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- Detection of increasing radiation in the CCWS from the CVCS HP coolers indicates leakage and triggers automatic isolation of the affected CVCS HP cooler via motor-operated valves (KBA11/12 AA001/003) in the CVCS. Refer to Section 11.5.4.17 and Table 11.5-1 (Monitors R-51 through R-54) for details on the associated radiation monitoring equipment..

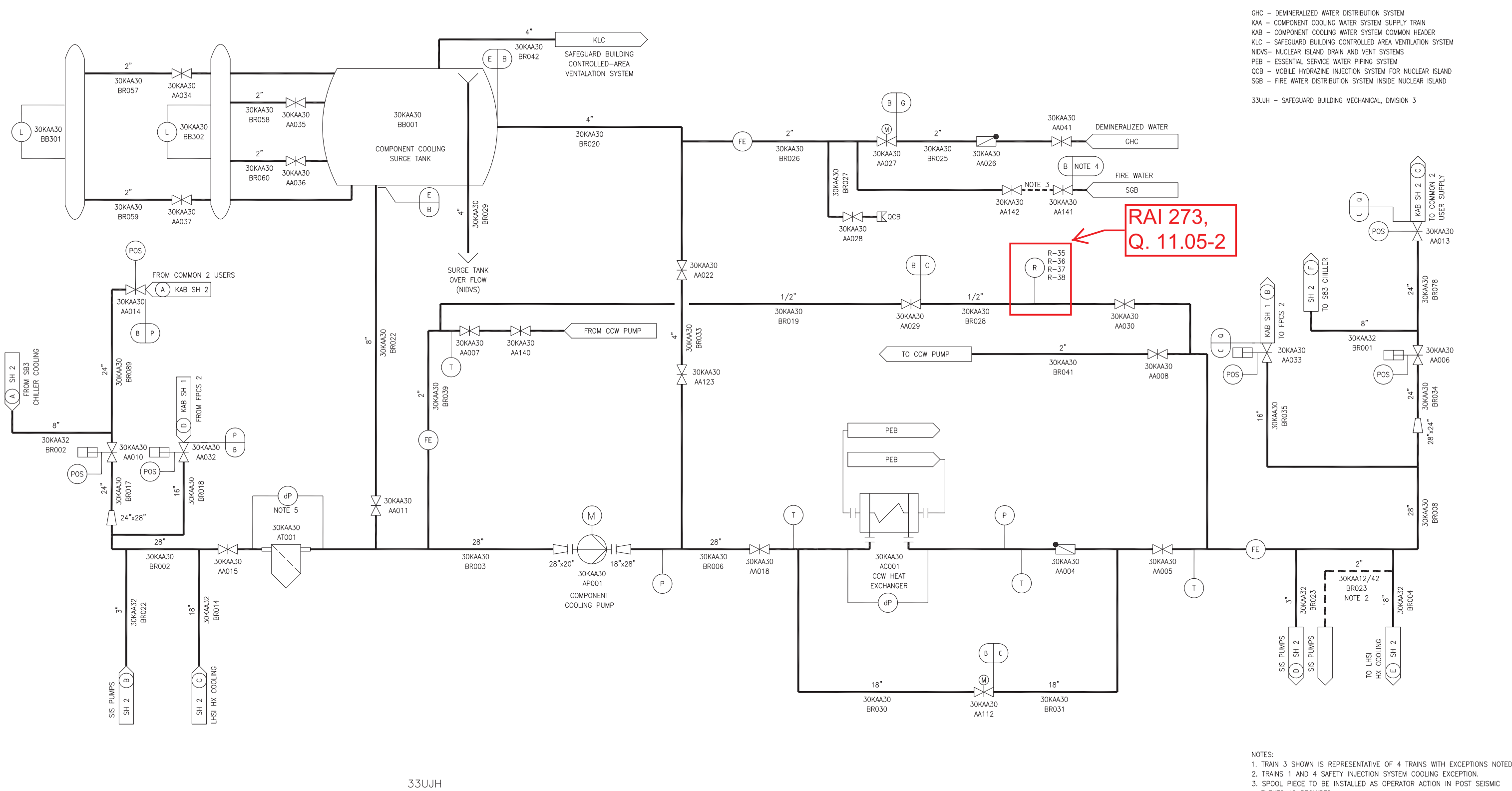
- Leakage of reactor coolant into the CCWS from such trains as the LHSI HXs is also indicated by increasing radiation in the CCWS and prompts isolation of the train. Refer to Section 11.5.4.4 and Table 11.5-1 (Monitors R-35 through R-38) for details on the associated radiation monitoring equipment.~~Only the RCP thermal barrier and CVCS HP cooler leaks result in automatic isolation of the failed users.~~

- Manual or automatic actuation of a CCWS pump automatically actuates the corresponding ESWS pump.

The CCWS is designed with redundant level indication for each surge tank that is transmitted to the control room. The demineralized water makeup line for each CCWS surge tank contains a flow indication device that transmits to the control room. The combination of continuously monitored tank level and demineralized water makeup flow in real time provides the operators the ability to retrieve trending data on surge tank levels and normal makeup flow at any time and for any range of operating time. The ability to retrieve and analyze this data in real time from the MCR provides operators the ability to realize when 7 day train leakage is trending near a threshold value. This provides the operators the ability to take corrective action prior to exceeding the maximum allowed 7 day train leakage. Trending CCWS surge tank levels is important to the operation of the system in post-seismic operation because the CCWS is designed to maintain a reserve volume of water in each tank to allow the system to operate for 7 days after an earthquake with no operator action if normal makeup from demineralized water is not available.

Surveillance Requirement 3.7.7.2 is written to verify CCWS train leakage on a 31 day frequency.

Figure 9.2.2-1—Component Cooling Water System Trains 1 through 4
Sheet 1 of 2

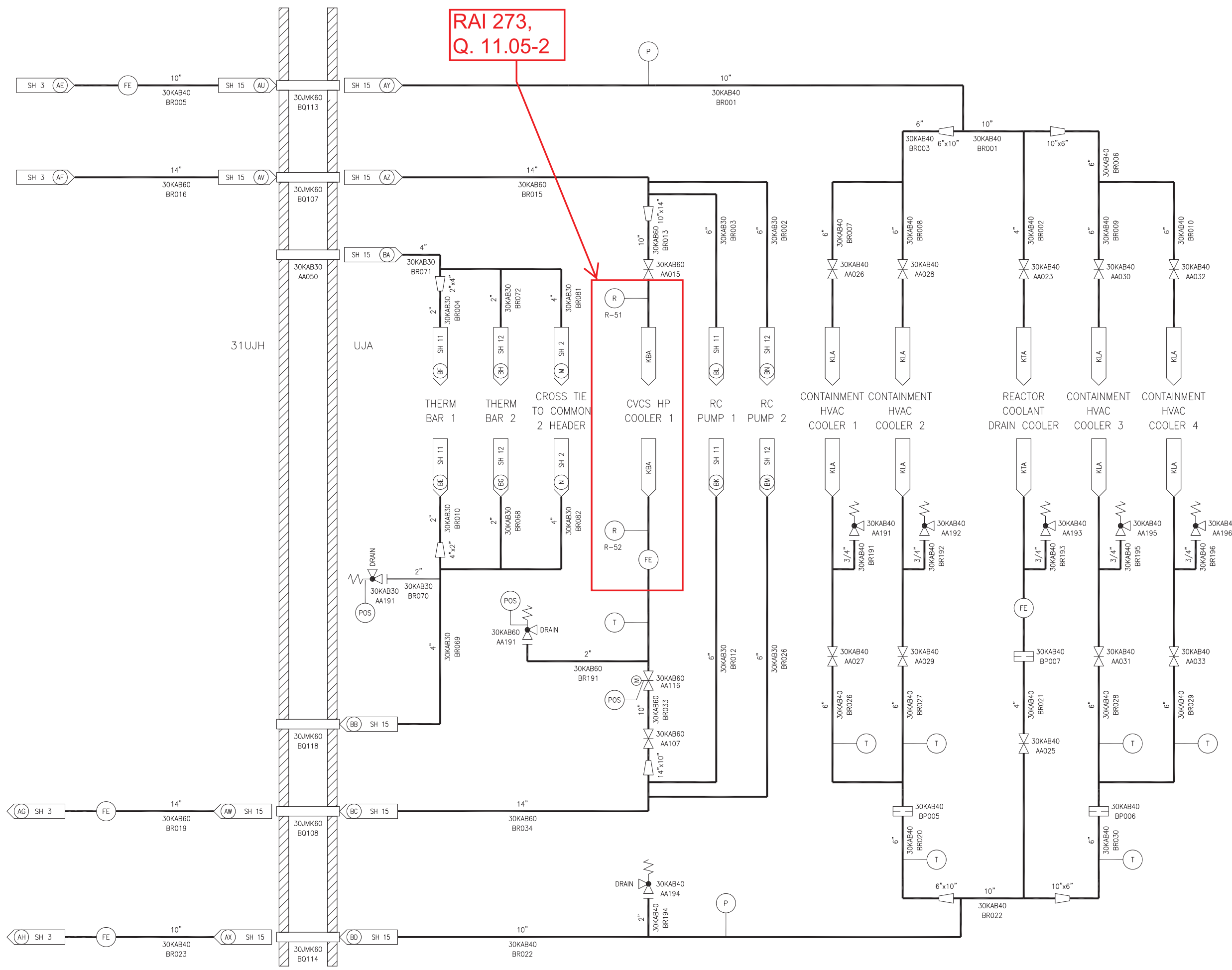


NOTES:
1. TRAIN 3 SHOWN IS REPRESENTATIVE OF 4 TRAINS WITH EXCEPTIONS NOTED.
2. TRAINS 1 AND 4 SAFETY INJECTION SYSTEM COOLING EXCEPTION.
3. SPOOL PIECE TO BE INSTALLED AS OPERATOR ACTION IN POST SEISMIC EVENTS AS REQUIRED.
4. FIRE WATER PIPING CONNECTION IS SEISMIC II.
5. STRAINER AND dP INSTRUMENTATION TO BE REMOVED AFTER COMMISSIONING.

E	C	15	225	I
G	E	175	140	NSC
C	C	175	225	I
B	C	175	225	I
DESIGN AREA	SSC QUALITY GROUP	DESIGN PRESSURE PSIG	DESIGN TEMPERATURE °F	SSC SEISMIC CLASS

REV 003
KAA01T2

Figure 9.2.2-2—Component Cooling Water System Common Loop 1
Sheet 2 of 7



JMK – PIPING PENETRATIONS
KAB – COMPONENT COOLING WATER SYSTEM COMMON HEADER
KBA – VOLUME CONTROL SYSTEM
KLA – CONTAINMENT BUILDING VENTILATION SYSTEM
KTA – NUCLEAR ISLAND DRAIN AND VENT SYSTEMS – PRELIMINARY EFFLUENTS

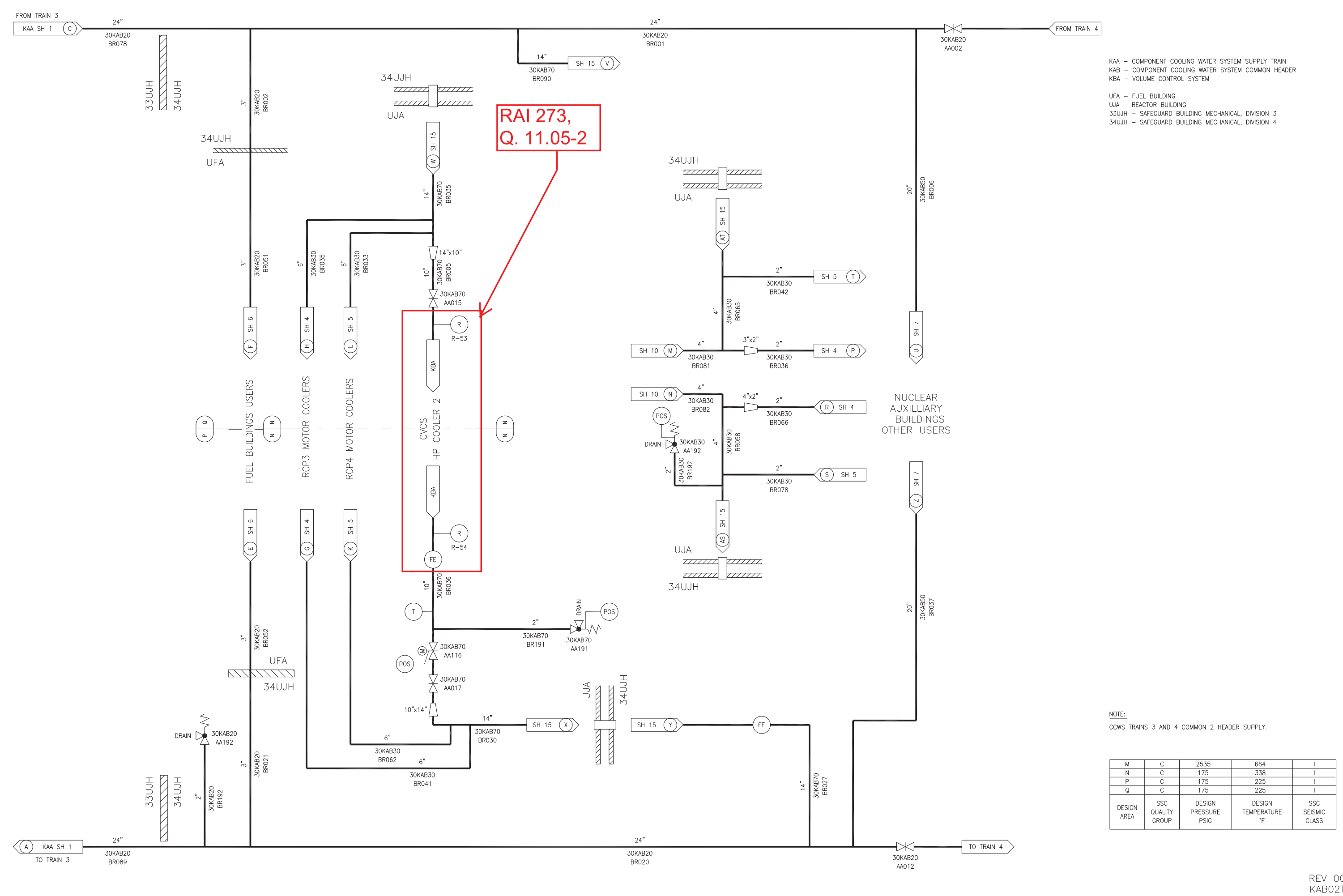
UJA – REACTOR BUILDING
31UJH – SAFEGUARD BUILDING MECHANICAL, DIVISION 1

NOTE: COMMON 1 HEADER RB USERS.

DESIGN AREA	SSC QUALITY GROUP	DESIGN PRESSURE PSIG	DESIGN TEMPERATURE °F	SSC SEISMIC CLASS
M	C	2535	664	I
N	C	175	338	I
P	C	175	225	I
Q	C	175	225	I

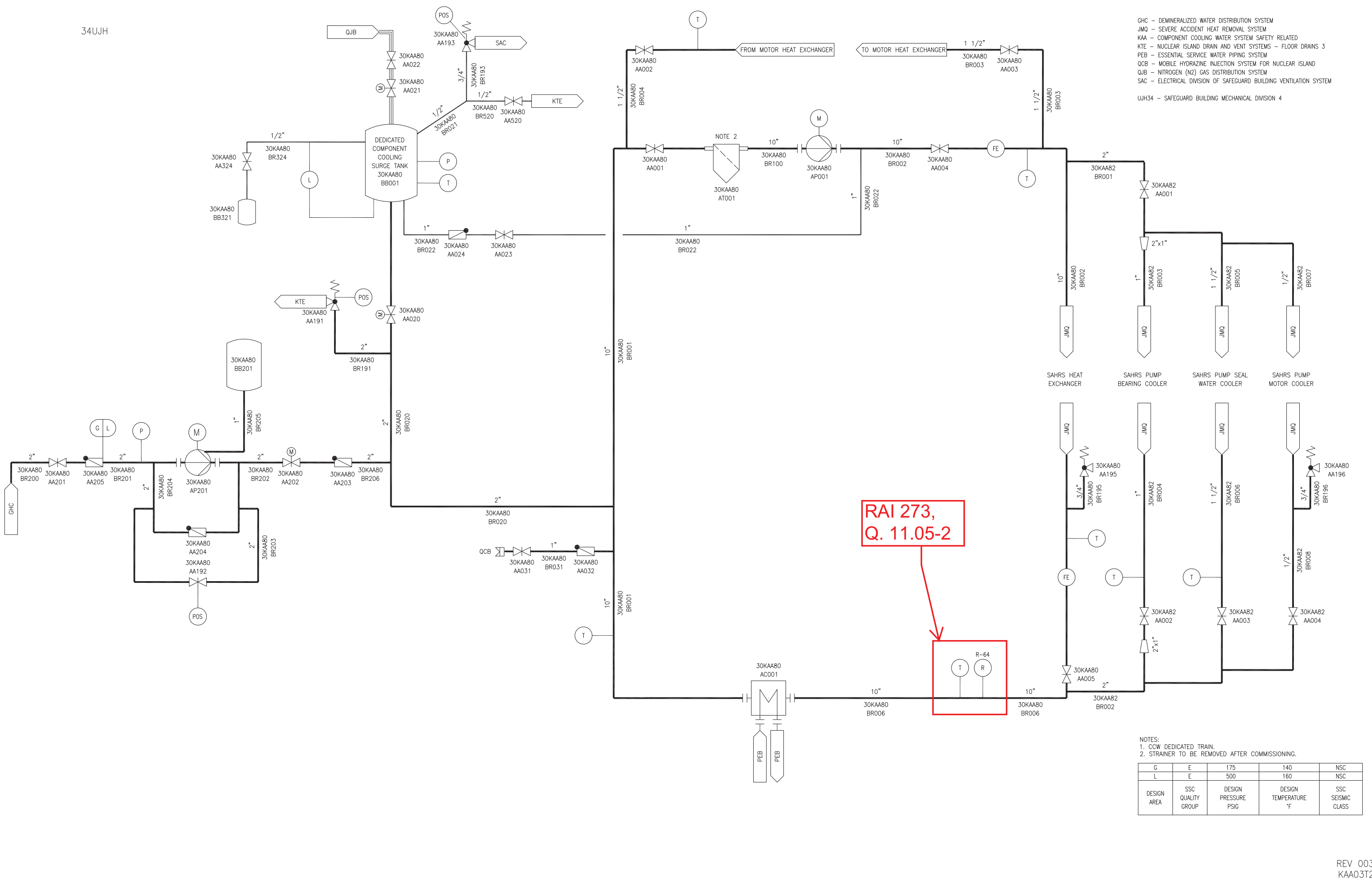
REV 003
KAB10T2

Figure 9.2.2-3—Component Cooling Water System Common Loop 2
Sheet 2 of 8



DESIGN AREA	SSC QUALITY GROUP	DESIGN PRESSURE PSIG	DESIGN TEMPERATURE °F	SSC SEISMIC CLASS
M	C	2535	664	I
N	C	175	338	I
P	C	175	225	I
Q	C	175	225	I

Figure 9.2.2-4—Component Cooling Water System Dedicated CCWS Trains



G	E	175	140	NSC
L	E	500	160	NSC
DESIGN AREA	SSC QUALITY GROUP	DESIGN PRESSURE PSIG	DESIGN TEMPERATURE °F	SSC SEISMIC CLASS

Next File

~~concept.~~ The four SCWS trains are backed up by the EDGs. Two of these trains, in Divisions 1 and 4, are also backed up by the SBO diesels.

- Structures, systems and components important to safety in the SCWS are not shared with any other co-located nuclear reactor units.
- Preoperational testing of the SCWS is performed as described in Chapter 14.0. Periodic inservice functional testing is done in accordance with Section 9.2.8.5.
- Section 6.6 provides the ASME Boiler and Pressure Vessel Code, Section XI (Reference 1) requirements that are appropriate for the SCWS.
- Section 3.2 delineates the quality group classification and seismic category applicable to the safety-related portion of this system. Table 9.5.4-1 shows that the components meet the design and fabrication codes given in Section 3.2. All the power supplies and control functions necessary for safe function of the SCWS are Class IE, as described in Chapter 7 and Chapter 8.
- Cooling diversity is created between the load heat sinks of ~~Divisions~~Train 1 and Train 4, and ~~Divisions~~Train 2 and Train 3. ~~Division~~Train 1 and Train 4 chillers are air cooled, and ~~Division~~Train 2 and Train 3 chillers are water cooled by the component cooling water system (CCWS).
- A process radiation monitor is provided in Trains 1 and 4 of the SCWS, downstream of the LHSI pump mechanical seal ~~heat exchanger~~cooler to monitor for possible leakage of radioactive fluid from the heat exchanger. Otherwise, migration of radioactive material from potentially radioactive systems is prevented with a minimum of two heat exchanger barriers. Radiation monitors are in the CCWS to detect radioactive contamination entering and exiting the system. Radiation monitors for the SCWS and the CCWS are specified in Table 11.5-1 Monitors R-59, R-60 (SCWS) and R-64 (CCWS).

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9.2.8.5 Inspection and Testing Requirements

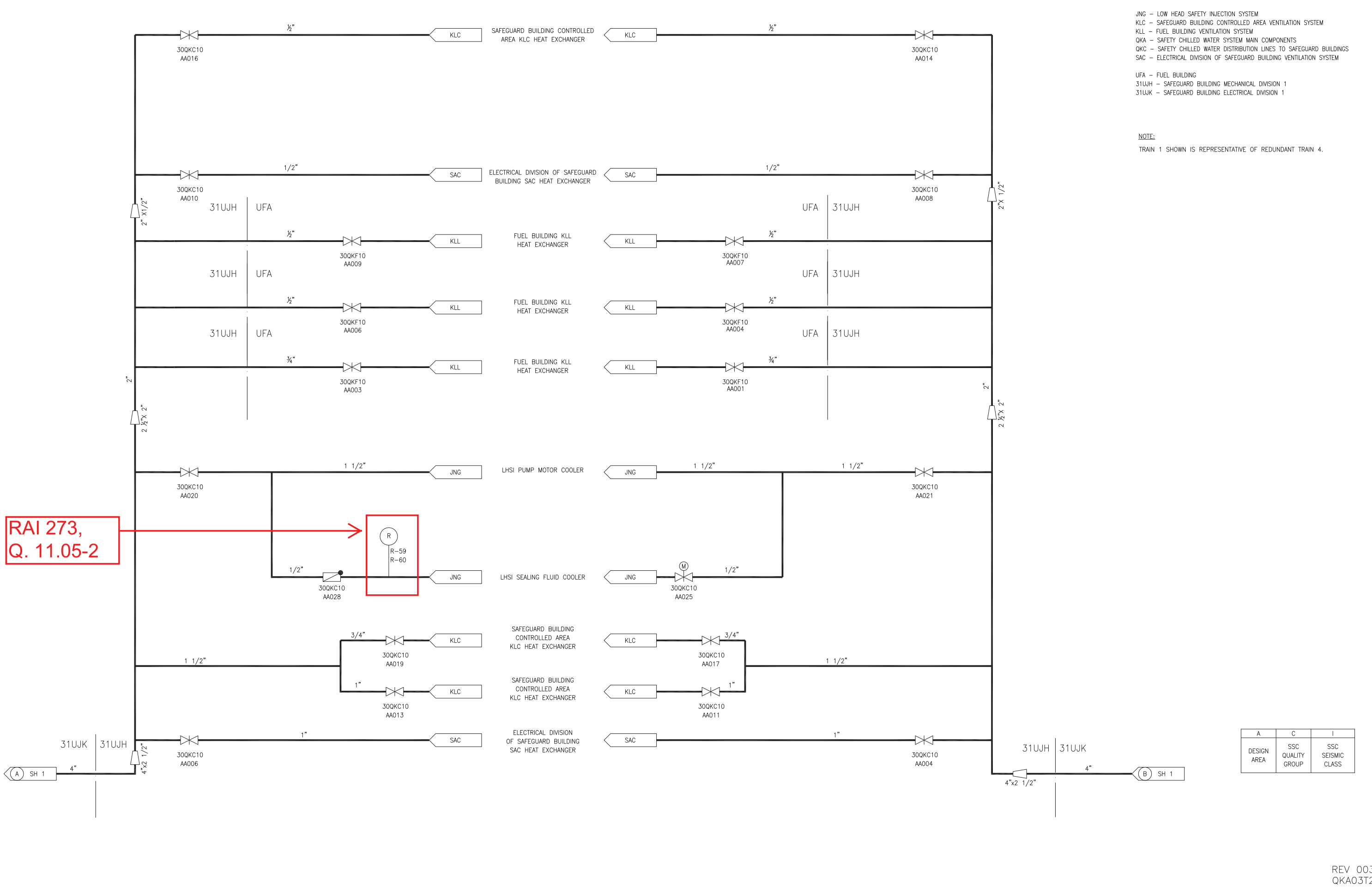
Prior to initial plant startup, a comprehensive performance test will be performed to verify that the design performance of the system and individual components is attained. Refer to Section 14.2, Test #052, for initial plant testing of the SCWS.

After the plant is brought into operation, periodic tests and inspections of the SCWS components and subsystems are performed to verify proper operation. Scheduled tests and inspections are necessary to verify system operability.

The installation and design of the SCWS provides accessibility for the performance of periodic inservice inspection and testing. Periodic inspection and testing of safety-related equipment verifies its structural and leak tight integrity and its availability and ability to fulfill its functions.

Inservice inspection and testing requirements are in accordance with Section XI of the ASME Code (Reference 1) and the ASME OM Code (Reference 2).

Figure 9.2.8-1—Safety Chilled Water System Diagram
Sheet 3 of 4



9.3.2 Process Sampling Systems

The process sampling systems comprise the following:

- Nuclear sampling system (NSS).
- Secondary sampling system (SECSS).
- Severe accident sampling system (SASS).
- Hydrogen monitoring system (HMS).

The HMS is described in U.S. EPR FSAR Tier 2, Section 6.2.5.2.2.

Activity monitoring associated with main steam are described in Sections 10.3.5.5 and 11.5.4.1.

These process sampling systems provide centralized and local facilities for obtaining liquid and gaseous samples for the purpose of determining the physical and chemical characteristics and control parameters by measurements and analyses. Samples are obtained from the following:

- Primary and secondary coolant.
- Containment atmosphere.
- Liquid and gaseous waste treatment systems.
- In-containment refueling water storage tank (IRWST).

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There is no shared equipment and there are no shared sample functions among the liquid waste management and gaseous waste management subsystems. Refer to Figure 11.2-1 and Figure 11.3-1. The liquid tanks are vented to a common header which discharges gases to the Radioactive Waste Processing Building Ventilation System and on to the vent stack. Liquid sampling is handled by sampling subsystem KUB while gaseous sampling is handled by sampling subsystem KUF. The gaseous waste management subsystem has residual radioactive liquid constituents from condensation and collected by the KTA system for processing in the LWS. There is no shared equipment or sampling.

The process sampling systems are a subset of the process and effluent monitoring system, which is described in Section 11.5.

9.3.2.1 Design Bases

The processing sampling systems perform the following safety-related functions:

- Non-safety-related portions of the process sampling systems are designed to monitor the fuel storage and radioactive waste systems and detect conditions that may result in excessive radiation levels (GDC 63).

- Non-safety-related portions of the process sampling systems include means for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident (LOCA) fluids, effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including AOOs and postulated accidents (GDC 64).

- Non-safety-related portions of the process sampling systems are designed to have provisions for a leakage detection and control program to minimize the leakage from those portions of the process sampling systems outside of the containment that contain or may contain radioactive material following an accident (10 CFR 50.34(f)(2)(xxvi)).

The process sampling systems are designed to meet the following functional criteria:

- Obtain liquid and gaseous samples from the primary coolant, liquid and gaseous waste treatment systems, auxiliary systems and inside containment.

- Purge sampling lines and reduce plateout (buildup of chemical residue) in sample lines, demonstrating compliance with RG 1.21, position C7. Sample lines are normally flushed/purged with sample media.

RAI 273, Q.11.05-2(4)

- Representative samples from gaseous process streams and tanks are in accordance with American National Standards Institute/Health Physics Society (ANSI/HPS) Standard N13.1-1999 (Reference 2). These criteria conform to RG 1.21, position C.6.

- Size RCS sample lines to minimize loss of reactor coolant following rupture of sample line.

- Recycle primary side samples according to their source to minimize waste.

- Continuously monitor secondary side activity and chemistry.

- Recycle secondary side samples to steam generator blowdown demineralizing system.

- Continuously monitor and obtain manual grab samples from selected points in the secondary side, main cycle and auxiliary systems.

- Sample glove boxes are purged by taking air from the room atmosphere. The air is passed through pre-filters, made available for purging, then discharged to the appropriate gaseous system.

- Maintenance and decontamination connections are either temporary or completely removed upon termination of the flush. Non-contaminated systems

Table 9.3.2-1—Primary Side Sampling Points
Sheet 1 of 3

Process or Equipment	Number of Sample Points	Type of Samples5	Process Measurement	Range4	Response Time4	Sensitivity4	Maximum anticipated drift between calibrations4
LHSI/RHR	4	Grab	See Notes 1,2				
CVCS	2	Continuous	Boron	0.015-20 ppb	6 minutes	± 3% or 0.005 ppb, whichever is greater	<1% of reading (per year)
			Hydrogen	0-10 ppm	20 seconds	± 3% or ± 60 ppb, whichever is greater	<1% of reading (per year)
			Oxygen	0-20,000 ppb	30 seconds	± 2 ppb	<1% of reading (per year)
			Conductivity	0.02-2,000 µS/cm	20 seconds	± 0.1 °F of temperature	<1% of reading (per year)
			Activity (gamma)	See Table 11.5-1, Monitor R-51 through R-54			
RCS	2	Continuous	Boron	0.015-20 ppb	6 minutes	± 3% or 0.005 ppb, whichever is greater	<1% of reading (per year)
			Hydrogen	0-10 ppm	20 seconds	± 3% or ± 60 ppb, whichever is greater	<1% of reading (per year)
			Oxygen	0-20,000 ppb	30 seconds	± 2 ppb	<1% of reading (per year)
			Conductivity	0.02-2,000 µS/cm	20 seconds	± 0.1 °F of temperature	± 0.1 °F of temperature
			Activity (Beta)	See Table 11.5-1, Monitor R-41			

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Table 9.3.2-1—Primary Side Sampling Points
Sheet 2 of 3

Process or Equipment	Number of Sample Points	Type of Samples5	Process Measurement	Range4	Response Time4	Sensitivity4	Maximum anticipated drift between calibrations4
Pressurizer	1	Continuous	Boron	0.015-20 ppb	6 minutes	± 3% or 0.005 ppb, whichever is greater	<1% of reading (per year)
			Hydrogen	0-10 ppm	20 seconds	± 3% or ± 60 ppb, whichever is greater	<1% of reading (per year)
			Oxygen	0-20,000 ppb	30 seconds	± 2 ppb	<1% of reading (per year)
			Conductivity	0.02-2,000 µS/cm	20 seconds	± 0.1 °F of temperature	<1% of reading (per year)
			Activity (Beta)	See Table 11.5-1, Monitor R-41			
CPS	6	Grab	See Notes 1,2,3	N/A	N/A	N/A	N/A
SIS Accumulators	4	Grab	See Notes 1,2,3	N/A	N/A	N/A	N/A
RBWMS (boric acid pump)	2	Grab	See Notes 1,2,3	N/A	N/A	N/A	N/A
FPCS	2	Grab	See Notes 1,2,3	N/A	N/A	N/A	N/A
FPPS	2	Grab	See Notes 1,2,3	N/A	N/A	N/A	N/A
CDS	2	Grab	See Notes 1,2,3	N/A	N/A	N/A	N/A
CTS	2	Grab	See Notes 1,2,3	N/A	N/A	N/A	N/A

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Table 9.3.2-1—Primary Side Sampling Points
Sheet 3 of 3

Process or Equipment	Number of Sample Points	Type of Samples5	Process Measurement	Range4	Response Time4	Sensitivity4	Maximum anticipated drift between calibrations4
GWPS			Activity (gamma and beta)	See Table 11.5-1, Monitors R-1 and R-2			
	4	Continuous/ Grab	See Notes 1,2,3	N/A	N/A	N/A	N/A
CSSS	1	Grab	See Notes 1,2,3	N/A	N/A	N/A	N/A
NIDVS (primary effluents)	3	Grab	See Notes 1,2,3	N/A	N/A	N/A	N/A
NSS (backfeed tank)	1	Grab	See Notes 1,2,3	N/A	N/A	N/A	N/A
Severe Accident Sampling System (SASS)	2	Grab	See Note 6	N/A	N/A	N/A	N/A

Notes:

1. Specific properties of liquid and gaseous grab samples to be measured are identified in plant procedures.
2. Laboratory instruments used to measure grab samples are identified in plant procedures for noble gas, iodine, and aerosols. Manufacturer's operating manual shall be consulted for calibration, measuring, maintenance, and cleaning and storage requirements.
3. Calibrating gases and solution for instrumentation are per the manufacturer's operating manual.
4. Values are typical; the actual values will meet or exceed the listed parameters.
5. Continuous on-line monitoring systems will be in accordance with ASTM D 3864 and ANSI N42.18-2004.

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6. Contains containment atmosphere and IRWST (see Section 7.5.2.1.1).

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Table 9.3.2-2—Secondary Side Sampling Points
Sheet 1 of 5

Process or Equipment	Number of Sample Points	Type of Samples[5]	Process Measurement of Continuous Samples	Range[4]	Response Time[4]	Sensitivity[4]	Maximum Anticipated Drift between Calibrations[4]
SG Blowdown	15	Continuous/grab	Activity (gamma)	See Table 11.5-1, Monitor R-46 through R-49			
			Cation Conductivity	0.06 to 1000 µS/cm	30 seconds	0.001 to 1 µS/cm (depends on range)	± 1% of measured value or ± digit (whichever is greater)
			Specific Conductivity	0.06 to 1000 µS/cm	30 seconds	0.001 to 1 µS/cm (depends on range)	± 1% of measured value or ± digit (whichever is greater)
			Sodium	0.1 – 1000 ppb	2 minutes	±0.1 ppb	<1% of reading (per year)
			pH	1-11 pH	30 seconds	±0.02 pH	<1% of reading (per standard solution volume)
Feedwater (upstream valve chamber)	1	Continuous/grab	Specific Conductivity	0.06 to 1000 µS/cm	30 seconds	0.001 to 1 µS/cm (depends on range)	± 1% of measured value or ± digit (whichever is greater)
			Cation Conductivity	0.06 to 1000 µS/cm	30 seconds	0.001 to 1 µS/cm (depends on range)	± 1% of measured value or ± digit (whichever is greater)
			pH	1-11 pH	30 seconds	±0.02 pH	<1% of reading (per standard solution volume)
			Oxygen	0-20,000 ppb	30 seconds	± 2 ppb	<1% of reading (per year)

Table 9.3.2-2—Secondary Side Sampling Points
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Process or Equipment	Number of Sample Points	Type of Samples5	Process Measurement of Continuous Samples	Range4	Response Time4	Sensitivity4	Maximum Anticipated Drift between Calibrations4
Feedwater Pumps discharge (combined with start-up feedwater pump)	1 per pump (4 total)	Continuous/grab	Hydrazine	0-200 ppb	4 minutes	±2 ppb	<1% of reading (per year)
			Specific Conductivity	0.06 to 1000 µS/cm	30 seconds	0.001 to 1 µS/cm (depends on range)	± 1% of measured value or ± digit (whichever is greater)
			Cation Conductivity	0.06 to 1000 µS/cm	30 seconds	0.001 to 1 µS/cm (depends on range)	± 1% of measured value or ± digit (whichever is greater)
			pH	1-11 pH	30 seconds	±0.02 pH	<1% of reading (per standard solution volume)
			Oxygen	0-20,000 ppb	30 seconds	± 2 ppb	<1% of reading (per year)
			Sodium	0.1 – 1000 ppb	2 minutes	±0.1 ppb	<1% of reading (per year)
Main Steam (upstream HP turbine)	4	Continuous/grab	Cation Conductivity	0.06 to 1000 µS/cm	30 seconds	0.001 to 1 µS/cm (depends on range)	± 1% of measured value or ± digit (whichever is greater)
			Degassed Cation Conductivity	0.06 to 1000 µS/cm	30 seconds	0.001 to 1 µS/cm (depends on range)	± 1% of measured value or ± digit (whichever is greater)
			Sodium	0.1 – 1000 ppb	2 minutes	±0.1 ppb	<1% of reading (per year)
Main Steam Lines (upstream of HP turbine)	4	Continuous	Activity (gamma)	See Table 11.5-1, Monitors R-55 through R-58			

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Table 9.3.2-2—Secondary Side Sampling Points Sheet 5 of 5

Process or Equipment	Number of Sample Points	Type of Samples5	Process Measurement of Continuous Samples	Range4	Response Time4	Sensitivity4	Maximum Anticipated Drift between Calibrations4
HP Heater Drains	2	Grab	See Notes 1,2,3	N/A	N/A	N/A	N/A
LP Heater Drains	2	Grab	See Notes 1,2,3	N/A	N/A	N/A	N/A
Clean Drains	1	Grab	See Notes 1,2,3	N/A	N/A	N/A	N/A
Demineralized Water Storage Tank	1	Grab	See Notes 1,2,3	N/A	N/A	N/A	N/A
Closed Cooling Water System	1	Grab	See Notes 1,2,3	N/A	N/A	N/A	N/A
Circulating Water Cooling Water Basin Makeup	1	Continuous	pH	1-11 pH		±0.02 pH	<1% of reading (per standard solution volume)
			Specific Conductivity	0.06 to 1000 µS/cm	30 seconds	0.001 to 1 µS/cm (depends on range)	± 1% of measured value or ± digit (whichever is greater)
Turbine Building Drains	1	Continuous/ Grab	See Notes 1,2,3	N/A	N/A	N/A	N/A
			Activity (gamma)	Table 11.5-1, Monitor R-50			
Vent System for Air Removal	1 Per System	Continuous	Activity (gamma)	Table 11.5-1, Monitor R-3			

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Notes:

1. Specific properties of liquid and gaseous grab samples to be measured are identified in plant procedures for noble gas, iodine, and aerosols.
2. Laboratory instruments used to measure grab samples are identified in plant procedures. Manufacturer's operating manual shall be consulted for calibration, measuring, maintenance, and cleaning and storage requirements.
3. Calibrating gases and solution for instrumentation are per the manufacturer's operating manual.
4. Values are typical; the actual values will meet or exceed the listed parameters.
5. Continuous on-line monitoring systems will be in accordance with ASTM D 3864 and ANSI N42.18-2004.

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Next File

Figure 9.3.2-1—Nuclear Sampling System
Sheet 2 of 9

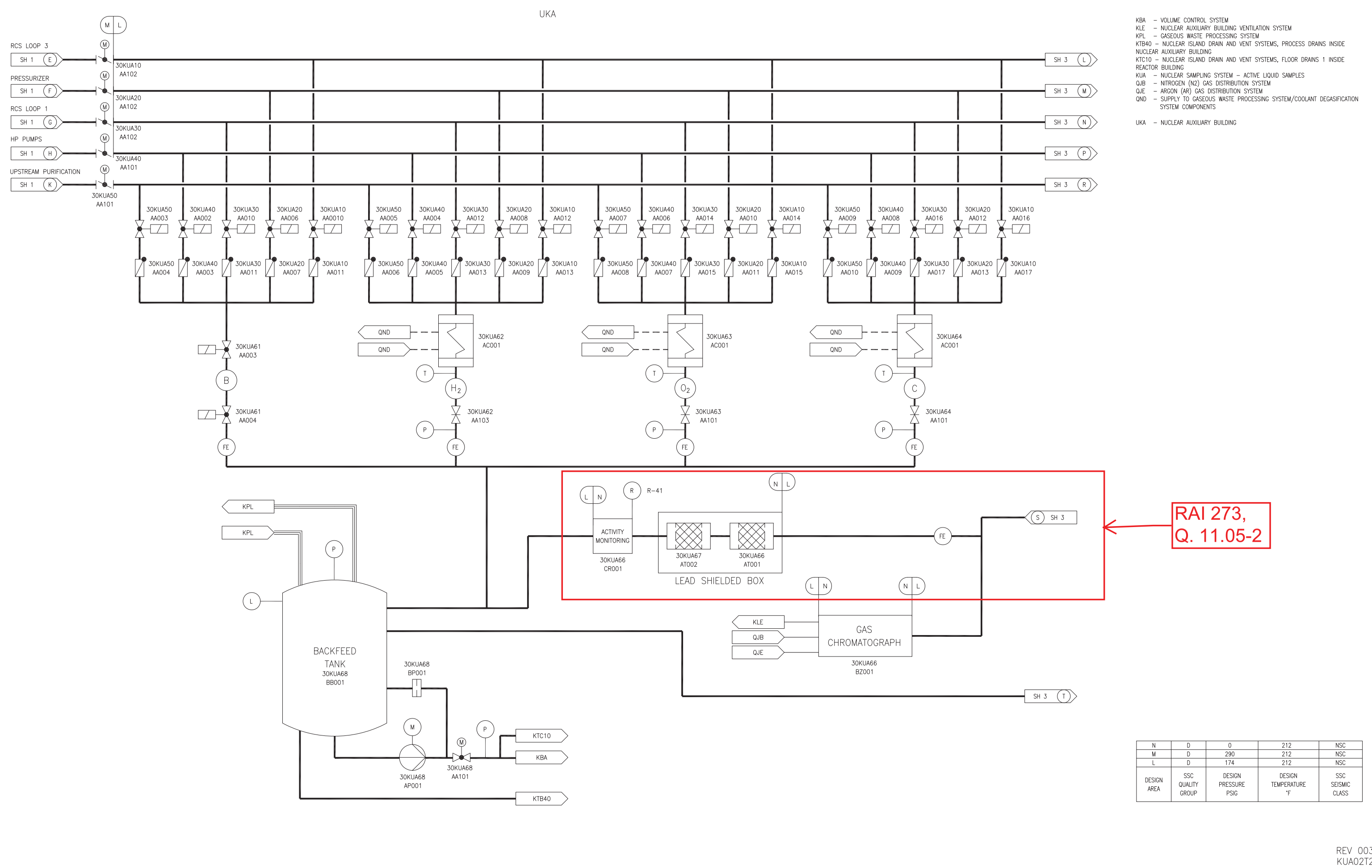
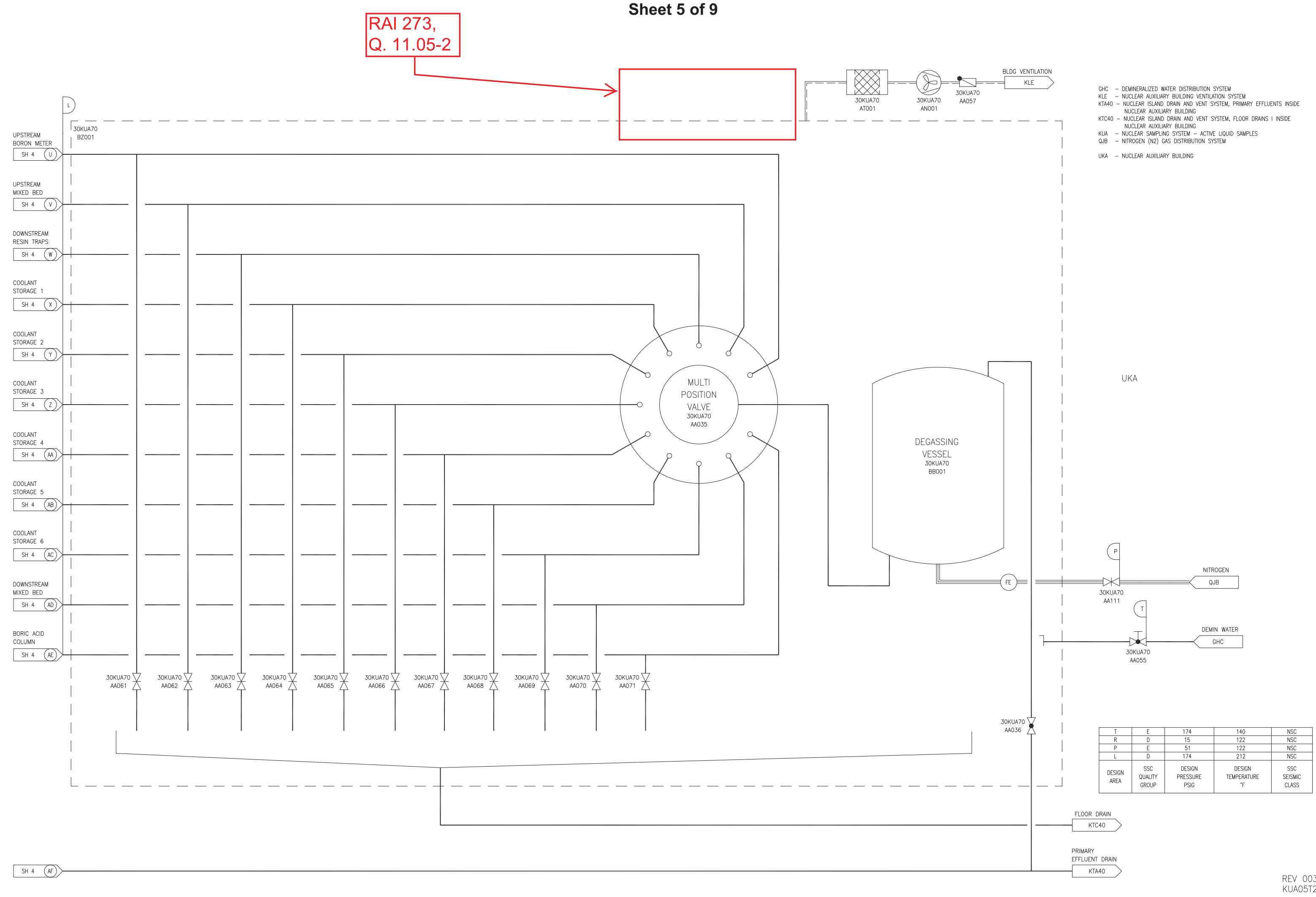


Figure 9.3.2-1—Nuclear Sampling System
Sheet 5 of 9



- Alarm interlocking: initiated by the VCT low water level.
- Sump high water level in the FB vent and drain system.
- Increased activity measurements in the exhaust air ducts of the FB ventilation system for noble gas, airborne, and iodine radiation monitors (refer to Section 9.4.2 and 11.5.3.1.7 and Table 11.5-1, Monitors R-17 and R-18).

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A manual isolation of the letdown line after 30 minutes results in an inventory loss of a maximum of 9625 gallons of coolant. When the break is identified, closing any one of the four HP isolation valves in series isolates the letdown flow.

In the event of a line break upstream of the check valves, redundant check valves close preventing backflow from the charging line. Also, a low charging pump discharge pressure trips the charging pumps, which terminates the charging flow.

Tube Rupture in the HP Cooler - The CCWS cools the CVCS HP coolers in the RB. An HP cooler tube rupture results in a leak from the CVCS into the component cooling system. If the leak occurs during plant operation, the differential pressure inside to the outside of the tubes is approximately 2160 psi (CVCS pressure – CCWS pressure). In this event, the tube break in the CVCS HP cooler results in a leak of reactor coolant into the component cooling water.

The opening of the component cooling relief valve protects the CCWS piping and CIVs and prevents the overpressurization of the CCWS. An increased flow from the CCWS flow meters or an increase in radioactivity measured by detectors in the component cooling water inlet and outlet to the cooler can indicate the leak. This high activity measurement generates a signal to automatically close the cooler isolation valves to isolate the CVCS HP cooler.

Postulated System Leaks in Containment - In the event of a leak in the CVCS or RCP seal water system, reactor coolant with temperatures between approximately 120°F and 565°F is released. This leakage can be detected by activity measurements (area dose rate monitoring system) inside containment, and also by the CVCS pressure, temperature and flow conditions and the pressurizer level.

Postulated System Leaks in the Fuel Building - In the event of a CVCS or RCP seal water system leak in the Fuel Building, reactor coolant with temperatures of approximately 120°F is released.

Due to the loss of reactor coolant, the following alarms are also generated:

- VCT low water level.
- Sump high water level in the FB vent and drain system.

- GDC 4, as it relates to the CRACS by design, to protect against adverse environmental conditions and dynamic effects. The CRACS accommodates the effects of, and is compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

- GDC 5, as it relates to the CRACS system because safety-related components are not shared with any other nuclear power units.

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- GDC 19, as it relates to the CRACS system to provide adequate protection against radiation releases and outside fire or smoke events~~and hazardous chemical releases~~ to permit access to and occupancy of the control room under accident conditions. The control room occupancy protection requirements meet the guidance of RG 1.78. RG 1.52 and 1.140 (GDC 60). In case of an alarm from the inlet air radiation monitors (refer to Section 11.5.3.1.1 and Table 11.5-1, Monitors R-29 and R-30), the CRACS directs the air intake automatically through activated carbon filtration beds. The air from CRE areas can also be recirculated through the same activated carbon filtration beds. The evaluation of potential toxic chemical accidents is addressed by the COL applicant in Section 2.2.3 and includes the identification of toxic chemicals. As described in Section 6.4.1, the COL applicant evaluates the impact of toxic chemical accidents on control room habitability in accordance with RG 1.78.

- GDC 60, as it relates to the release of radioactive materials to the environment.

Consideration of the environmental and dynamic effects of internal and external missiles and postulated piping failures on the CRACS is addressed in Section 3.5.1.1, Section 3.5.2, and Section 3.6.1.

Capability for withstanding or coping with a SBO event is provided to comply with the requirements of 10CFR 50.63. Acceptance is based on meeting the applicable guidance of RG 1.155, including position C.3.2.4. Refer to Section 8.4 for a description of the design features to cope with the SBO event.

The CRACS maintains habitability of the CRE areas during a site radiological ~~contamination~~ event ~~or toxic contamination of the environment~~ (Refer to Section 6.4).

During a postulated event, the control room is maintained at a minimum positive pressure of 0.125 inches water gauge relative to the surrounding environment to prevent uncontrolled incoming leakage.

During normal operation, the control room is maintained at a ~~minimum positive~~ pressure above ambient~~of 0.01 inches water gauge relative to the surrounding environment~~.

The CRACS maintains system performance in the event of failure of a single active safety-related component.

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The CRACS outside air intake is capable of detecting radiation~~, toxic gases (which include CO, CO_2)~~ (see Section 6.4.2.4), and smoke. Associated monitors actuate alarms in the MCR. Upon receipt of a containment isolation signal, or high radiation alarm signal in the outside air intake duct (Monitors R-29 and R-30, Table 11.5-1), the CREF (iodine filtration) train starts automatically and the outside air and CRE recirculation air are automatically diverted through the CREF (iodine filtration) train. The outside makeup air maintains a positive pressure inside the CRE area relative to the adjacent areas. The CRE air inlet and recirculation dampers operate automatically.

~~Upon receipt of a toxic gas alarm from a toxic gas sensor in the outside air intake duct, the CREF (iodine filtration) trains are placed in the filtered alignment mode and the CRE air is diverted to the recirculation mode on both trains. The outside air inlet isolation damper at the outside inlet in alarm will be closed by the control room operator.~~

The CRACS is capable of isolating all non-safety-related system penetrations of the CRE boundary so that occupation and habitability of the control room is not compromised.

Air conditioning and heating loads for the CRE rooms is calculated using methodology identified in ASHRAE Handbook (Reference 8) as follows:

- Summer air conditioning loads are calculated with a maximum outside air design temperature 0 percent exceedance value, using U.S. EPR Site Design Envelope temperature (See Table 2.1-1). The analysis is completed for both a normal and accident plant alignment configuration.
- The CRACS cooling supply units are designed to provide cooling as required to prevent the CRE room temperatures from exceeding their maximum design temperature.
- Winter heating loads are calculated with the plant operating in an outage alignment configuration. Winter heat loads are calculated with a minimum outside air design temperature 0 percent exceedance value, using U.S. EPR Site Design Envelope temperature (See Table 2.1-1).

The CRACS supply air duct heaters are designed to operate for "comfort conditions only" as required when the CRE room temperature is less than the minimum "comfort temperature" set point value. The CRACS supply air duct heaters are not required to operate during accident conditions.

The CRACS maintains the following temperature ranges for the areas serviced:

Room	Temperature	Humidity
– Main Control Room:	68°F to 78°F	30 – 60%

–	I&C Computer Rooms, Rest Rooms:	65°F to 78°F	30 – 60%
–	HVAC Rooms:	50°F to 95°F	30 – 60%
–	Other areas of CRE:	65°F to 7978°F	20 – 80%

9.4.1.2 System Description

9.4.1.2.1 General Description

The CRACS is designed to maintain acceptable ambient conditions inside the CRE areas to provide for proper operation of equipment and for personnel access to conduct inspection, testing and maintenance. The CRE area is shown in Figures 6.4-1 through 6.4-3.

The CRACS consists of following subsystems:

- Air intake.
- ICREF (iodine filtration) train.
- RAir conditioning and recirculation air handling.
- CRE air supply and recirculation.
- Kitchen and restrooms exhaust.

Refer to Section 12.3.6.5.6 for ventilation system design features which demonstrate compliance with the requirements of 10 CFR 20.1406.

Air Intake Subsystem

The air intake subsystem is illustrated in Figure 9.4.1-1—Control Room Air Intake and CREF (Iodine Filtration) Train Subsystem.

The control room air conditioning systemCRACS has two outside air intakes. The train 1 intake is located in Safeguard Building 2 and the train 4 intake is located in Safeguard Building 3. Outside air is supplied by each outside air intake through a wire mesh grille. Each outside air intake is equipped with an electrically heated, weather protected grille to prevent ice formation. A sensor is installed in each outside air intake to protect against toxic gas (refer to Section 6.4.2.4), while sSmoke detectors and radiation monitors (refer to Section 11.5.3.1.11 and Table 11.5-1, Monitors R-29 and R-30) are installed in the outside air intake ducting.

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Outside air intakes on each train isare interconnected through ducting to allow the outside inlet air to travel through a CREF iodine filtration unit (filtered alignment), or the outside air can bypass the CREF iodine filtration unit (unfiltered bypass alignment).

Figure 9.4.1-1—Control Room Air Intake and CREF (Iodine Filtration) Train Subsystem

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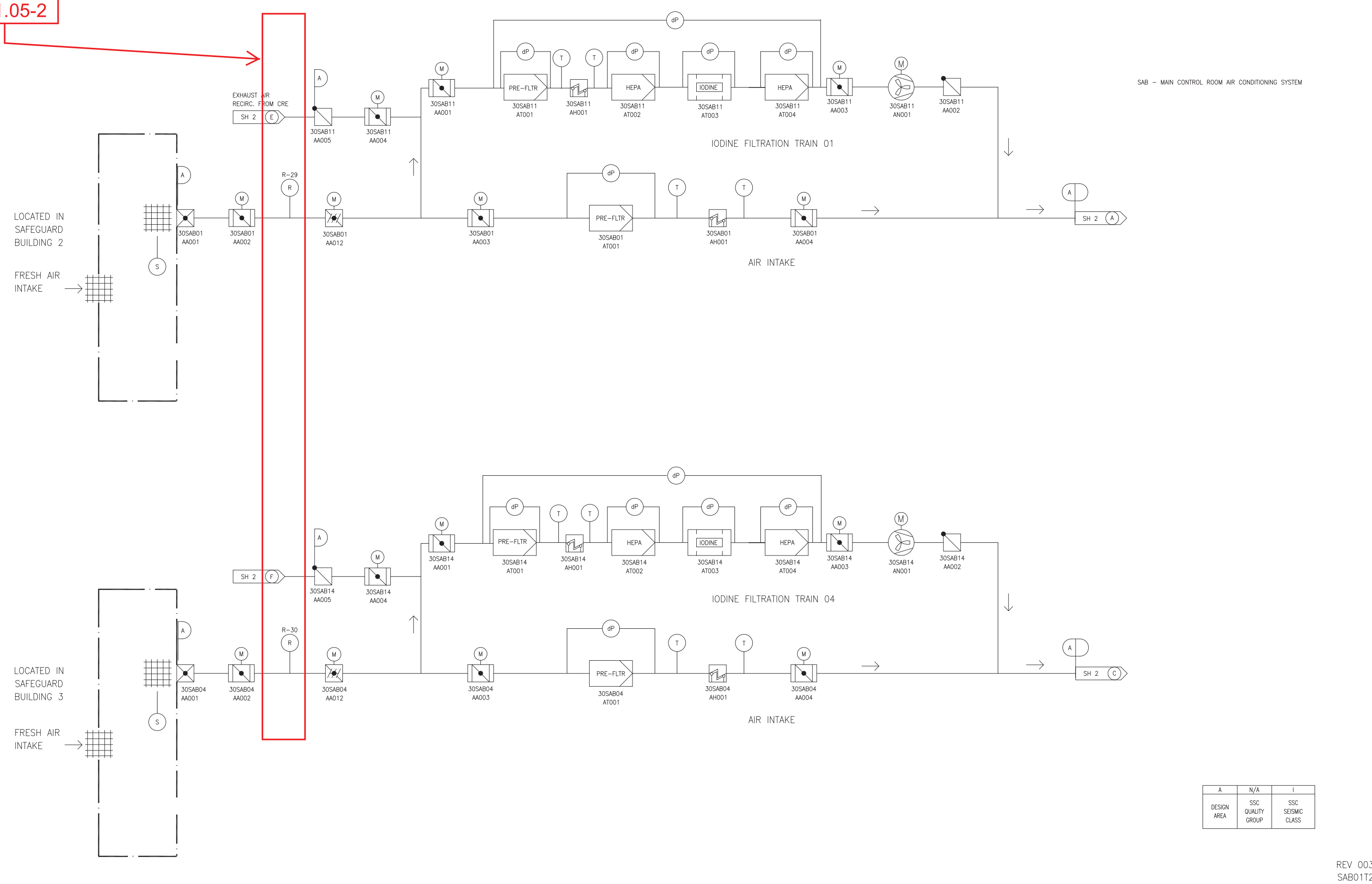

The seismic design of the system components meets the guidance of RG 1.29 (Position C.1 for the safety-related portion and Position C.2 for the non-safety-related portion). ~~The quality group classification and Seismic Category of system components meet the requirements of RG 1.26 and 1.29.~~Table 3.2.2-1 provides the seismic design and other design classifications for components in the FBVS.

The safety-related components and systems of the FB ventilation system are not shared among nuclear power units (GDC-5).

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The release of radioactive material to the environment is controlled by meeting the guidance of RG 1.140, positions C.2 and C.3 (GDC-60). RG 1.52 is not applicable because the FBVS is not required to operate during post-accident engineered safety features (ESF) atmospheric cleanup. In case of high radiation alarm in the FB (refer to Table 11.5-1, Monitors R-17 and R-18) ~~radioactive contamination~~, the system will automatically direct the building exhaust through activated charcoal filtration beds located in the NABVS.

The FBVS provides appropriate ventilation and filtration to limit potential release of airborne radioactivity to the environment from the fuel storage facility under normal operation and in the event of a fuel handling accident in the fuel pool area. The design of the ventilation system meets the guidance of RG 1.13, Position C.4 (GDC-61).

Air conditioning and heating loads for the FB Rooms are calculated using methodology identified in ASHRAE Handbook (Reference 3).

- Summer air conditioning loads will be calculated with a maximum outside air design temperature 0 percent exceedance value, using U.S. EPR Site Design Envelope Temperature (See Table 2.1-1). The analysis will be completed for both a normal and accident plant alignment configuration.

- The cooling supply units are designed to provide cooling as required to prevent the FB room temperatures from exceeding their maximum design temperature.

- Winter heating loads will be calculated with the plant operating in an outage alignment configuration. Winter heat loads will be calculated with a minimum outside air design temperature 0 percent exceedance value, using U.S. EPR Site Design Envelope Temperature (See Table 2.1-1).

The FBVS provides the following important non-safety-related functions:

- Controls and maintains a negative pressure during normal operation within the FB relative to the outside environment. Rooms identified as having possible radioactive contamination are designed to be at a negative pressure relative to the adjacent rooms to make sure air flows from areas of low radioactivity to areas of potentially higher radioactivity.

The supply air is the conditioned outside air that is filtered, cooled or heated, humidified by the NABVS, and delivered to the FB rooms through the FBVS supply duct network.

The FBVS exhaust system is designed to limit spread of the airborne contaminants and to maintain a negative pressure in the FB with respect to the outside environment. The FBVS exhaust is processed through the filtration trains of the NABVS prior to discharge through the vent stack. The FBVS is divided into two subsystems referred to as Cell 4 and Cell 5. The cells separate the ventilation systems serving the redundant systems in the FB and each cell serves approximately half of the building. The supply and exhaust duct branches to each room are fed from the main supply and exhaust HVAC shafts in the building. These HVAC shafts are connected to the NABVS.

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If high radiation is detected within the FB (Monitors R-17 and R-18, Table 11.5-1), the exhaust air is diverted to the iodine filtration trains of the NABVS prior to discharge through the vent stack (refer to Section 9.4.3, Section 11.5.3.1.7, and Table 11.5-1, Monitors R-17 and R-18).

Isolation dampers are provided to isolate the supply and exhaust ducts of the room in front of the equipment hatch, fuel pool area, and the room in front of the emergency airlock.

Isolation dampers are also provided to isolate the FB from NABVS supply and exhaust ducts.

Electric heaters are provided for heating of the boron rooms and the rooms surrounding the extra borating system tanks to avoid boron crystallization in borating system piping. Electric heaters are also provided for the fuel pool room to prevent condensation on the walls, and other selected rooms to maintain room ambient conditions.

Recirculation cooling units are provided in the fuel pool cooling pump rooms and extra-borating system pump rooms to limit the maximum room temperature, allowing proper operation of the equipment in these rooms.

9.4.2.2.2 Component Description

The major components of the FBVS are described as follows. ~~Refer to Section 3.2 for the seismic and system quality group classification for these components~~Table 3.2.2-1 provides the seismic design and other design classifications for components in the FBVS. Individual codes and standards applicable to each component are also listed in the following paragraphs.

9.4.2.2.3 System Operation

Normal Plant Operation

During normal plant operation, fresh conditioned air is supplied to the FB rooms by the FBVS supply duct network. The supply air to the FB is provided by the NABVS. The room air conditioning is provided by the supply and exhaust air flows based on the minimum required air renewal rate, equipment heat load, and heat balance between the rooms. The air is heated or cooled to maintain the required ambient conditions of the rooms.

During normal operation, isolation dampers are open to provide ventilation of the FB. These isolation dampers also can be controlled by the NABVS.

During normal operation, system fire dampers are in the open position.

A negative pressure is maintained in the FB relative to the outside environment by regulating the FBVS supply and exhaust flows. A negative pressure is also maintained for rooms having the potential for radioactive contamination (principally due to iodine) relative to the adjacent rooms to provide air flows from areas of low radioactivity to areas of potentially higher radioactivity.

Electrical heaters operation is controlled by temperature sensors in the boron rooms and the fuel pool rooms. Non-safety-related electrical heaters are operated as needed, depending on the room temperatures.

Recirculation cooling units are used for fuel pool cooling system pump rooms, and extra-borating system pump rooms to make sure that acceptable temperatures are maintained within the rooms for proper operation of the components and safe personnel access. The recirculation cooling units for the fuel pool cooling system pump rooms operate when the pumps are in operation. The recirculation cooling units for the extra-borating system pump rooms operate based on room temperature to provide recycled cool air.

During plant outages, the supply and exhaust ducts of the room in front of the equipment hatch are isolated so that the air flow is from the FB to the Reactor Building (RB). When the equipment hatch is opened, this room is considered as part of the RB and is therefore ventilated by the RB ventilation system.

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In the event radioactive contamination is detected in the FB (refer to Table 11.5-1, Monitors R-17 and R-18) during normal operation, or a potential airborne radioactive hazard exists during maintenance of equipment or systems, the exhaust air is diverted to iodine filtration trains of the NABVS prior to discharge through the vent stack (refer to Section 11.5.3.1.7 and Table 11.5.1, Monitors R-17 and R-18). Iodine activity is detected separately in each cell.

Abnormal Operating Conditions

Failure of Supply and Exhaust Air

The FBVS supply and exhaust air systems are non-safety related. Failure of supply and exhaust air systems in the NABVS will lead to the loss of supply and exhaust functions of FBVS. In this case, negative pressure with respect to the outside atmosphere and room temperatures of the FB cannot be maintained; however, the recirculation cooling units and heaters will maintain acceptable temperatures in the fuel pool cooling and extra borating system pump rooms.

Failure of Heaters and Recirculation Cooling Units

In each room provided with safety-related heaters, two 100 percent capacity heaters are provided to fulfill the single failure criteria of the heaters. For heaters serving a safety-related function, the required power has been calculated based on failure of an electrical division. Thus, failure of one electrical division will not prevent other divisions from supplying power and fulfilling their functions.

Failure of one recirculation cooling unit will lead to the loss of cooling in the corresponding room. As a result, the extra borating and fuel pool cooling system pumps located in that room may not operate properly. Redundant extra borating and fuel pool cooling system pumps located in a separate room and served from a separate train will, however, still be operational.

Failure of Isolation Dampers

For safety-related isolation functions, automatic isolation is provided in the design by placing two dampers in series, with power for each damper supplied by a different electrical division. Failure of one electrical division thus does not hinder the isolation function of the system.

Fuel Handling Accident in the Fuel Building

In the event of a fuel handling accident in the FB, the air exhaust and supply of the space above the fuel pools are isolated by closing the isolation dampers serving this room. This occurs automatically by the sampling activity monitoring system signal. Alternatively, this isolation also can be performed via local push buttons located in the fuel pool room.

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To prevent spread of airborne contamination, the iodine filtration trains of the safeguard building ventilation are used to process the exhaust air and to maintain the required pressure in the FB fuel pool hall (refer to Section 9.4.5, Section 11.5.3.1.7, and Table 11.5-1, Monitor R-19). The remainder of the FB is ventilated by the NABVS.

Fuel Handling Accident in the Containment Building

In the event of a fuel handling accident in the Containment Building, to preclude uncontrolled migration of contamination, the FB areas in front of the emergency airlock and in front of the equipment hatch are isolated by closing the air exhaust and supply dampers dedicated to these areas.

Prior to opening the emergency airlock during an outage, the air exhaust in front of the emergency airlock is isolated by closing the dampers dedicated to this area.

Prior to opening the equipment hatch during an outage, the air supply and exhaust for the equipment area in front of the hatch are isolated by closing the dampers dedicated to this area.

Loss of Coolant Accident (LOCA)

In the event of LOCA, the containment isolation signal or high radiation signal in the RB initiates isolation of the FB from NABVS supply and exhaust duct to limit leakage into the FB. The SBVS maintains negative pressure in the FB and exhaust air from the FB is directed to the SBVS iodine filtration trains.

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Loss of Offsite Power (LOOP)

Upon loss of offsite power, all motorized dampers will fail ~~to the "close" position~~as is, limiting pathways for potentially contaminated air to leak out to the environment.

The following equipment will remain operational during LOOP:

- Electric heaters in the extra borating pump rooms and pipe chase.
- Recirculation cooling units in the fuel pool cooling system pump rooms, and extra borating system pump rooms.
- Dampers for isolating the fuel pool room and FB.

The power for the equipment listed above is supplied from the corresponding emergency diesel generators.

Station Blackout (SBO)

In the event of SBO, the following equipment will remain operational:

- Electric heaters in the extra borating system pump rooms and pipe chase.
- Isolation dampers for the fuel pool room so that the dampers can be closed in the event of high temperature in the fuel pool.

The power for the equipment listed above is supplied from the SBO emergency diesel generators (SBODG).

9.4.2.3 Safety Evaluation

The FBVS provides the following safety-related functions:

- Automatic isolation of the supply and exhaust air to the fuel handling hall in order to mitigate the consequences of a fuel handling accident in the hall.
- Manual isolation from the main control room (MCR) of the supply and exhaust airflow to the hall in front of the equipment hatch prior to opening of the hatch. This isolation mitigates the consequences of a fuel handling accident in the RB with the hatch opened.
- Automatic isolation of the supply airflow to the room in front of the emergency airlock in order to mitigate the consequences of a fuel handling accident in the RB. The isolation of the exhaust airflow from the room in front of the emergency airlock is performed manually from the MCR prior to opening of the emergency airlock.

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- Automatic isolation of the FB from NABVS supply and exhaust ducts in the event of containment isolation signal ~~or high radiation signal in the RB~~. The SBVS maintains negative pressure in the FB and filters the FB atmosphere through SBVS iodine filtration trains.
- Maintains ambient conditions in the extra borating system pump rooms and pipe chase and the fuel pool cooling system pump rooms during normal, abnormal, and postulated accident events.
- Safety-related components can function as required with failure of a single active component. The safety-related redundant components are powered from different electrical divisions so that the system can remain operable in case of failure of one of the electrical divisions.

9.4.2.4 Inspection and Testing Requirements

The FBVS major components, such as dampers, cooling units, heaters, and ducts are located to provide access for initial and periodic testing to verify their integrity.

Test and analysis will be completed during normal operation with the system operating in an accident alignment. Analysis will use as-built information from equipment to extrapolate the performance of the air-conditioning system. Analysis will show that the equipment performance is adequate to maintain design conditions during plant operating conditions.

Initial in-place acceptance testing of the FBVS is performed as described in Section 14.2 (test abstracts #081 and #203), Initial Plant Test Program, to verify the system is build in accordance with applicable programs and specifications.

The FBVS is designed with adequate instrumentation and temperature indicating devices to enable testing and verification of equipment function and heat transfer capability.

During normal plant operation, periodic testing of FBVS is performed to demonstrate system and component operability and integrity.

Isolation dampers are periodically inspected and damper seats replaced as required.

Recirculation cooling units are tested by manufacturer in accordance with Air Movement and Control Association (AMCA) standards (References 4, 5, and 6). Cooling coils are hydrostatically tested in accordance with ASME AG-1 (Reference 1) and their performance is rated in accordance with the Air Conditioning and Refrigeration Institute (ARI) standards (Reference 7).

Ductwork is leak-tested in accordance with the Sheet Metal and Air Conditioning Contractors' National Association (SMACNA) technical manual "HVAC Air Duct Leakage Test Manual" (Reference 8), American Society of Mechanical Engineers, ASME N510 (Reference 9), ASME AG-1 (Reference 1), and RG 1.52 (Reference 11).

Fan heaters are tested in accordance with ASME AG-1, Section CA (Reference 1).

Periodic testing and inspections identify systems and components requiring corrective maintenance, and plant maintenance programs correct deficiencies.~~Refer to Section 14.2 (test abstracts #081and #203) for initial plant startup test program. Initial in-place acceptance testing of FBVS components will be performed in accordance with Reference 1.~~

9.4.2.5 Instrumentation Requirements

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Indication of the operational status of the equipment, position of dampers, instrument indications and alarms are provided in the MCR. Fans, motor-operated dampers, heaters and cooling units are operable from the MCR. Local instruments are provided to measure flow, temperature, and pressure. The fire detection and sensors information is delivered to the fire detection system. The radiation instrumentation requirements for controlling airborne radioactivity releases via the vent stack are addressed in ~~Section 11.5~~Section 11.5.3.1.7 and Table 11.5-1, Monitors R-17, R-18, and R-19.

9.4.2.6 References

1. ASME AG-1 ~~2003~~, "Code on Nuclear Air and Gas Treatment," The American Society of Mechanical Engineers, 1997~~2003~~ (including the AG-1a~~,~~ -2000~~2004~~ "Housings" Addenda).

- SB Cell 6 exhaust (refer to Section 9.4.5).
- Containment Building full flow purge exhaust (refer to Section 9.4.7).

The filtration trains to process exhaust air from the above areas are located inside the NAB. Each filter train consists of a prefilter and a HEPA filter. Under normal operating conditions, these flow paths open into a common exhaust plenum. Four exhaust fans take suction from this plenum and discharge into another exhaust plenum which directs the exhaust air to the vent stack for an elevated release.

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If high radiation is detected in any of the rooms within the NAB (refer to Table 11.5-1, Monitors R-11, R-12, and R-13), Reactor Building (RB), FB (refer to Table 11.5-1, Monitors R-17 and R-18), or SBs (refer to Table 11.5-1, Monitor R-25), the NABVS exhaust is diverted to an iodine filtration plenum. It is then directed to one of the four redundant independent iodine filtration units. Each iodine filtration unit includes fire dampers, preheater, iodine adsorber using activated carbon, HEPA filters, dampers, and a booster fan. The exhaust air from the booster fan is directed to the exhaust plenum for discharge through the vent stack. See Figure 9.4.3-4—Nuclear Auxiliary Building Exhaust Iodine Filtration Train Subsystem.

The NABVS also has two iodine filtration train units and fans to serve the laboratory exhaust air. Each laboratory iodine filtration train unit includes preheater, HEPA filters, iodine adsorber, motor-operated dampers, and booster fans. The exhaust air from the booster fans is directed to the exhaust plenum for discharge through the vent stack. See Figure 9.4.3-5—Nuclear Auxiliary Building Laboratory Iodine Exhaust Filtration Train. Non-condensables from the turbine gland steam condenser and the condenser evacuation system exhaust into the NABVS exhaust plenum. Air is pulled from the exhaust plenum by the NABVS exhaust fans and discharged at the vent stack.

9.4.3.2.2 Component Description

The major components of the NABVS are listed below, along with the applicable code and standards. Refer to Section 3.2 for the seismic and system quality group classification of these componentsTable 3.2.2-1 provides the seismic design and other design classifications for components in the NABVS.

Ductwork and Accessories

The supply and exhaust air ducts are constructed of galvanized sheet steel and are structurally designed for fan shutoff pressures. The ductwork meets the design, testing and construction requirements per ASME AG-1-2003 (Reference 1).

Heaters

Supply air trains have hot water heaters. The heater design is based on the minimum outside air design temperature and supply air temperature requirements. The coils are

Iodine Activity Detection

In the event iodine is detected in the NAB, FB, or SB, the affected exhaust flow paths are redirected through the iodine filtration train prior to discharge through the vent stack. Iodine activity is detected separately in each cell.

Fuel Handling Accident in the Fuel Building

In the event of a fuel handling accident in the FB, the FB exhaust and supply are isolated by closing the appropriate dampers (refer to Section 9.4.2). To prevent spread of airborne contamination, the iodine filtration trains of the SB ventilation system process the exhaust air to maintain the required pressure in the FB pool hall (refer to Section 9.4.5). The remainder of the FB is ventilated by the NABVS. During and after the fuel handling accident, proper NABVS supply and exhaust flow rates are maintained by adjusting the control dampers.

Fuel Handling Accident in the Containment Building

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In the event of a fuel handling accident in the Containment Building, the containment isolation valves close (refer to Section 9.4.7). Exhaust from the Containment Building is routed to the iodine filtration trains of the ~~SB~~CBVS ~~ventilation system~~. Excess air supply from the NABVS is redirected by adjusting the supply air control dampers.

Operation of Safety Injection System during LOCA

In the event of a loss of coolant accident (LOCA), leakages in the safety injection system (SIS) can lead to iodine activity levels that are above the limits of the NABVS iodine filtration trains. In such a case, the SB exhaust is routed through the SB ventilation system (refer to Section 9.4.5). Excess air supply from NABVS is redirected by adjusting the supply air control damper. The NABVS supply and exhaust to the FB are isolated (refer to Section 9.4.2).

Loss of Offsite Power (LOOP)

Upon loss of offsite power, the isolation dampers fail to the closed position, preventing any pathway for potentially contaminated air to leak out to the environment.

Station Blackout (SBO)

In the event of SBO, there will be no power to any of the electrical components of the NABVS. Isolation dampers with spring return will fail to the closed position. Other isolation dampers will fail “as-is”.

9.4.3.3 Safety Evaluation

None of the components of the NABVS perform a nuclear safety-related function. The NABVS components are not required to operate during a design basis accident (DBA).

adsorber material replacement, the unit is inspected and tested in-place in accordance with the requirements of RG 1.140 (Reference 10), ASME N510 (Reference 3) and ASME AG-1 (Reference 1). The charcoal adsorber samples are tested for efficiency in a laboratory in accordance with RG 1.140 (Reference 10) and ASTM D3803 (Reference 11). Air filtration and adsorption unit heaters are tested in accordance with ASME AG-1, Section CA (Reference 1).

Periodic testing and inspections identify systems and components requiring corrective maintenance, and plant maintenance programs correct deficiencies.

~~Refer to Section 14.2 (test abstracts #079 and #203) for initial plant startup test program. Initial in-place acceptance testing of NABVS components will be performed in accordance with Reference 1 and Reference 3.~~

9.4.3.5 Instrumentation Requirements

Indication of the operational status of the equipment, position of dampers, instrument indications and alarms are provided in the MCR. Fans, motor-operated dampers, heaters, and cooling units are operable from the MCR. Local instruments are provided to measure differential pressure across filters, flow, temperature and pressure.

The fire detection and sensors information is delivered to the fire detection system.

All instrumentation provided with the filtration units is as required by RG 1.140.

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The radiation instrumentation requirements for controlling airborne radioactivity releases via the vent stack are addressed in ~~Section 11.5~~Section 11.5.3.1.6 and Table 11.5-1.

9.4.3.6 References

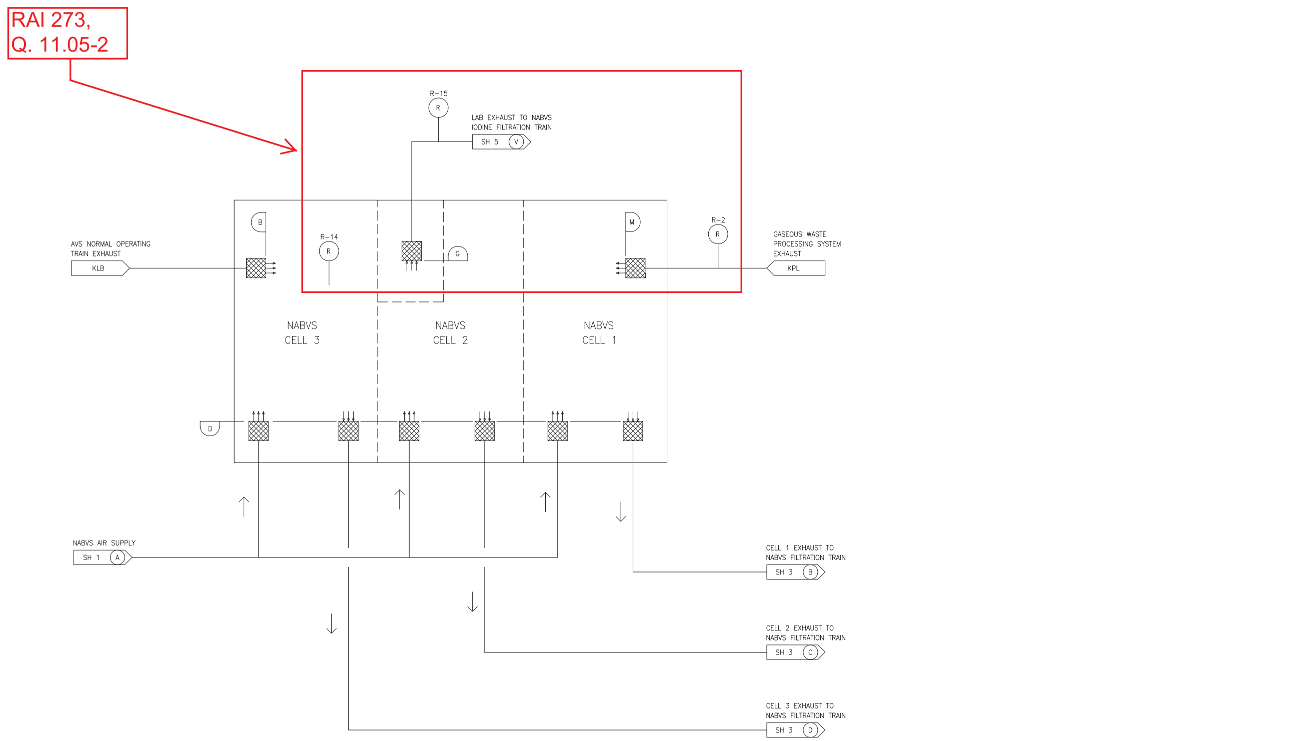
1. ASME AG-1-~~2003~~, "Code on Nuclear Air and Gas Treatment," The American Society of Mechanical Engineers, 1997~~2003~~ (including the AG-1a~~,~~ -2000, "Housings"~~2004~~ Addenda).

2. ANSI/ASHRAE Standard 52.2-1999, "Method of Testing General Ventilation Air-Cleaning Devices for Removal Efficiency by Particle Size," ANSI/American Society of Heating, Refrigerating and Air Conditioning Engineers, 1999.

3. ASME N510-1989 (R1995), "Testing of Nuclear Air-Treatment Systems," The American Society of Mechanical Engineers, 1989.

4. ANSI/AMCA-210-99, "Laboratory Methods of Testing Fans for Aerodynamic Performance Rating," American National Standards Institute/Air Movement and Control Association International, December 1999.

Figure 9.4.3-2—Nuclear Auxiliary Building Air Supply and Exhaust Subsystem



DESIGN AREA	SSC QUALITY GROUP	SSC SEISMIC CLASS
M	N/A	RS
G	N/A	NSC
D	N/A	NSC
B	N/A	NSC

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Figure 9.4.3-3—Nuclear Auxiliary Building Exhaust Filtration Trains Subsystem

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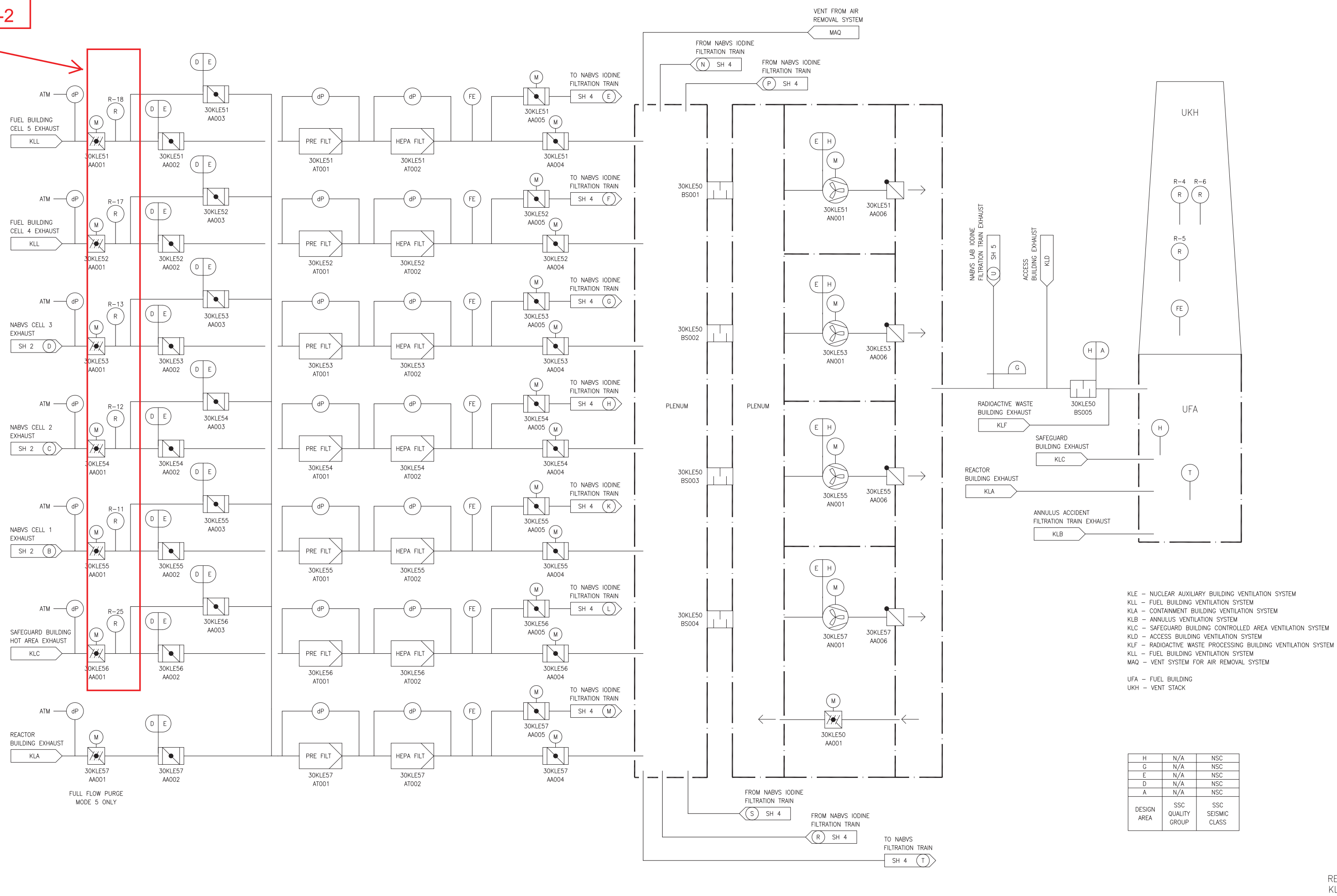
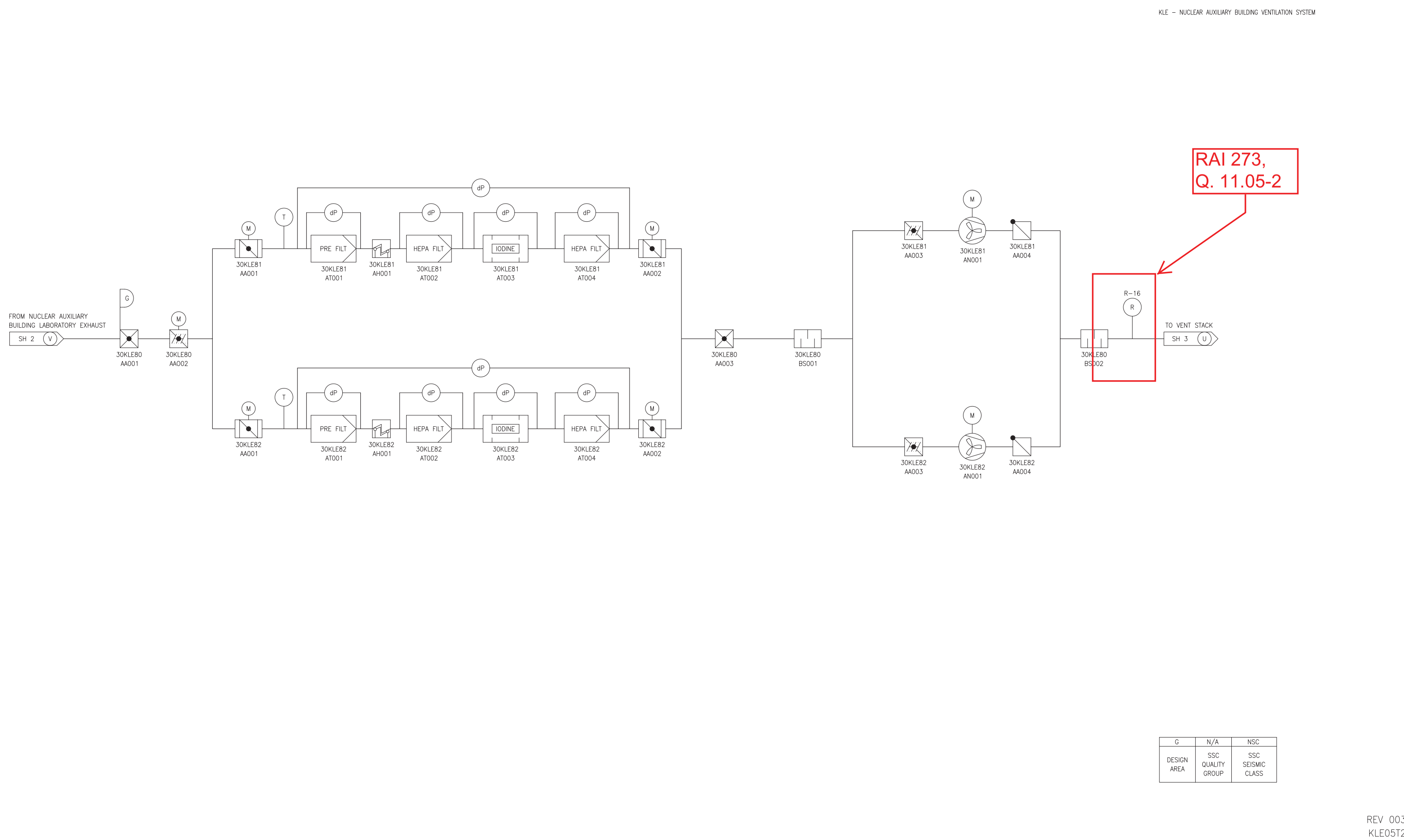


Figure 9.4.3-5—Nuclear Auxiliary Building Laboratory Iodine Exhaust Filtration Train



9.4.4 Turbine Island Ventilation Systems

Ventilation systems for the turbine island consists of the turbine building ventilation system (TBVS) and the switchgear building ventilation system, turbine island (SWBVS).

The turbine building ventilation system (TBVS) provides heating and ventilation in the Turbine Building (TB) for normal operating modes as well as during refueling outages. The TBVS supplies conditioned air to maintain TB temperatures within the operating requirements for equipment operation and provides an acceptable environment for personnel who operate and maintain the equipment within the building. The TBVS is classified as a non-safety-related system; it does not provide accident response nor does it provide radioactive effluent control functions for the U.S. EPR.

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The TBVS does not exhaust turbine gland seal or condenser evacuation. These gaseous exhausts are directed to the NABVS, where they are exhausted through the vent stack and monitored.

As noted in Table 1.8-1—Summary of U.S. EPR Plant Interfaces with Remainder of Plant, the TB and its associated ventilation system is an interface with the U.S. EPR standard design. A COL applicant that references the U.S. EPR design certification will provide site-specific design information for the TBVS.

The switchgear building ventilation system, turbine island (SWBVS) provides heating and ventilation in the remainder of the Switchgear Building (SWGB) for normal operating modes as well as during refueling outages. The SWBVS supplies conditioned air to maintain SWGB temperatures within the operating requirements for equipment operation and provides an acceptable environment for personnel who operate and maintain the equipment within the building. The SWBVS is classified as a non-safety-related system; it does not provide accident response nor does it provide radioactive effluent control functions for the U.S. EPR.

As noted in Table 1.8-1—Summary of U.S. EPR Plant Interfaces with Remainder of Plant, the SWBVS is an interface with the U.S. EPR standard design. A COL applicant that references the U.S. EPR design certification will provide site-specific design information for the SWBVS.

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(GDC 4). Refer to Section 3.5.1.1, Section 3.5.1.4, Section 3.5.2, and Section 3.6.1 for information on compliance with GDC 4 as it relates to protection from missiles and postulated piping failures.

The safety-related components and systems of the SBVS are not shared among nuclear power units (GDC 5).

The essential onsite electrical power systems meet the guidance of NUREG-CR/0660 (subsection A–item 2, and subsection C-item 1) (Reference 1) for protection of essential electrical components (such as contactors, relays, circuit breakers) from failure due to the accumulation of dust and particulate materials (GDC 17). This is accomplished by the roughing prefilters and filters of the supply air units of the SBVSE as described in Section 9.4.6.

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The release of radioactive materials to the environment is controlled by meeting the guidance of RG 1.52 (position C.3) (GDC 60). Upon receipt of a high radiation alarm in the hot mechanical areas of the SBs (refer to Table 11.5-1, Monitor R-25), the SBVS will direct the exhaust air (accident exhaust) through NABVS activated charcoal filtration beds located in the Fuel Building (FB)NAB prior to release through the plant stack. As a backup, the contaminated air also can be processed through activated charcoal filtration beds of the NABVS.

Filtration during normal operation is provided by the NABVS by meeting the guidance of RG 1.140 (positions C.2 and C.3). Refer to Section 9.4.3.

Capability for withstanding or coping with a station blackout (SBO) event is provided to comply with the requirements of 10 CFR 50.63. Acceptance is based on meeting the applicable guidance of RG 1.155, including position C.3.2.4. Refer to Section 8.4 for a description of the design features to cope with the SBO event.

The SBVS provides isolation and confinement of the hot mechanical areas of the SBs. The system also provides reduction of a possible radioactive release into the environment.

In case of fuel handling accident in the FB, or fuel handling accident in the Reactor Building (RB), the exhaust air (accident air) from these buildingsthe FB and the hot mechanical area of the SB is directed through the SBVS activated charcoal filtration beds located in the FB prior to release through the plant stack.

On receipt of containment isolation signal or high radiation signal in the RB, the volume of the FB is isolated NABVS supply and exhaust isolation dampers are closed to limit leakage out of the FB. The SBVS maintains negative pressure in the FB and air from the FB is directed to the SBVS iodine filtration trains (refer to Section 9.4.2).

dampers and one volume control damper. The exhaust air is processed by the NABVS through a filtration train prior to release through the plant stack (refer to Section 9.4.3).

- Accident Air Exhaust Mode–If airborne contamination is detected in any of the four hot mechanical areas of the SBs or there is a containment isolation signal, the SBVS will automatically direct the exhaust air (accident exhaust) via four separate exhaust air ducts and isolation dampers to one common concrete duct in the annulus. This exhaust duct connects to two accident iodine exhaust filtration trains located in the FB. The exhaust air is processed through one of two redundant and independent iodine filtration trains prior to release through the plant stack. Each iodine filtration train includes inlet and outlet dampers, moisture separator, electric heater, inlet and outlet high efficiency particulate air (HEPA) filters, carbon adsorber, exhaust fan, and backdraft damper. The fans direct the exhaust air to the plant stack.

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~~As a backup, during the accident mode, the contaminated air also can be processed through the iodine filtration units of NABVS (refer to Section 9.4.3).~~

In case of a fuel handling accident in the FB, ~~or a fuel handling accident in the RB,~~ the accident exhaust air from these buildings is directed and filtered through the SBVS iodine exhaust filtration trains located in the FB, and released through the plant stack.

In case of containment isolation signal ~~or high radiation signal in the RB~~, the SBVS maintains a negative pressure ~~in the FB~~ and filters all areas of the FB and the hot mechanical area of the SB in addition to performing the SBVS accident air exhaust filtration function.

The supply and exhaust duct network of the hot mechanical area in the SBs is equipped with isolation dampers to isolate the following areas from the other rooms:

- Rooms where safety injection and residual heat removal system components in divisions one and four are installed.
- Rooms where severe accident heat removal system components in division four are installed.
- Personnel air lock area in division two.

Recirculation cooling units are provided for the following rooms where high heat load equipment is located:

- Rooms in the SB, divisions one through four, where safety injection and residual heat removal system components are installed.
- Valve rooms in the SB, divisions one through four, where component cooling water system and emergency feedwater system components are installed.

- If the personnel air lock is open, the air supply and exhaust air flow to and from the personnel air lock area is placed in service by manually closing or opening the associated dampers.
- If maintenance is performed on the equipment or systems which pose delayed iodine release hazard, the exhaust air from these areas is diverted to the iodine filtration plenum of the NABVS prior to discharge through the plant stack (refer to Section 9.4.3).

Abnormal Operating Conditions

Loss of Recirculation Cooling or Area Ventilation

Failure of recirculation cooling or area ventilation in one SB division has no effect on safety function of SBVS since other three unaffected SB divisions are capable of performing the necessary safety function.

Two supply and exhaust dampers are provided for isolation of the hot mechanical area of each SB division. If one damper fails, the other damper can perform the safety function.

Loss of an Accident Iodine Exhaust Filtration Train

The SBVS provides two accident exhaust iodine filtration trains. Failure of one filtration train has no effect since the unaffected train can perform the necessary filtration function.

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Redundant switching dampers are arranged in parallel configuration. As such failure of one damper has no consequence since the unaffected damper can perform the necessary function.

~~If both SBVS exhaust iodine filtration trains fail, the negative pressure and filtering function of SB hot mechanical area, and fuel handling accident exhaust in the FB can be provided by the exhaust filtration trains of NABVS (See Section 9.4.3).~~ The fuel handling accident exhaust in the RB can be provided by the exhaust filtration trains of CBVS (refer to Section 9.4.7).

Loss of Offsite Power

The following equipment will remain operational during loss of offsite power (LOOP). The power supply for this equipment is supplied from the corresponding emergency diesel generators.

- Dampers in all divisions of SB.
- Iodine exhaust filtration trains located in the FB.
- Dampers to the zones that need to be isolated and confined.

Station Blackout

Station black out (SBO) does not lead to release of radioactivity inside the SB, FB and RB. The system filtering function is therefore not required during SBO. However, the following components are supplied from the SBO diesel generators alternate AC (AAC) power:

- Supply air control dampers to the controlled areas in divisions one and four, to isolate air supply when exhaust is not in operation due to SBO.
- Recirculation cooling units in the SB divisions one and four, where the EFW valves are located.

Loss of Ultimate Heat Sink

During loss of ultimate heat sink (LUHS), the air flow of the recirculation cooling units is cooled by the chilled water provided by the SCWS. Two water-cooled chillers are located in divisions two and three, and two air-cooled chillers are located in divisions one and four. In case of LUHS, the water-cooled chillers are not available. With the safety chilled water divisions 1/2 and 3/4 interconnect, the safety chilled water is then supplied by air-cooled chillers which provide the cooling function for the recirculation cooling units located in divisions one, two, three and four.

Loss of Coolant Accident

In the event of a loss of coolant accident (LOCA), ~~the air supply from the SBVSE and the exhaust to the NABVS are closed automatically. Exhaust air is then diverted by opening or closing the associated dampers to process the exhaust air through the SBVS iodine exhaust filtration trains located in the FB~~a containment isolation signal initiates isolation of the SB controlled areas from the supply to the SBVSE and the exhaust from the NABVS. Air is supplied from the SB controlled areas and exhausted through the SBVS iodine exhaust filtration trains (located in the FB) and discharged to the plant vent stack. To support the operation of plant equipment, recirculation cooling units maintain rooms in the SB controlled areas at ambient conditions.

In the event of a LOCA, the containment isolation signal ~~or high radiation signal in the RB~~ initiates isolation of the FB from NABVS supply and exhaust duct to limit leakage into the FB. The SBVS maintains a negative pressure in the FB and exhaust air from the FB is directed to the SBVS iodine filtration trains (refer to Section 9.4.2).

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Iodine Presence in the SB Rooms

In the event of a failed fuel element and residual heat removal pump seal leakage, high iodine is expected to be present in only one of the four divisions at a time, and it is necessary to purify the air in this division for personnel access. The air supply and exhaust flow for the affected division is increased to purge the possibly contaminated areas, while air supply and exhaust for the other three divisions is decreased. This is achieved by opening or closing the isolation dampers and partially opening the control dampers in order to maintain an acceptable total exhaust air flow to the NABVS iodine filtration train.

Fuel Handling Accident in the FB

In the event of a fuel handling accident in the FB, the exhaust air from the FB is processed through the SBVS iodine filtration trains located in the FB. The damper configuration is as follows:

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- Associated dampers in the ducts from the FB to the SBVS filtered exhaust are in the open position.
- Associated dampers in the ducts from the RB to the SBVS filtered exhaust are in the closed position.
- Associated dampers for the SBVS accident air exhaust to the SBVS filtered exhaust are in the closed position.
- One (or both) SBVS iodine filtration trains are in service.

Fuel Handling Accident in the RB

In the event of a fuel handling accident in the RB, the exhaust air from the RB is processed through the SBVS iodine filtration trains located in the FB. The damper configuration is as follows:

- Associated dampers in the ducts from the RB are in the open position.
- Associated dampers in the ducts from the FB are in the closed position.
- Associated dampers for the SBVS accident air exhaust from the SB are in the closed position.
- One (or both) SBVS iodine filtration trains are in service.

Residual Heat Removal System Break outside Containment

The rooms inside SB divisions one and four containing the residual heat removal (RHR) equipment and piping are equipped with isolation dampers in the supply and exhaust air ducts. These dampers are manually closed during RHR operation to prevent the spread of steam and airborne contamination due to a pipe failure.

Operation of Containment Heat Removal System in Severe Accidents

The rooms inside SB division four containing the severe accident heat removal system equipment are isolated from the other rooms by closing the associated dampers located in the supply air ducts for each room.

9.4.5.3 Safety Evaluation

The SBVS is designed to maintain ambient conditions in areas of the SB divisions (refer to Section 11.5.3.1.9 and Table 11.5-1, Monitor R-25) one through four where engineered safety equipment is located. This permits personnel access and allows safe operation of the equipment during normal plant operation, outages, and under all anticipated occupational occurrences, including postulated accident events.

The SBVS provides isolation and confinement of the hot mechanical areas of the SBs (refer to Section 11.5.3.1.9 and Table 11.5-1, Monitor R-25). Two isolation dampers and one volume control damper are provided in the supply and exhaust ducts to make sure that hot mechanical areas can be purged or isolated without any leakage. The hot mechanical areas are maintained at negative pressure with respect to the outside atmospheric air pressure. The system also provides reduction of radioactive release into the environment.

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Each recirculation cooling unit for SB divisions one through four operates independently of the recirculation cooling unit in the other divisions. In case of a recirculation cooling unit failure inside one division, the recirculation cooling units for the other three divisions are unaffected.

Upon receipt of a high radiation alarm in the hot mechanical areas of the SBs (refer to Section 11.5.3.1.9 and Table 11.5-1, Monitor R-25), the SBVS directs the exhaust through the ~~SBVS~~NABVS activated charcoal filtration beds located in the ~~FB~~NAB prior to release through the plant stack. ~~As a backup, the contaminated air also can be processed through the activated charcoal filtration beds of NABVS.~~ Sufficient redundancy ~~is provided to provide~~provides reasonable assurance of proper system operation with one active component out of service.

Confinement of the four SB hot mechanical areas and startup of the SBVS accident iodine filtration trains is initiated by the safety automation system (SAS) signal.

Isolation dampers in the supply and exhaust ducts are provided for the SB division one through four rooms where safety injection and residual heat removal system equipment is located. These dampers close during RHR operation to prevent the spread of steam and airborne contamination due to a RHR system pipe failure.

Housings and ductwork are leak-tested in accordance with the Sheet Metal and Air Conditioning Contractors' National Association (SMACNA) technical manual "HVAC Air Duct Leakage Test Manual" (Reference 14), American Society of Mechanical Engineers, ASME N510 (Reference 4), ASME AG-1 (Reference 2), and RG 1.52 (Reference 10).

Emergency filtration units are tested by manufacturer for housing leakage, filter bypass leakage and airflow performance. Periodically and subsequent to each filter or adsorber material replacement, the unit is inspected and tested in-place in accordance with the requirements of RG 1.52 (Reference 10), ASME N510 (Reference 4) and ASME AG-1 (Reference 2). The charcoal adsorber samples are tested for efficiency in accordance with RG 1.52 (Reference 10) and ASTM D3803 (Reference 15). Air filtration and adsorption unit heaters are tested in accordance with ASME N510 (Reference 4).

Periodic testing and inspections identify systems and components requiring corrective maintenance, and plant maintenance programs correct deficiencies.

In-service test program and test frequency requirements are described in Section 16, "Technical Specification" Subsection 3.7.12 and per Ventilation Filter Test Program (VFTP) described in Section 16, "Technical Specification" Subsection 5.5.10.

~~Refer to Section 14.2 (test abstracts #083 and #203) for initial plant startup test program. Initial inplace acceptance testing of SBVS components is performed in accordance with Reference 2, and Reference 4.~~

~~Refer to Section 16 (SR 3.7.12) for surveillance requirements.~~

9.4.5.5 Instrumentation Requirements

Indication of the operational status of the equipment, position of dampers, instrument indications and alarms are provided in the MCR. Fans, motor-operated dampers, heaters and cooling units are operable from the MCR. Local instruments are provided to measure differential pressure across filters, flow, temperature and pressure. The fire detection and sensors information is delivered to the fire detection system. The radiation instrumentation requirements for controlling airborne radioactivity releases via the plant stack are addressed in ~~Section 11.5~~Section 11.5.3.1.9 and Table 11.5-1, Monitor R-25.

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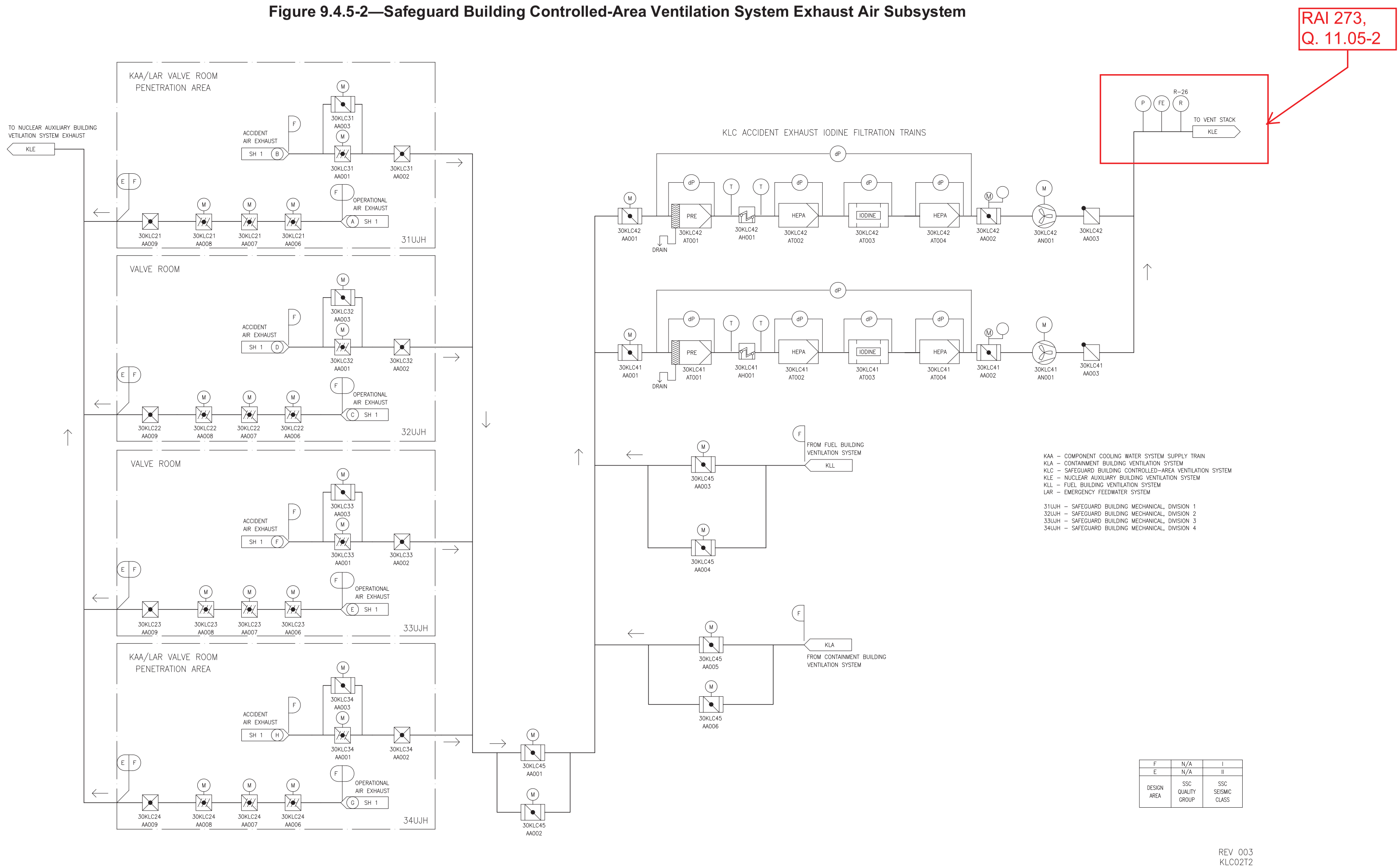
The minimum instrumentation, indication and alarms for the SBVS ESF filter system~~s~~ are provided in Table 9.4.5-1 per the requirements of ASME N509 (Reference 9).

Figure 9.4.5-2—Safeguard Building Controlled-Area Ventilation System Exhaust Air Subsystem



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Containment Purge Subsystem

The containment purge subsystem includes low-flow and full-flow purge supply and exhaust systems. See Figure 9.4.7-1—Containment Building Low Flow and Full Flow Purge Supply Subsystem and Figure 9.4.7-2—Containment Building Low Flow and Full Flow Purge Exhaust Subsystem.

The containment low-flow purge subsystem is normally not in operation during the plant normal operation. However, the low-flow purge subsystem can be used during normal operation and outage conditions. The containment full-flow purge subsystem is used during plant outages. The supply side ducts receive air from NABVS (refer to Section 9.4.3) through the Fuel Building (FB) concrete plenum. The supply air is then directed through the containment annulus penetration ducts into the containment plenum which discharges air into the service compartments of the Containment Building. The service compartments include technical rooms, instrument rooms, staircases, tank rooms, annular space at the operating floor, and annular space at the lower level. With the purge subsystem in operation, the air from the service compartments flows into equipment compartments as a result of pressure differential.

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The low-flow purge exhaust subsystem contains two redundant filtration trains located in the FB. Radiation monitors are located upstream of the filtration trains for monitoring the containment exhaust air prior to filtration (refer to Section 11.5.3.1.4 and Table 11.5-1, Monitors R-7 and R-8). The filtration trains receive air from the exhaust duct of the low-flow purge exhaust subsystems. The full-flow purge exhaust is directed to the NABVS during a fuel handling accident in the RB. The CBVS low flow purge exhaust can also be directed to the safeguard building controlled-area ventilation system (SBVS) iodine filtration trains in an emergency for redundancy (refer to Section 9.4.5). Each filtration train consists of an electric heater, prefilter, upstream HEPA filters, carbon adsorber, downstream HEPA filters, and exhaust fan. The exhaust air from the filtration trains is directed to the plant vent stack. The radiation monitor located downstream of the CBVS low flow purge iodine filtration trains monitors and records the release of radioactive contaminants to the vent stack (refer to Section 11.5.3.1.4 and Table 11.5-1, Monitor R-9). The full-flow purge exhaust subsystem directs the containment exhaust air through the NABVS exhaust filtration train (refer to Section 9.4.3).

The dampers downstream of the supply plenum regulate pressure inside the Containment Building. The equipment compartment exhaust dampers regulate differential pressure between the service and equipment compartments when the low-flow purge subsystem is operating.

The containment purge subsystems provide automatic isolation of containment atmosphere by quick closure of containment isolation valves and closure of the air supply in front of the hatch.

The containment purge subsystem is designed in accordance with ASME AG-1~~-2003~~ (Reference 1) and RG 1.52 for atmospheric cleanup.

Internal Filtration Subsystem

The internal filtration subsystem (See Figure 9.4.7-3—Containment Building Internal Filtration Subsystem) limits the release of radioactive material by reducing radioactive iodine contamination inside the equipment compartment with air circulation and filtration during normal plant operation. The internal filtration subsystem contains one filtration train which consists of an electric heater, prefilter, upstream HEPA filter, carbon adsorbers, and a downstream HEPA filter; with two redundant fans downstream of the filtration train. The air is drawn from the equipment compartments, filtered, and returned to the equipment compartments.

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Radiation monitors are located upstream of the filtration trains for monitoring the radiation in the equipment compartments prior to filtration (refer to Section 11.5.3.1.5 and Table 11.5-1, Monitor R-10).

The system is designed in accordance with Reference 1 and RG 1.140.

Containment Building Cooling Subsystem

The containment building cooling subsystem (See Figure 9.4.7-4—Containment Building Cooling Subsystem) provides cool air into a stainless steel sheet metal circular header located above the residual heat removal-safety injection room, and into the reactor pit cooling fan plenum. The containment building cooling subsystem provides cool air to the reactor coolant pumps, steam generators, chemical volume control system (CVCS), control rod drive mechanism system (CRDMS), and vent and drain system. There are two trains of two main fans and four cooling coils located in the equipment compartments. The cooling coils receive cold water from the operational chilled water system (OCWS).

Two trains of two reactor pit cooling fans located in the equipment compartments supply cool air to the reactor pit area. These fans are used to ventilate the reactor pit during normal and station blackout (SBO) conditions. The reactor pit is cooled by air from a plenum between the main fans and the reactor pit cooling fans. The supply air subsystem to the reactor pit is composed of a 16 duct layout around the main coolant piping.

The exhaust from these areas is recycled through the cooling coils located in the equipment compartments.

The system is designed in accordance with Reference 1.

assigned during the laboratory tests. The periodic in-place testing of adsorbers to determine the leak-tightness is performed per Reference 3.

Fans

The supply and exhaust fans are centrifugal or vane-axial designed with electric motor drivers. Fan performance is rated in accordance with ANSI/AMCA-210-99 (Reference 4), ANSI/AMCA-211~~-1987~~ (Reference 5), and ANSI/AMCA-300~~-1985~~ (Reference 6).

Isolation Dampers

Manual dampers are adjusted during initial plant startup testing to establish accurate air flow balance between rooms. The motor-operated isolation dampers will fail in position in case of power loss. The performance and testing requirements of the dampers will be conducted in accordance with Reference 1.

Fire Dampers

Fire dampers are installed where ductwork penetrates a fire barrier. Fire damper design meets the requirements of UL 555 (Reference 7) and the damper fire rating is commensurate with the fire rating of the barrier penetrated.

Recirculation Cooling Units

The recirculation cooling units consist of a fan section, a water cooling section, and a moisture separator. The housing is constructed of heavy gauge steel. The fan is driven by an electric motor. The cooling coils are finned coil type and are connected to the operational chilled water system. The cooling coils are designed in accordance with Reference 1. The moisture separator collects condensate which is directed to drain system.

9.4.7.2.3 System Operation

Normal Plant Operation

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The containment low-flow purge subsystem can operate during normal operation. The containment building negative pressure is maintained by controlling the supply air flow through the motorized dampers. The internal filtration subsystem equipment compartment is isolated unless airborne radioactivity contamination is detected (refer to Section 11.5.3.1.5 and Table 11.5-1, Monitor R-10) and personnel access is required in the service compartment. When the low-flow purge subsystem is in operation, a negative pressure is maintained between the equipment and service compartments.

When the reactor is in cold shutdown, ventilation in the Containment Building is provided by both low-flow and full-flow purge subsystems. The negative pressure in containment is regulated by the supply air flow of both low-flow and full-flow purge subsystems.

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The internal filtration subsystem is in operation during plant operation to detect activity level in the building, and air flow purges the equipment compartment in a recirculation mode (refer to Section 11.5.3.1.5 and Table 11.5-1, Monitor R-10). This system is not required during outages since there are no fission products being produced.

The containment building cooling subsystem operates during normal and shutdown conditions to remove heat generated in the equipment compartments. This system operates continuously to maintain ambient conditions in the equipment compartments. If the supply air temperature downstream of fans is 82°F or higher, the cooling coils provide cool air.

The service compartment cooling subsystem also operates during normal and shutdown conditions to maintain acceptable room temperatures in the service compartments.

Abnormal Operating Conditions

The containment isolation valves located on the low-flow and full-flow purge supply and exhaust ducts automatically close when a containment isolation signal is initiated. In the event of loss of the internal filtration subsystem, the exhaust air can be filtered through the containment low-flow purge exhaust subsystem prior to release to the plant stack.

In the event of loss of the chilled water system, the component cooling water system (CCWS) provides a water supply to the cooling coils.

In the event of failure of the containment building cooling subsystem fans, the fresh air to the annular space and the operating floor and equipment compartment can be supplied by the full-flow purge subsystem in conjunction with a reconfiguration of the dampers.

Loss of Ultimate Heat Sink

In case of loss of ultimate heat sink (LUHS) or the loss of CCWS, the cooling fans in the Containment Building are kept in operation to avoid localized areas of high temperature.

Loss of Offsite Power

Upon loss of offsite power (LOOP), the containment penetration isolation valves fail to the closed position. The dampers on the internal filtration subsystem and containment building cooling subsystem fail to the “as-is” position. The power supply to main fans and reactor pit cooling fans is supplied from corresponding emergency diesel generators. Air cooling unit fans stop in the service compartment cooling subsystem.

Fuel Handling Accident in the Containment Building

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In the event of a fuel handling accident in the Containment Building, the containment isolation valves on the containment purge subsystem can be manually closed by pushing the emergency push button located in the fuel handling area inside the Containment Building. The dampers are closed when the hatch is opened. The low-flow purge exhaust subsystem is used to avoid the spread of contamination by keeping a negative pressure in the Containment Building. To achieve this safety function, the low-flow purge subsystem exhaust is switched over to the iodine filtration trains of the safeguard building controlled-area ventilation system (refer to Section 9.4.5, Section 11.5.3.1.5, Section 11.5.4.8, and Table 11.5-1, Monitor R-10).

High Pressure Level or Safety Injection Signal

In case of high-pressure level or a safety injection signal, the containment penetration valves on the containment purge subsystem are closed and air flow in the Containment Building is stopped.

Station Blackout

In the event of a SBO, the reactor pit area is air cooled to prevent degradation of the concrete structure. The reactor pit cooling fans take air from the supply air shaft. The air is supplied to the bottom of the pit and transferred through openings in the pit wall around the main coolant piping to maintain a temperature less than 150°F. The power supply to the reactor pit cooling fans is provided by the alternate AC (AAC) diesel generators.

Small-Break Loss-of-Coolant Accident and Loss-of-Coolant Accident

In the event of a small-break loss-of-coolant accident (SBLOCA) or loss-of-coolant accident (LOCA), containment isolation valves automatically close after receipt of the containment isolation signal. These valves are designed to perform their isolation function under LOCA conditions and will close within five seconds after receipt of a containment isolation signal.

9.4.7.3 Safety Evaluation

The CBVS maintains proper temperatures in the Containment Building during normal operations and shutdown conditions. Sufficient redundancy is included for proper operation of the system when one active component is out of service. The CBVS is ~~not~~ an engineered safety feature and ~~has no~~the safety-related function provides containment isolation and low-flow purge exhaust from the containment isolation

9.4.7.5 Instrumentation Requirements

Indication of the operational status of the equipment, position of dampers, instrument indications and alarms are provided in the main control room (MCR). Fans, motor-operated dampers, heaters and cooling units are operable from the MCR. Local instruments are provided to measure differential pressure across filters, flow, temperature and pressure. The fire detection and sensors information is delivered to the fire detection system.

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The minimum instrumentation, indication and alarms for CBVS ESF filter systems are provided in Table 9.4.7-1 per the requirements of ASME N509 (Reference 15).

The radiation instrumentation requirements for controlling airborne radioactivity releases via the plant stack are addressed in ~~Section 11.5~~Sections 11.5.3.1.4, 11.5.3.1.5, Section 11.5.4.8, and Table 11.5-1, Monitors R-7 and R-8 (Low Purge Subsystem) and R-10 (Internal Filtration Subsystem).

9.4.7.6 References

1. ASME AG-1 ~~2003~~, "Code on Nuclear Air and Gas Treatment," The American Society of Mechanical Engineers, ~~2003~~1997 (including the AG-1a~~, 2004~~2000, "Housings" Addenda).

2. ANSI/ASHRAE Standard 52.2-1999, "Method of Testing General Ventilation Air-Cleaning Devices for Removal Efficiency by Particle Size," ANSI/American Society of Heating, Refrigerating and Air Conditioning Engineers, 1999.

3. ASME N510-1989 (R1995), "Testing of Nuclear Air-Treatment Systems," The American Society of Mechanical Engineers, 1989.

4. ANSI/AMCA-210-99, "Laboratory Methods of Testing Fans for Aerodynamic Performance Rating," American National Standards Institute/Air Movement and Control Association International, December 1999.

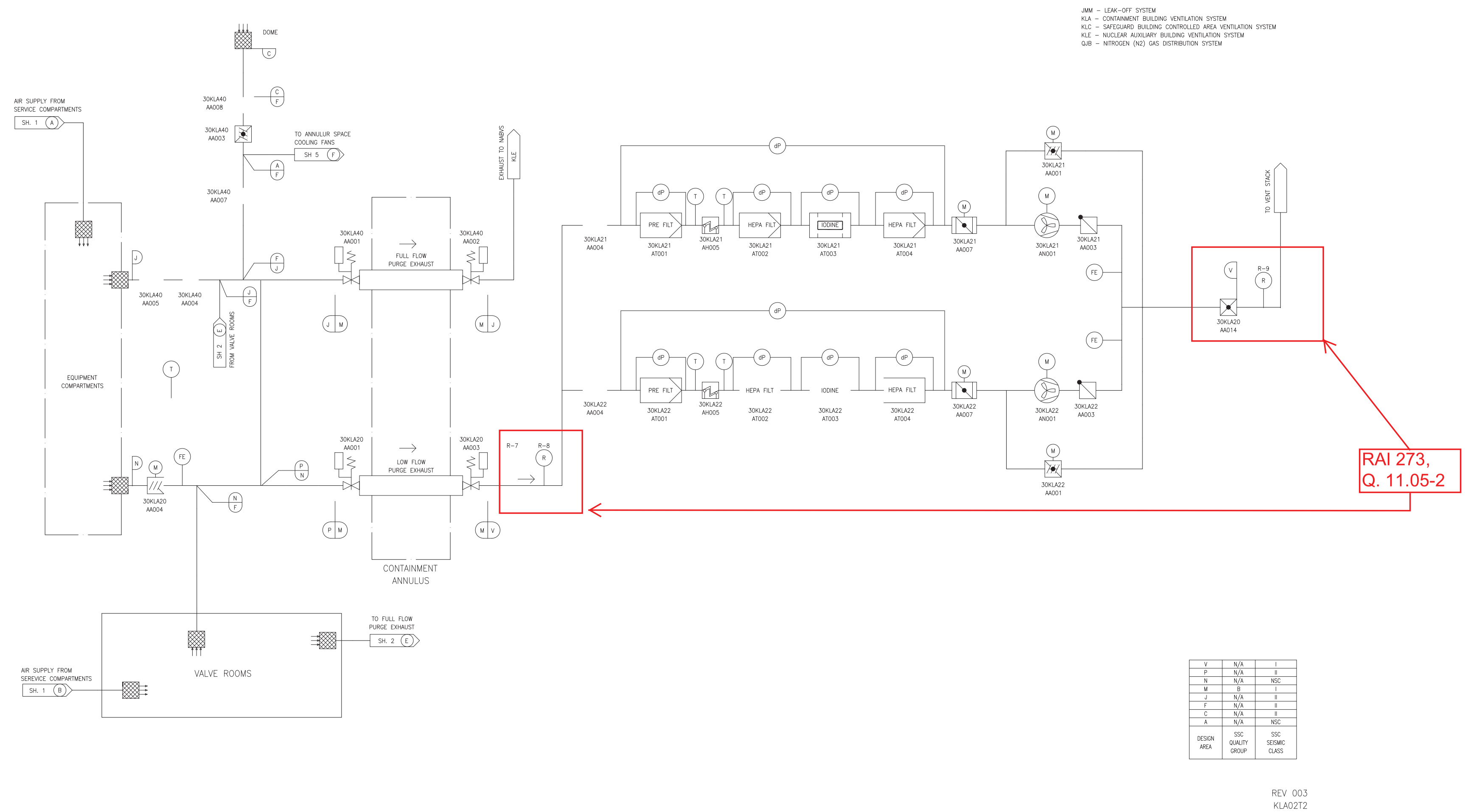
5. ANSI/AMCA-211-1987, "Certified Ratings Program–Air Performance," American National Standards Institute/Air Movement and Control Association International, 1987.

6. ANSI/AMCA-300-1985, "Reverberant Room Method of Testing Fans for Rating Purposes," American National Standards Institute/Air Movement and Control Association International, 1985.

7. UL 555, "Standard for Fire Dampers," Underwriter's Laboratories, Sixth Edition, June 1999.

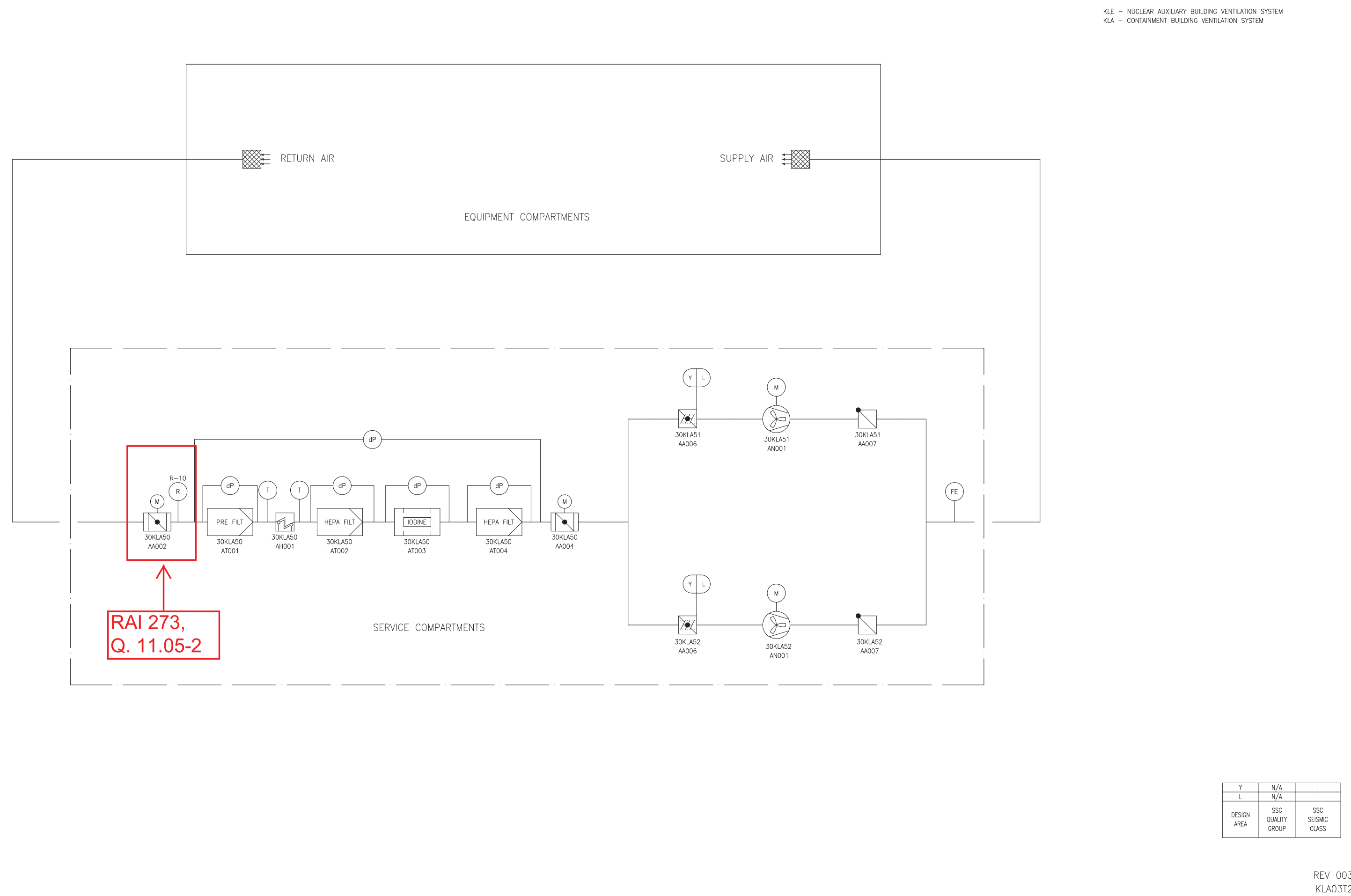
8. NUREG-0800, BTP 6-4, Revision 3, "Containment Purging During Normal Plant Operations," U.S. Nuclear Regulatory Commission, March 2007.

Figure 9.4.7-2—Containment Building Low Flow and Full Flow Purge Exhaust Subsystem



DESIGN AREA	SSC QUALITY GROUP	SSC SEISMIC CLASS
V	N/A	I
P	N/A	II
N	N/A	NSC
M	B	I
J	N/A	II
F	N/A	II
C	N/A	II
A	N/A	NSC

Figure 9.4.7-3—Containment Building Internal Filtration Subsystem



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9.4.8 Radioactive Waste Building Ventilation System

The radioactive waste building ventilation system (RWBVS) provides fresh conditioned air to the Radioactive Waste Building (RWB) to maintain acceptable ambient conditions within the building. There are two exhaust air systems - system exhaust air, which draws air from locations where radioactivity is likely, and room exhaust air (Cells 1 and 2), which draws air from locations not normally expected to contain radioactivity. The RWBVS provides filtration of exhaust from ~~the rooms of~~ the RWB to limit the release of airborne contaminants exhausted from the plant stack. Additionally, the RWBVS maintains sub-atmospheric pressure in the RWB, to prevent the release of airborne contaminants into the outside atmosphere. The RWBVS functions during normal plant operation.

9.4.8.1 Design Bases

The RWBVS is non-safety-related and is located in a building that is not Seismic Category I. The U.S. EPR meets:

- GDC 2, as it relates to meeting the guidance of RG 1.29 for radioactive waste management systems to be designed in accordance with RG 1.143.
- GDC 5, as it relates to the RWBVS because there are no safety-related components that are shared with any other nuclear power units.
- GDC 60, as it relates to the ability of the system to limit the release of gaseous radioactive effluents to the environment. The RWBVS exhaust filtration units are designed, tested, and maintained in accordance with RG 1.140. The air flow rate of a single cleanup filtration unit will not exceed 30,000 cfm.

The RWBVS performs no safety-related function and is Non-Seismic. ~~Refer to Section 3.2 for the seismic and system quality group classification of the RWBVS.~~ Failure of the system does not affect the reactor coolant system (RCS) pressure boundary or the safe shutdown of the plant; nor is the system required to mitigate the consequences of a 10 CFR Part 100 release.

The RWBVS performs the following important non-safety-related system functions:

- Maintains the RWB at sub-atmospheric pressure. Maintaining the building sub-atmospheric is accomplished by flow balancing of the intake and exhaust air flow with air dampers.
- Maintains adequate building temperatures for personnel in the working areas and removes waste heat from the equipment located in the building. The RWBVS maintains the following temperature and humidity values in the RWB permanent working areas based on normal outdoor temperatures specified in Table 2.1-1:
  - Temperature from 68° to 91°F.

During normal operation, the RWBVS provides fresh air to the RWB stairwells.

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Radiation Monitors R-23 (decontamination room) and R-24 shown in Figure 9.4.8-1 provide airborne and sampling points (refer to Section 11.5.3.1.8 and Table 11.5-1). The radiation monitors (R-23 and R-24) provide local and control indications, but do not initiate any automatic control functions.

The RWB has two exhaust air systems–system exhaust air and room exhaust air (see Figure 9.4.8-2).

System exhaust air draws air from RWB locations where radioactivity is likely. The exhaust air and gases from activity-bearing systems, vented air from tanks and releases from working areas and machinery are collected by the system exhaust air. The exhaust air is monitored by the sampling activity monitoring system (SAMS) prior to entering the system exhaust air filtration system. System exhaust air is continuously filtered by two filter systems consisting of prefilters, HEPA filters, and iodine adsorption charcoal filters. The treated air is then exhausted to the plant stack by two exhaust fans located in the RWB at elevation +36 ft. Air temperature and relative humidity are maintained within design requirements by water droplet separators and electrical heaters installed upstream of the filter trains. The system exhaust air has no automatic isolation functions. In the event of a high radiation alarm from the SAMS (refer to Section 11.5.3.1.8 and Table 11.5-1, Monitor R-21), operators can manually shutdown the RWBVS from the main control room (MCR).

Room exhaust air serves rooms in the RWB that are not normally expected to contain radioactivity. Room exhaust air is monitored by the SAMS prior to entering the filter section. The room exhaust air is continuously filtered by five parallel filter trains. Each filter train consists of a prefilter and a HEPA filter. Room exhaust air from these filter trains can be directed to two room exhaust fans or to a filter system consisting of a carbon adsorber and a HEPA filter. Normal operation of the room exhaust air bypasses the carbon adsorber and HEPA filter system. If radioactivity is detected by the SAMS (refer to Section 11.5.3.1.8 and Table 11.5-1, Monitors R-20 and R-22), in any of the rooms served by the room exhaust air system, the contaminated air is ~~manually~~automatically rerouted to pass through the iodine filtration system.

The iodine filter unit, installed at the RWB at elevation +36 ft, consists of one air train equipped with two manually operated isolation dampers, one electric heater, one carbon adsorber, one HEPA filter, and two booster fans connected in parallel.

Refer to Section 12.3.6.5.6 for ventilation system design features which demonstrate compliance with the requirements of 10 CFR 20.1406.

Fire Dampers

Fire dampers are installed where ductwork penetrates a fire barrier. Fire damper design meets the requirements of UL 555 (Reference 7) and the damper fire rating is commensurate with the fire rating of the barrier penetrated.

9.4.8.2.3 System Operation

Normal Operation

The RWBVS exhaust air fan and room exhaust air fan are started manually. With the exhaust fans running, a building supply air fan is manually started. The supply air fan draws outside air through the preheater and filters the air through medium efficiency particulate filters, either cooling the air with a chilled water cooling coil or heating the air with a hot water heating coil, then humidifying the air through a humidifier and distributing the conditioned air throughout the RWB.

The supply air trains are equipped with temperature and humidity sensors that control the cooling water flow for the cooling coils, the hot water flow for the preheater and the system heater, and steam flow for the humidifier. The preheater is equipped with a freeze protection temperature sensor, which shuts down supply air fans and closes air inlet dampers if the supply air temperature decreases below a predetermined set point. The steam humidifier is controlled by a moisture sensor in the supply air duct.

The RWBVS exhaust fans are started manually. During normal operation, a system exhaust air fan and a room air exhaust fan run continuously. The standby fans are actuated upon a failure of the running fan, when maintenance is being performed on their respective running fans, or if iodine booster fans are required.

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The system exhaust air is drawn through a filter train that consists of a prefilter, a HEPA filter, a carbon adsorber, and exhaust air fan. The air is then exhausted to the plant stack. The discharge of the system exhaust fans is monitored for radioactivity. In the event of a high radioactivity level alarm (refer to Table 11.5-1, Monitor R-21), the system can be manually shut down and isolated. To maintain a constant exhaust air flow, the system exhaust air fans work in conjunction with the system exhaust air control damper to adjust for increasing pressure resistance of the filters.

The room exhaust air, which takes exhaust air from the areas that do not normally contain radioactivity, is monitored for radioactivity concentrations in the air upstream of the room exhaust air filter units. The exhaust air is drawn through five parallel filter trains (four are required for normal operation) by one of the two room exhaust fans. Each of the filter trains consists of a medium efficiency filter and a HEPA filter. The air is then exhausted to the plant stack. The room exhaust air fans work in conjunction with the room exhaust air control damper to adjust for the increasing pressure resistance of the filters to maintain constant air flow. In the event that the

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monitored radioactivity reaches the high radioactivity alarm setpoint (refer to Table 11.5-1, Monitors R-20 and R-21), the exhaust from the rooms exhibiting radioactivity is manually directed to the iodine filtration unit. The iodine filtration unit consists of a heater, a carbon adsorber, a HEPA post-filter, and one of two booster fans that discharge to the room exhaust air fans and then to the plant stack.

Abnormal Operation

The RWBVS is not required to operate during a loss of offsite power (LOOP) or station blackout (SBO) and the RWBVS is not required to operate during or after a design basis accident; therefore the system is provided with no emergency or backup power. A failure in the SHS, AHS, OCWS, or the SAMS has no major impact on the RWBVS. A failure in the RWBVS has no impact on the above support systems.

9.4.8.3 Safety Evaluation

The RWBVS is not required for the safe shutdown of the plant or for mitigating the consequences of a design basis accident or a 10 CFR Part 100 event. Therefore, the RWBVS has no safety-related function.

9.4.8.4 Inspection and Testing Requirements

The RWBVS major components, such as dampers, motors, fans, filters, coils, heaters, and ducts are located to provide access for initial and periodic testing to verify their integrity.

Initial in-place acceptance testing of the RWBVS is performed as described in Section 14.2 (test abstracts #080 and #203), Initial Plant Test Program, to verify the system is built in accordance with applicable programs and specifications.

The RWBVS is designed with adequate instrumentation for differential pressure, temperature, and flow indicating devices to enable testing and verification of equipment function, heat transfer capability and air flow monitoring.

During normal plant operation, periodic testing of RWBVS is performed to demonstrate system and component operability and integrity.

During normal operation, equipment rotation is utilized to reduce and equalize wear on redundant equipment during normal operation.

Isolation dampers are periodically inspected and damper seats replaced as required.

Fans are tested by the manufacturer in accordance with Air Movement and Control Association (AMCA) standards (References 4, 5, and 6). Air filters are tested in accordance with the American Society of Heating, Refrigerating and Air-Conditioning Engineers (ASHRAE) standards (Reference 2). Cooling coils are hydrostatically tested and their performance is rated in accordance with the Air Conditioning and Refrigeration Institute (ARI) standards (Reference 8).

Housings and ductwork are leak-tested in accordance with the Sheet Metal and Air Conditioning Contractors' National Association (SMACNA) technical manual "HVAC Air Duct Leakage Test Manual" (Reference 9), American Society of Mechanical Engineers, ASME N510 (Reference 3), ASME AG-1 (Reference 1), and RG 1.140 (Reference 8).

Heaters are tested in accordance with ASME AG-1, Section CA (Reference 1). Carbon filtration units are tested for housing leakage, filter bypass leakage and airflow performance. Periodically and subsequent to each filter or adsorber material replacement, the unit is inspected and tested in-place in accordance with the requirements of RG 1.140 (Reference 8), ASME N510 (Reference 3) and ASME AG-1 (Reference 1). The charcoal adsorber samples are tested for efficiency in accordance with RG 1.140 (Reference 8) and ASTM D3803 (Reference 10). Air filtration and adsorption unit heaters are tested in accordance with ASME AG-1, Section CA (Reference 1).

Periodic testing and inspections identify systems and components requiring corrective maintenance, and plant maintenance programs correct deficiencies.

Refer to Section 14.2 (test abstracts #080 and #203) for initial plant startup test program. Initial inplace acceptance testing of RWBVS components is performed in accordance with Reference 1, and Reference 3.

9.4.8.5 Instrumentation Requirements

Indication of the operational status of the equipment, position of dampers, instrument indications and alarms are provided in the MCR. Fans, motor-operated dampers, heaters, and cooling units are operable from the MCR. Local instruments are provided to measure differential pressure across filters, flow, temperature and pressure.

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The fire detection and sensors information is delivered to the fire detection system.

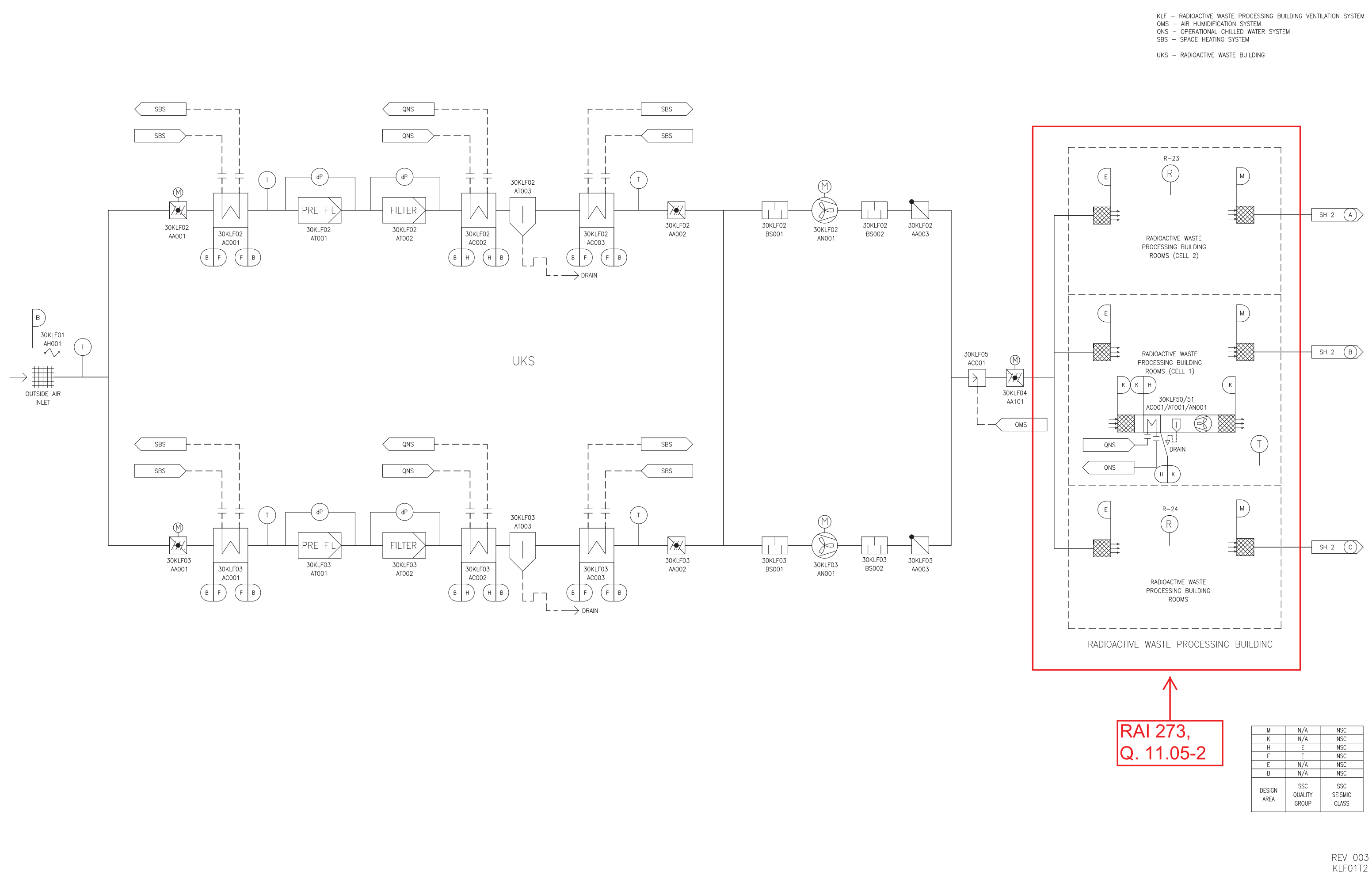
The radiation instrumentation requirements for controlling airborne radioactivity releases via the plant stack are addressed in Section 11.5Section 11.5.3.1.8 and Table 11.5-1, monitor/sample points R-20, R-21, R-22, and R-23 and R-24.

All instrumentation provided with the filtration units is as required by RG 1.140.

9.4.8.6 References

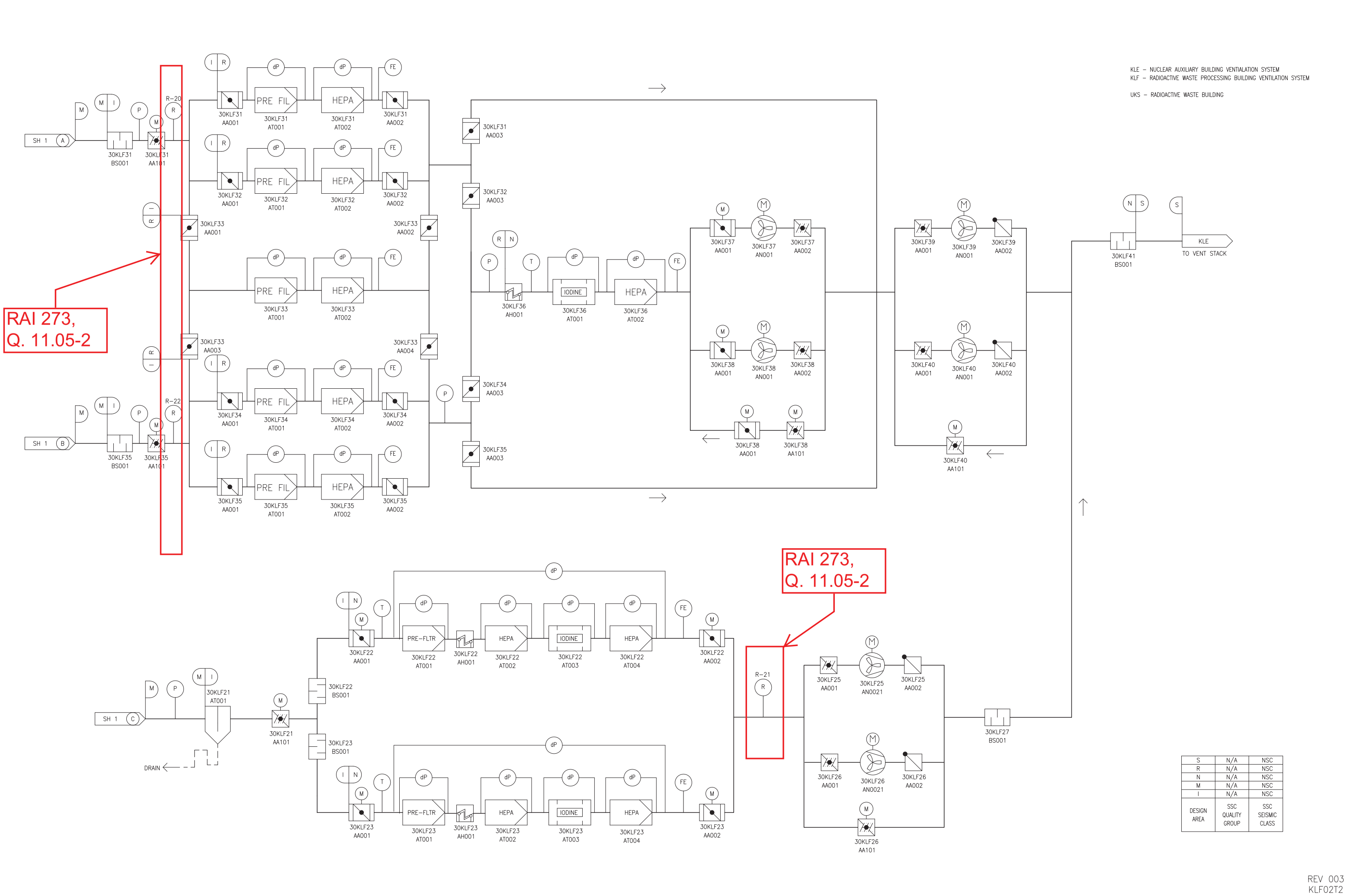
1. ASME AG-1-2003, “Code on Nuclear Air and Gas Treatment,” The American Society of Mechanical Engineers, 19972003 (including the AG-1a, -2000, “Housings,”2004 Addenda).

Figure 9.4.8-1—Radioactive Waste Building Ventilation System Air Supply



DESIGN AREA	SSC QUALITY GROUP	SSC SEISMIC CLASS
M	N/A	NSC
K	N/A	NSC
H	E	NSC
F	E	NSC
E	N/A	NSC
B	N/A	NSC

Figure 9.4.8-2—Radioactive Waste Building Ventilation System Exhaust Air Station



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to measure differential pressure across filters, flow, temperature and pressure. The fire detection and sensors information is delivered to the fire detection system.

The radiation instrumentation requirements for controlling airborne radioactivity releases via the plant stack are addressed in ~~Section 11.5~~Section 11.5.3.1.1 and Table 11.5-1, measurement point R-31.

9.4.14.7 References

1. UL 555, "Standard for Fire Dampers," Underwriter's Laboratories, Sixth Edition, June 1999.

2. ASME N509-1989~~2002~~, "Nuclear Power Plant Air-Cleaning Units and Components," The American Society of Mechanical Engineers, 1989~~2002~~.

3. ASME N510-1989 (R1995), "Testing of Nuclear Air-Treatment Systems," The American Society of Mechanical Engineers, 1989.

4. ASME AG-1~~-2003~~, "Code on Nuclear Air and Gas Treatment," The American Society of Mechanical Engineers, 2003 (including the AG-1a~~,~~-2000, "Housings,"~~2004~~ Addenda).

5. ANSI/AMCA-210-1999, "Laboratory Methods of Testing Fans for Aerodynamic Performance Rating," American National Standards Institute/Air Movement and Control Association International, December 1999.

6. ANSI/AMCA-211-1987, "Certified Ratings Program-Air Performance," American National Standards Institute/Air Movement and Control Association International, December 1987.

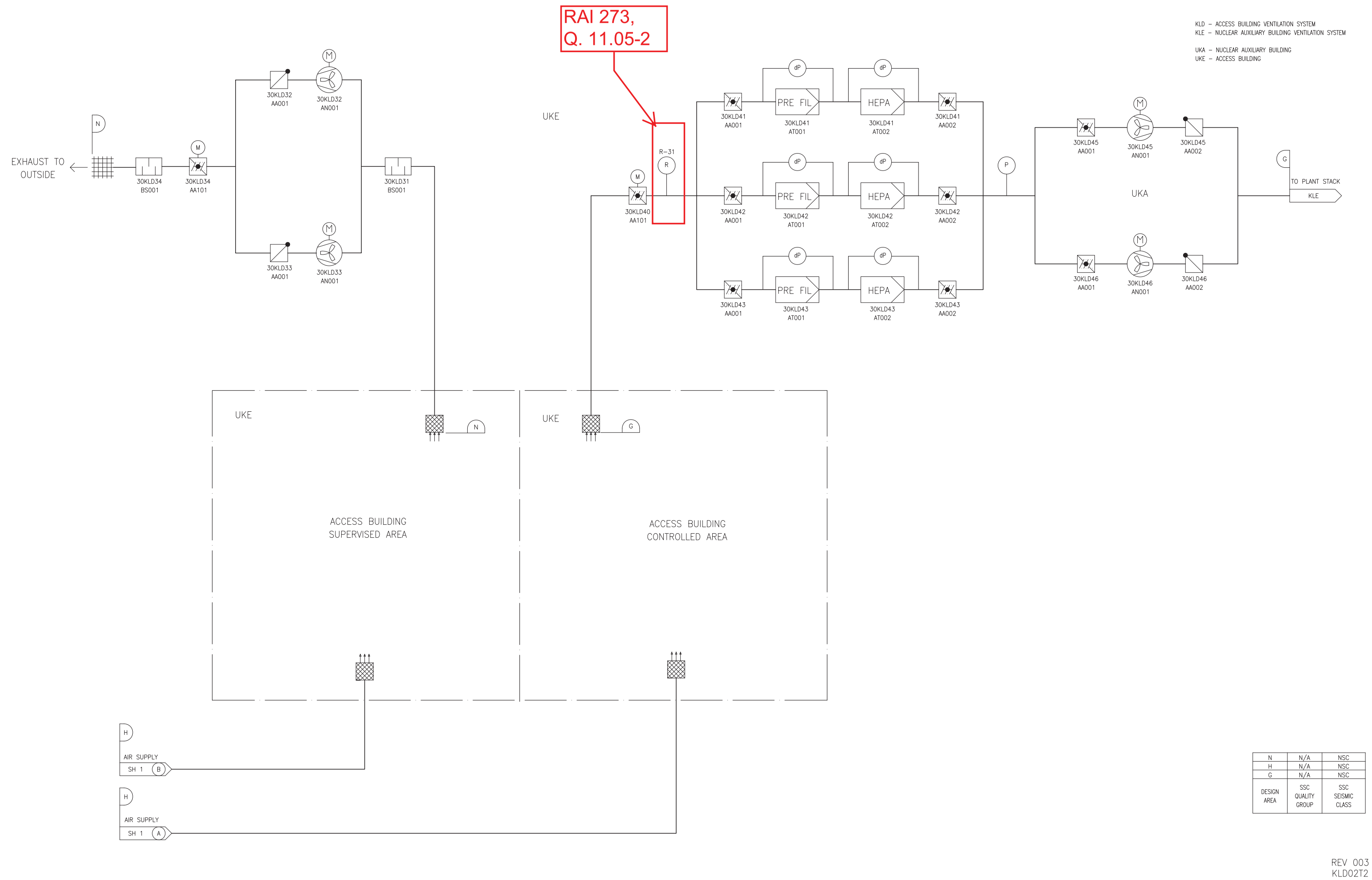
7. ANSI/AMCA-300-1985, "Reverberant Room Method of Testing Fans for Rating Purposes," American National Standards Institute/Air Movement and Control Association International, December 1987.

8. ANSI/ASHRAE Standard 52.2-1999, " Method of Testing General Ventilation Air-Cleaning Devices for Removal Efficiency by Particle Size," ANSI/ American Society of Heating, Refrigerating and Air Conditioning Engineers, 1999.

9. ANSI/ ARI Standard 410-2001, "Forced-Circulation Air-Cooling and Air-Heating Coils," Air Conditioning and Refrigeration Institute, 2001.

10. "HVAC Air Duct Leakage Test Manual," Sheet Metal and Air Conditioning Contractors' National Association, 1985.

Figure 9.4.14-2—Access Building Ventilation System – Supply and Exhaust Air Subsystem



DESIGN AREA	SSC QUALITY GROUP	SSC SEISMIC CLASS
N	N/A	NSC
H	N/A	NSC
G	N/A	NSC

Next File

Beyond Design Bases Events

The main steam system provides a non-safety-related redundant means to depressurize SG (i.e., achieve fast cooldown) to depressurize the RCS during beyond design basis events to allow for RC makeup to mitigate ICC and avoid a severe accident.

Main Steam Isolation Valves

Each main steam line includes an MSIV, located in the Valve Room just outside the containment. The MSIVs provide a safety-related function of isolating the main steam lines in the event of excessive steam flow to prevent over cooling the reactor coolant.

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In response to a main steam isolation signal, the MSIVs quickly and automatically close. Each MSIV is capable of closure in five seconds or less against a flow of approximately $5x10^6$ lbm/hr and a differential pressure of 1320 psid in either direction. Each MSIV is designed with a capability to periodically test the operability of the MSIVs and associated apparatus and determine if valve leakage is within acceptable limits. Each MSIV is seat leakage tested in the forward and reverse flow directions by the valve supplier. Periodic leak testing of each MSIV is tested by pressurizing the valve cavity between the disks.

The MSIVs are gate valves with hydraulic-pneumatic actuators and are Reference 1, Class 2, pressure boundary.

The hydraulic-pneumatic actuator is a piston actuator with its upper chamber charged with high pressure nitrogen and its lower chamber connected to a hydraulic oil system. The nitrogen stored in the upper chamber serves as a spring to close the valve without failure. The hydraulic oil supplied to the lower chamber opens the valve.

The actuator upper chamber is closed and continuously maintained at high pressure. In the event of leakage, the upper chamber is equipped with pressure transmitters to alert the operator; in which case the upper chamber is manually connected to a nitrogen gas cylinder to restore the nominal pressure.

Each MSIV actuator has its own hydraulic oil system that pumps hydraulic oil from a tank into the actuator lower chamber. Fast closure is performed by dumping the hydraulic oil back to the oil tank via two redundant lines. Figure 10.3-2 illustrates this subsystem. Only one dump line is shown for clarity. On each dump line there is a dump valve pilot-operated by two solenoid valves in series and operating on the de-energize-to-trip principle. It is necessary to de-energize the two pilots in series to open the dump valve and therefore close the MSIV. This arrangement prevents a failure of any one pilot valve from causing either spurious MSIV closure (two pilots in series) or failure to close (two redundant control lines).

Safety-related active components in the MSSS are designed to be tested during plant operation. Provisions are made to allow for inservice inspection of components at times consistent with those specified in the ASME BPV Code, Section XI (Reference 6). Section 3.9.6 describes inservice testing and Section 6.6 describes the inservice inspection program. Periodic testing to demonstrate operability of the MSSS and components is performed as specified in the Technical Specifications in Chapter 16.

10.3.5 Secondary Side Water Chemistry Program

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Control of the secondary side water chemistry and proper feedwater conditioning are required to maintain the operational capability of the U.S. EPR steam generators. Good water quality minimizes both fouling of the steam generators that can result in heat transfer losses and also the potential for environments that can cause degradation of steam generator materials. The secondary water chemistry program is based on the EPRI PWR Secondary Water Chemistry Guidelines (Reference 9) and NEI 97-06 (Reference 7).

A COL applicant that references the U.S. EPR design certification will identify the authority responsible for implementation and management of the secondary side water chemistry program.

10.3.5.1 Chemistry Control Basis

The objectives of the U.S. EPR secondary water chemistry program are:

- To protect the steam generators, turbine and supporting feedwater systems from general and localized corrosion caused by the ingress of oxygen and other chemical contaminants.
- To minimize the metal release rate from the steam-water cycle materials in order to reduce the transport of corrosion products into the steam generators.

Water chemistry recommendations for secondary systems invoke plant and operational philosophies which address the control of corrosion products and dissolved impurities by minimizing potential sources and by implementing effective monitoring. Secondary system components and piping are all-ferrous materials with the exception of the steam generator tubing. Elimination of copper alloys from the steam-water cycle allows the secondary system pH to be set at a higher value, which significantly reduces both erosion-corrosion and the corrosion of the carbon steel and low-alloy steel materials and directly reduces the transport and deposition of their corrosion products into the steam generators. Steam generator tubing is fabricated from thermally treated Alloy 690, which has been shown by both testing and operating experience to be more resistant to intergranular corrosion than Alloy 600. Alloy 690 remains, however, susceptible to stress corrosion cracking in some environments, including those with lead and low-valence sulfur species at both low and high pH. Therefore, emphasis is placed on excluding lead and lead compounds from the secondary chemistry environment.

Condensate polishing is used in the recirculation cleanup system during plant startup and shutdown evolutions to remove both dissolved and particulate contaminants prior to admitting feedwater to the steam generators. During power operation, continuous blowdown and cleanup of steam generator water limits the concentration of ionic species in the steam generator to acceptable levels.

10.3.5.2 Chemistry Control Program

U.S. EPR steam generator water and feedwater quality requirements are based on current water chemistry technology. ~~and t~~The EPRI PWR Secondary Water Chemistry Guidelines (Reference 9) and NEI 97-06 (Reference 7), and include both control and diagnostic parameters and associated action limits. Additional control and diagnostic parameters have been included as appropriate based on AREVA NP experience and information available in the literature.

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The Secondary Water Chemistry Control Program is implemented by plant operating procedures, which control the recording and management of data and require appropriate corrective actions in response to abnormal chemistry conditions. Records of chemistry program data are maintained as part of the plant records management system.

10.3.5.2.1 Control and Diagnostic Parameters

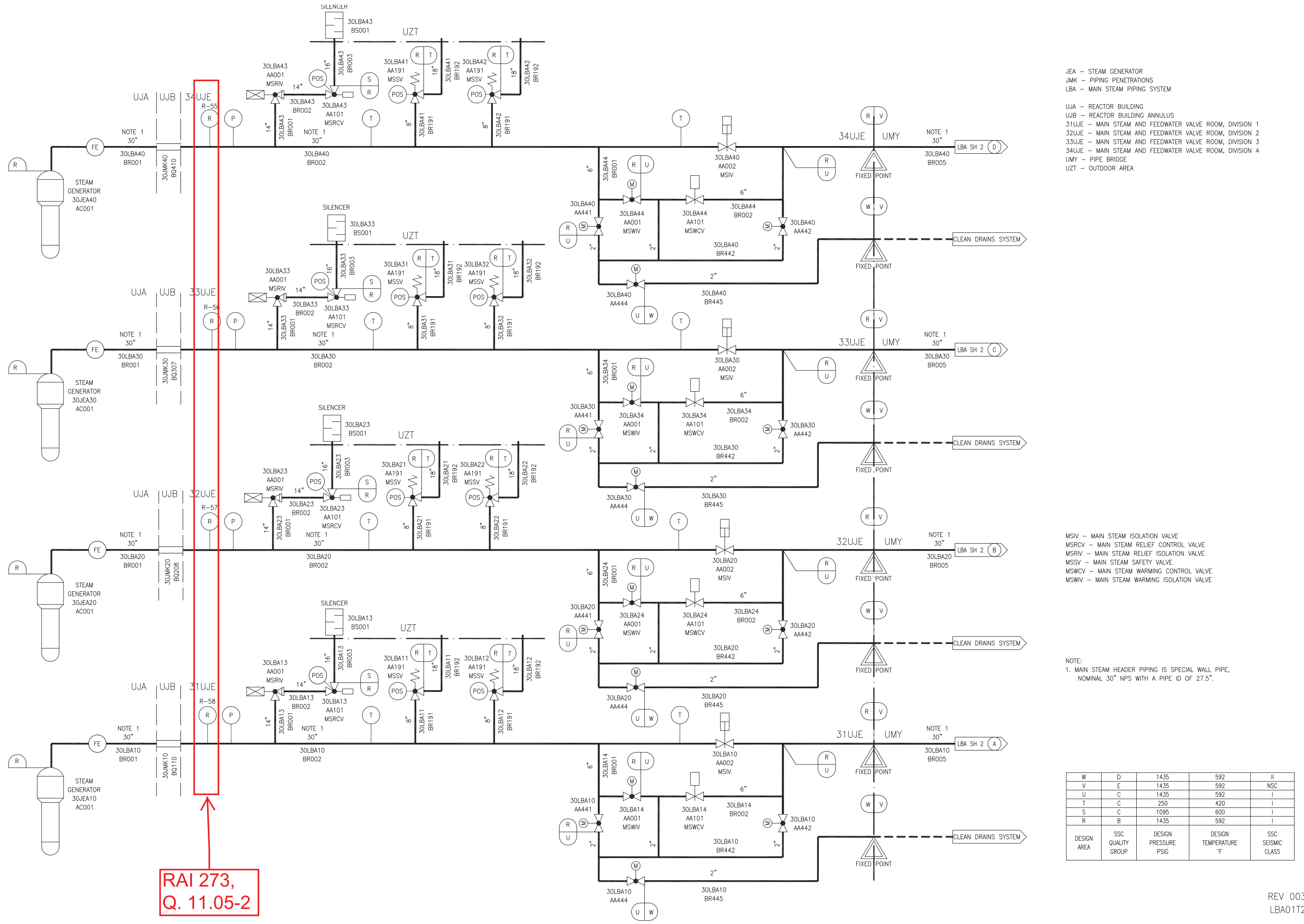
EPRI Reference 9 water chemistry guidelines place chemical parameters into two categories:

- Control Parameters are those chemistry parameters that have a demonstrated relationship to steam generator or turbine degradation and must be maintained within specified values. Control parameters have action levels, hold values and monitoring requirements associated with them.

- Diagnostic Parameters are those chemistry parameters that may affect the corrosion performance of system materials but do not impose restrictions on plant operation. They are used to monitor program effectiveness, identify programmatic problems, assist in problem diagnosis and provide early detection of a parameter that may be trending toward an action level value.

Each Control Parameter has an associated action or actions to be taken within a specified time frame when its value is outside the defined operating range. These Action Level responses are intended to prevent steam generator or secondary system materials degradation caused by abnormal chemistry conditions.

Figure 10.3-1—Main Steam Supply System
Sheet 1 of 2



DESIGN AREA	SSC QUALITY GROUP	DESIGN PRESSURE PSIG	DESIGN TEMPERATURE °F	SSC SEISMIC CLASS
W	D	1435	592	II
V	E	1435	592	NSC
U	C	1435	592	I
T	C	250	420	I
S	C	1095	600	I
R	B	1435	592	I

that are subsequently measured. Control of contaminants, including air leakage, that could enter the main condenser, is described in Section 10.3.5 as part of the secondary side water chemistry program.

Butterfly valves are provided in the circulating water system (CWS) to permit half of the condenser tube bundles to be isolated and removed from service for maintenance.

During anticipated operational occurrences, the main condenser is capable of accepting steam from the TBS while maintaining condenser vacuum, provided the CWS remains in operation and spray water pressure is available if operating conditions require spray water (refer to Section 10.4.4 for a description of the TBS).

The operation of the main condenser supports other system operations within the steam and power conversion system.

10.4.1.3 Safety Evaluation

The main condenser has no nuclear safety-related function.

The design of the main condenser satisfies general design criterion (GDC 60), as it relates to the control of radioactive material releases to the environment. During normal operation and shutdown, the main condenser contains negligible quantities of radioactive contaminants. However, it is possible for the main condenser to become contaminated in the event of primary-to-secondary system leakage.

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- Air and non-condensable gases are discharged from the main condenser by the MCES. Radiological activity of the MCES exhaust is monitored (refer to Section 10.4.2). The radiological aspects of primary-to-secondary leakage, which includes anticipated operating concentrations of radioactive contaminants, are addressed in Section 11.5 and Table 11.5-1, Monitor R-3.
- For the U.S. EPR, no hydrogen buildup is anticipated in the main condenser. Dissolved oxygen is present in the condensate and condenser hotwell inventory, but only trace amounts of this oxygen are released in the condenser, and the amounts are negligible compared to the amount of gas and vapor being evacuated by the MCES. There is no potential for explosive mixtures within the main condenser which would result in excessive releases of radioactivity; therefore, the main condenser design satisfies GDC 60 and is not required to be designed to withstand the effects of an explosion.

Failure of the main condenser and the resulting flooding does not prevent the operation of any essential system because no safety-related equipment is located in the Turbine Building.

Main condenser operation does not directly affect the reactor coolant system. If the main condenser performance is degraded, the turbine backpressure increases. This increase in backpressure causes a lowering of turbine cycle efficiency, which requires

10.4.2 Main Condenser Evacuation System

The main condenser evacuation system (MCES) removes air and non-condensable gases from the main condenser and connected steam side systems during plant startup, cooldown and normal operation.

10.4.2.1 Design Basis

The MCES performs no safety-related function and therefore has no nuclear safety-related design basis.

The MCES is designed to meet the following functional criteria:

- Air and non-condensable gases are removed from the condenser and connected steam side systems during plant startup, cooldown and normal operation.
- Vacuum is established and maintained in the condenser and connected steam side systems during plant startup and normal operation by using mechanical vacuum pumps.

10.4.2.2 System Description

10.4.2.2.1 General Description

The MCES is non-safety related and is located in the Turbine Building. The MCES is used to evacuate air rapidly from the main condenser and connected steam side systems during plant startup, and to continuously remove non-condensable gases during normal operation to maintain optimum condenser performance.

The MCES and air vent system are shown in Figure 10.4.2-1—Main Condenser Evacuation System and Figure 10.4.2-2—Vent System for Air Removal.

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The steam and air mixture extracted from each condenser shell is routed to one of two 100 percent capacity holding vacuum pumps. Discharge from the vacuum pumps is routed by a header to the air vent system, where the radiological activity of the exhausted air is monitored (refer to Section 11.5.4 and Table 11.5-1, Monitor R-3 .

Isolation valves serve to stop air removal from the air cooler tube bundle when the circulating water side of the associated condenser shell is shut off.

The connection of the condensers to the vacuum pumps is controlled by air-operated control valves at the inlet of each vacuum pump. These valves are normally open. The vacuum pumps are connected to the air removal header by air intake isolation valves. The vacuum pumps pass the steam and air mixture to the moisture separators–silencers. As a result of compression, the steam component condenses while the exhausted air is vented through the air vent system into the nuclear auxiliary building ventilation system.

material. The generation rate of hydrogen from these sources is very small compared to the amount of steam in the vapor space in the main condenser and the amount of water vapor in the MCES.

- Dissolved oxygen is present in the condensate and condenser hotwell inventory but only trace amounts of this oxygen is released into the main condenser, and the amounts are negligible compared to the amount of gas and vapor being evacuated by the system.

- There is no potential for explosive mixtures within the MCES, which would result in excessive releases of radioactivity; therefore, the MCES design satisfies GDC 60 and is not required to be designed to withstand the effects of an explosion. The design capacity of the holding (continuously operating) vacuum pumps in the MCES is such that the water vapor content is above 58% by volume of the total mixture. There is no buildup of non-condensable gas in the main condenser because the MCES operates continuously whenever the main condenser is in operation. The vacuum pumps in the MCES are liquid ring vacuum pumps. Cooling water is used to seal the vacuum pumps. The mixture passing through the MCES is at low temperature and high humidity due to contact with the water ring in the vacuum pumps. As indicated in Table 10.1-1—Design Heat Balance for Steam and Power Conversion System Cycle, the average design backpressure is 2.5 inches HgA. Based upon standard steam tables the corresponding saturation temperature is approximately 109° F. The MCES operates at a lower pressure and temperature than the backpressure in the condenser.

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- The exhaust from the MCES is discharged to the air vent system. The exhaust flow is monitored for radioactivity as described in Section 11.5.4 and discharged to the nuclear auxiliary building ventilation system. The radiological aspects of primary-to-secondary leakage, including anticipated releases from the system, are described in Section 11.5.4 (refer to Table 11.5-1, Monitor R-3).

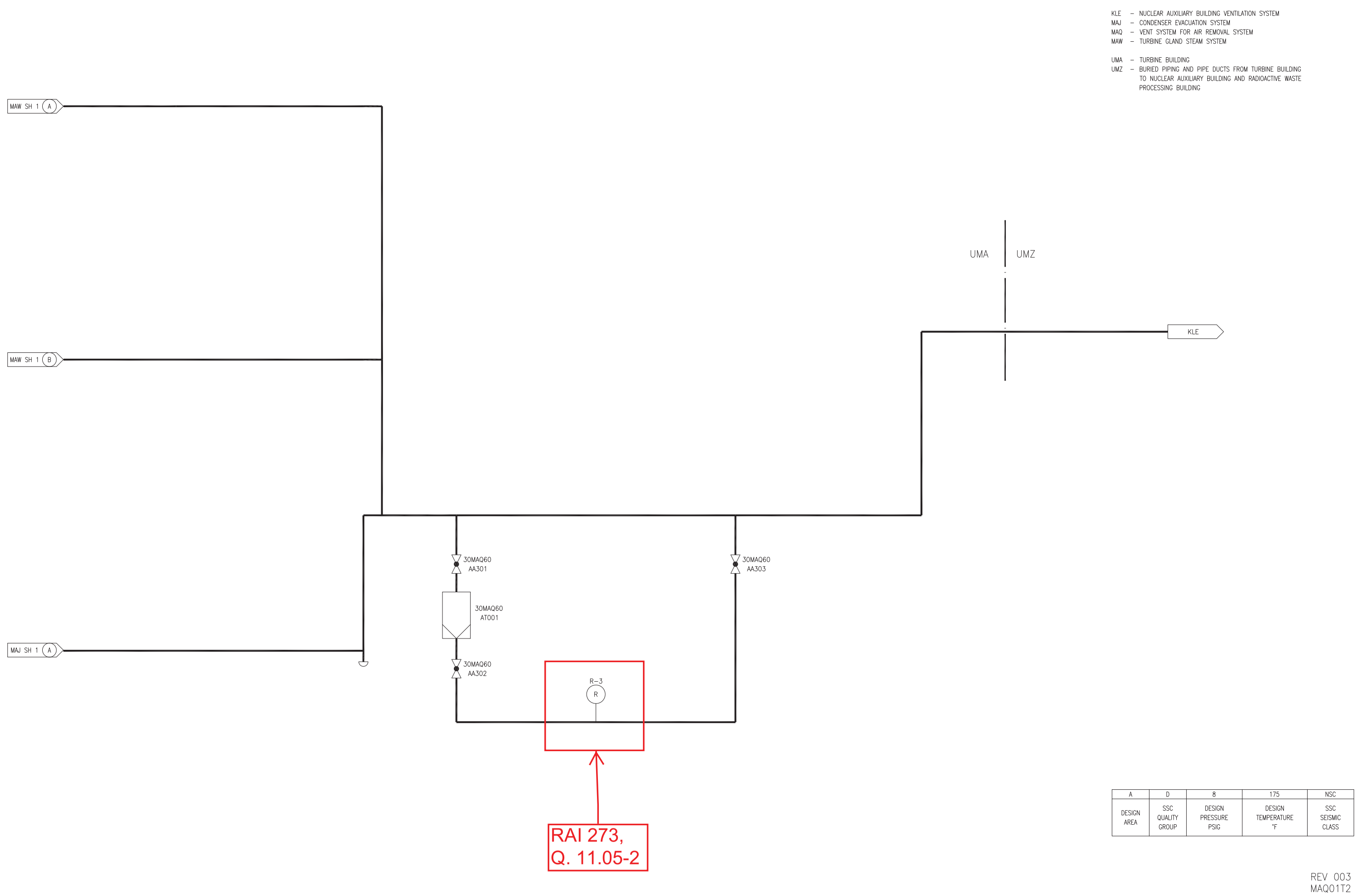
Malfunction of MCES components does not affect the safe operation of the plant or any safety related system.

- MCES operation does not directly affect the reactor coolant system. If the air removal system fails completely, a gradual reduction in condenser vacuum results from the buildup of non-condensable gases. This reduction in vacuum causes a lowering of turbine cycle efficiency, which requires an increase in reactor power to maintain the demanded electrical power generation level. The reactor power increase is limited by the reactor control system, as described in Section 7.7. The reactor protection system, described in Section 7.2, maintains the plant within safe operation limits. If the MCES remains inoperable, condenser vacuum increases to the turbine trip setpoint and a turbine trip is initiated. A loss of condenser vacuum is addressed in Section 15.2.

10.4.2.5 Inspection and Testing Requirements

Inspection and testing of the system is performed prior to plant operation. Refer to Section 14.2 (test abstract #065) for initial plant startup test program.

Figure 10.4.2-2—Vent System for Air Removal



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The steam side of the condenser can be bypassed in the event of gland steam condenser failure. Any condensate produced is passed to the Turbine Building drains system via a loop line upstream of the gland steam condenser isolation valve. This line also drains condensate that accumulates in the event of a tube leak in the gland steam condenser.

10.4.3.2.2 Component Description

~~Table 3.2.2-1 provides the quality group and seismic design classification of components and equipment in the~~ Table 3.2.2-1 provides the seismic design and other design classifications for components in the TGSS. Section 3.2 describes how the guidance of RG 1.26 is implemented for the U.S. EPR.

Condenser

The gland steam condenser receives steam and noncondensable gases from the TGSS and condenses the steam. The gland steam condenser is integrated into the condensate system with condensate passing through the tubing. The drains from the gland steam condenser discharge to the main condenser over a drip leg and through a steam trap.

Exhausters

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The gland steam exhausters produce a slight vacuum in the gland steam condenser to remove air and noncondensable gases that carry over into the gland steam condenser with the leak-off steam from the shaft glands.

Steam and noncondensable gases from the gland steam exhausters are routed to the ~~turbine building air removal system (see Section 10.4.2), where they are monitored for radioactivity and released into the~~ nuclear auxiliary building ventilation system.

Piping

The TGSS piping provides steam for sealing the shafts of the HP and LP turbines. The system piping supplies the sealing steam from the HP turbine to the shaft seals of the HP and LP turbines and onto the gland steam condenser.

10.4.3.2.3 System Operation

Plant Startup and Shutdown

During startup and shutdown, and in the low-load power range, steam from the auxiliary steam system is supplied to the TGSS through the seal steam supply valves as required to maintain the gland steam header pressure. The motor-operated isolation valve fails as is. The pneumatically actuated pressure control valve fails to the position designated by the turbine supplier.

Normal Operation

During normal operation, enough steam escapes through the HP turbine shaft seals that it can be used as a seal steam supply for the shaft seals of the LP turbines. The excess steam escaping at higher loads is dumped to the main condenser through the seal steam leak-off valve. Leak-off steam from the seals of the main stop and control valves is also discharged into the TGSS header. The steam seal leak-off valve is a pneumatically actuated pressure control valve, which fails to the position designated by the turbine supplier.

10.4.3.3 Safety Evaluation

The TGSS performs no safety-related functions and is not required to operate during or after an accident. The design of the TGSS satisfies general design criteria GDC 60 and GDC 64, related to the TGSS design for control and monitoring of release of radioactive materials.

The air and noncondensable gases discharged from the gland steam exhausters are not normally radioactive during plant operation. However, in the event of significant primary-to-secondary system leakage due to a steam generator tube rupture, it is possible for the seal steam to become contaminated resulting in the potential to discharge a radioactively contaminated mixture.

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The gland steam condenser receives steam and noncondensable gases from the TGSS and condenses the steam. Air and non-condensable gases are evacuated from the gland steam condenser and discharged into the air vent system by exhaust fans. The exhaust flow is monitored for radioactivity as described in Section 11.5.4 (refer to Table 11.5-1, Monitor R-3) and discharged to the Nuclear Auxiliary Building ventilation system.

10.4.3.4 Inspection and Testing Requirements

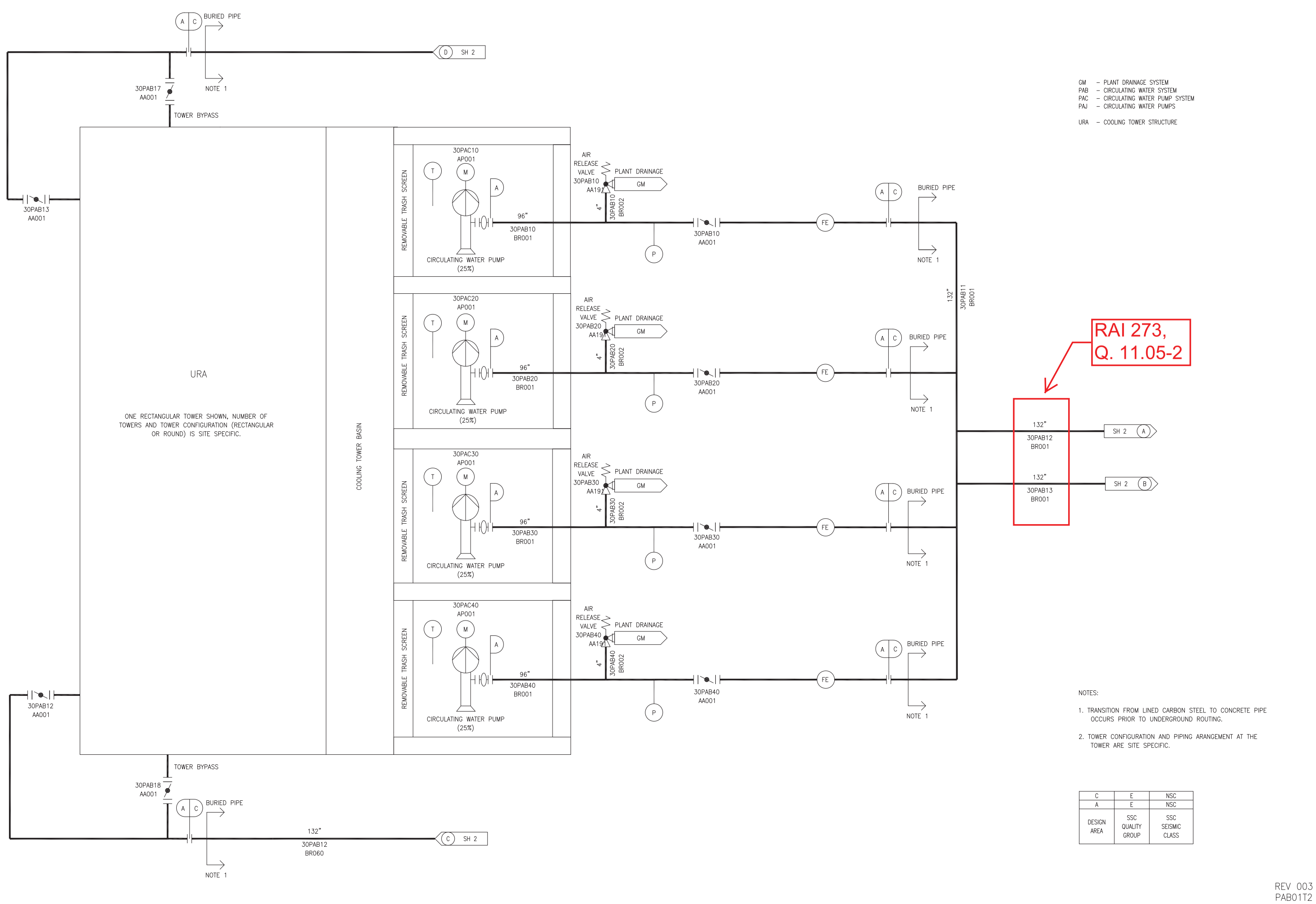
The TGSS components are inspected during construction and functionally tested during plant startup. Refer to Section 14.2 (test abstract #064) for initial plant startup test program. Components of the TGSS are designed to permit periodic inspection and testing during plant operation. Components of the system are monitored during operation to demonstrate satisfactory functioning of TGSS equipment.

10.4.3.5 Instrumentation Requirements

The following TGSS parameters are monitored during plant operation:

- Steam seal header temperature.
- Steam seal header pressure.
- Gland steam exhauster vacuum.

Figure 10.4.5-1—Circulating Water System Flow Diagram Sheet 1 of 2



Polisher vessels are constructed of carbon steel with a protective interior lining, coating or cladding.

Condensate Polisher

The condensate polisher consists of at least four trains of deep mixed bed demineralizers with downstream resin traps. Spent polisher resin is replaced or regenerated offsite. The number and size of the ion exchangers allow functional requirements to be met while permitting replacement of resin in one ion exchanger at a time.

Resin Trap

Resin traps are installed downstream of each ion exchanger to remove resin fines.

Spent Resin Tank

The spent resin tank is used for storage of exhausted or spent resin prior to shipping offsite for regeneration or disposal.

Resin Addition Equipment

Equipment is provided to replace the ion exchange resin.

10.4.6.3 System Operation

The CPS cleans up the condensate during startup to meet condensate and feedwater system water chemistry specifications as described in Section 10.3.5. The condensate is recirculated to the hotwell during startup until the desired water quality is attained. Condensate and feedwater system operation is described in Section 10.4.7.

During power operation, the condensate polishers are used only when abnormal secondary cycle conditions exist. This allows continuous operation of the plant with condenser tube leakage until repairs can be made. Flow through the condensate polisher is controlled by the condensate polisher bypass valve.

Exhausted or spent resin is removed from the polisher vessel and replaced with new or regenerated resin. Resin replacement requires the polisher vessel to be out of service. The standby vessel is placed in service when another vessel needs to be removed from service. Spent resin may be transferred directly to a truck or the spent resin storage tank until it can be removed offsite. Spent resin will normally be nonradioactive and not require any special packaging prior to disposal. In the event the resin becomes contaminated with radioactive material, shielding is provided, if required. Spent resins are shipped offsite per the Process Control Program (PCP). Information about the PCP can be found in Section 11.4.3. Radiation monitors associated with the steam generator blowdown system, main steam system and ~~condenser evacuation~~

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~~system~~Vent System for Air Removal System vent are used to detect secondary side radioactive contamination (refer to Section 11.5, Table 11.5-1, Monitors R-46 through R-49 (Steam Generator BLowdown System), and R-3 (Vent System for Air Removal)).

10.4.6.4 Safety Evaluation

The condensate polishing system has no safety-related function and is not required to function during or after an accident.

The design of the CPS satisfies GDC 14, as it relates to maintaining water quality to avoid corrosion-induced failure of the reactor pressure boundary.

- The CPS functions to remove corrosion products and impurities from the condensate. The secondary water chemistry program is based on the EPRI PWR Secondary Water Chemistry Guidelines (Reference 1). The program description and associated chemistry control parameters is provided in Section 10.3.5.

10.4.6.5 Inspection and Testing Requirements

The CPS components are inspected and tested as part of the initial plant startup. Refer to Section 14.2 (test abstracts #066 and #071) for initial plant startup test program. The system operating parameters are monitored during power operation.

10.4.6.6 Instrumentation Requirements

Instrumentation is provided to measure the pressure drop and outlet conductivity from each resin demineralizer to monitor performance. Should the pressure drop or outlet conductivity become too high, the train is removed from service for resin replacement or regeneration and the spare train is placed in service, if required.

Condensate system sampling points are shown on Figure 10.4.7-1—Condensate and Feedwater System, and described in Section 9.3.2.

10.4.6.7 References

1. EPRI Report 1008224, "Pressurized Water Reactor Secondary Water Chemistry Guidelines," Electric Power Research Institute, Revision 6, December 2004.

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preservice inspection requirements. Refer to Section 6.6 for a description of the inservice inspection program.

10.4.8.6 Instrumentation Requirements

The SGBS instrumentation is provided to facilitate automatic operation, remote control and continuous indication of system parameters.

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Process radiation monitors are provided in the SG blowdown sampling system. These monitors are discussed in Section 11.5.4.3, Monitor R-46 through R-49.

Safety-related isolation functions of the SGBS are performed by the protection system as described above in Section 10.4.8.3.3.

Non-safety related instrumentation and control (I&C) functions are performed by the process automation system.

10.4.8.7 References

1. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Class 2 Components, Subsection NC: Article 7000 “Overpressure Protection,” The American Society of Mechanical Engineers, 2004.

2. ASME Boiler and Pressure Vessel Code, Section XI: “Rules for Inservice Inspection of Nuclear Power Plant Components,” The American Society of Mechanical Engineers, 2004.

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release, the system returns the wastewater in the affected monitoring tank to the five liquid waste storage tanks. This wastewater is subsequently returned to the liquid waste processing system for additional removal of radioactive constituents. If the measured activity of a monitoring tank is within release limits, the water in that tank is discharged to the release line.

A locked, closed valve normally shuts the liquid waste storage system release line. Administrative controls preclude unlocking the valve until activity measurements of the liquid waste held in the monitoring tank are below the concentration limits for release. The release line contains an activity-measurement tank. Radiation sensors (refer to Section 11.5.3.2 and Table 11.5-1, Monitor R-32) are mounted in the activity-measurement tank, and flow sensors are mounted in the liquid waste release piping downstream of the activity-measurement tank; together these sensors continually measure and record the total actual activity and activity release rate during each release of processed liquid waste effluents to the environment. Each radiation sensor can generate control signals that stop the discharge pump and isolate the release path if the sensor detects activity in excess of the anticipated level or release rate. Discrepancies between the two radiation sensors or between the two flow sensors also result in control signals that terminate the discharge and isolate the release line.

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Operator Error or Malfunction

The radiation sensors ~~in~~(refer to Section 11.5.3.2 and Table 11.5-1, Monitor R-32) the liquid waste storage system release line generate alarm signals in the main control room and signal interlocks that close the release line isolation valves to prevent further releases if an operator error or equipment malfunction occurs during release. If these isolation valves are closed, the liquid waste management system is designed with enough redundancy and capacity to operate without discharging until the alarm condition is resolved. Although the evaporator, centrifugal separator, and demineralizer are each separate subsystems in the liquid waste processing system, the configuration of that system provides sufficient redundancy that a failure to one subsystem is covered by another subsystem. Also, sufficient storage capacity exists in the five liquid waste storage tanks and two monitoring tanks to collect up to a week's volume of liquid wastes without processing and release. Operator actions are required to align different liquid waste storage tanks to the various liquid waste processing systems, to align the system for recirculation of a given tank, or to return the contents of a monitoring tank or concentrate tank for additional processing. Administrative control of the locked closed release path adds additional confidence that operator error does not cause inadvertent discharges of liquid waste that contains activity in excess of the limits for release.

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If the activity and pH of the treated wastewater in a monitoring tank are within limits, then the recirculation and discharge pumps are aligned to pump that tank to the release line. The release line contains an administratively controlled, locked-closed upstream isolation valve. Personnel in the main control room maintain custody of the key to this valve and only issue the key upon receipt of a completed analysis demonstrating that the treated wastewater in a monitoring tank is within limits for release. When this valve is opened, the treated wastewater enters the activity-measurement tank in the release line. Radiation sensors (refer to Section 11.5.3.2 and Table 11.5-1, Monitor R-32) in this tank continuously measure and record the activity as the treated wastewater is released. Flow sensors downstream of the activity-measurement tank continuously measure and record the volume and flow rate as the treated wastewater is released. If the total activity indicated by sensors exceeds predetermined limits, control signals are generated automatically to close two downstream isolation valves, close the upstream isolation valve, and shut down the operating recirculation and discharge pump(s).

These components are upstream of both the high point in the release line and a drain tank connected to the release line; automatic closure occurs before the treated wastewater that generated the control signal exits the release line. This equipment prevents the release of treated wastewater that exceeds the dose limits to individual members of the public, as specified in 10 CFR 20.1301 and 10 CFR 20.1302, 10 CFR Part 20, Appendix B, Table 2, Column 2 limits, or the 10 CFR Part 50, Appendix I ALARA design objectives in the event that an operator error initiates an inadvertent discharge.

11.2.2.2 Liquid Waste Processing System Operation

The liquid waste processing system consists of three operations that employ different physical processes to separate radioactive material or chemicals from the wastewater generated during plant operation: the evaporator, the centrifuge, and the demineralizer.

11.2.2.2.1 Evaporator System

The evaporator system treats liquid wastes with high activity but low concentrations of organic chemicals or particulate solids. Typically, Group I wastewater, which contains dissolved boric acid, is sent to the evaporator. However, the five liquid waste storage tanks have discharge piping that connects to the supply header for the evaporator. If a batch of Group II wastewater contains activity that is not bound on solids (meaning it is not removable by the centrifugal separation), the wastewater can also be treated by evaporation.

The evaporator unit is a vapor-compressor evaporator with forced recirculation. An evaporator feed pump takes suction from the liquid waste storage tank that contains returned to the liquid waste storage tanks for treatment. If the wastewater in the monitoring tanks meets activity limits, but not chemistry (pH) criteria for release, chemical additions of either acidic or alkaline solutions can be injected as needed to balance the pH of the wastewater for release.

Recirculation and Discharge Pumps

The recirculation and discharge pumps are centrifugal pumps configured in parallel for redundancy. The recirculation and discharge pumps are used to recirculate treated wastewater in the monitoring tanks for sampling and to facilitate the injection and mixing of chemicals. These pumps also return treated wastewater back to the liquid waste storage tanks (via the recirculation pumps discharge header) if sample analysis results indicate that further processing is required.

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Activity-Measurement Tank

The activity-measurement tank is a small stainless steel tank located upstream of the release line isolation valves. Two radiation sensors (refer to Section 11.5.3.2 and Table 11.5-1, Monitor R-32) are mounted in the release line to continuously measure and record activity discharged during wastewater releases to the environment. Flow sensors are located in the release line downstream of the activity-measurement tank.

Control Valves

The liquid waste storage system employs motor-operated control valves in the influent lines to and effluent lines from the liquid waste storage tanks, the concentrate tanks, the monitoring tanks, and the release line. These valves allow operation of the liquid waste storage system in automatic mode, group or subgroup control mode, interlocked manual mode, and manual mode.

11.2.2.4.2 Liquid Waste Processing System Components

11.2.2.4.2.1 Evaporator System Components

Evaporator Feed Pumps

The evaporator feed pumps are centrifugal pumps that draw wastewater from the liquid waste storage tanks and discharge to the pre-heater. One evaporator feed pump is normally aligned to take suction from the Group I liquid waste storage tanks while the other is normally aligned to take suction from the Groups II and III liquid waste storage tanks; however, either pump can be aligned to draw suction from any of the liquid waste storage tanks.

- Automatic termination and isolation of the release path from the monitor.
- Automatic control functions that govern evaporator system operation.
- Vapor compressor compression ratios.
- Pressure integrity of liquid waste processing system piping and components for pressure transients expected during system operation.

11.2.2.5.2 Preoperational Inspection

The U.S. EPR liquid waste storage and processing systems incorporate several features subject to performance validation by preoperational inspection. Performance validation includes the inspection and testing of the following system installations and components:

- Pump installation and rotation.
- Heat exchanger installation and connection.
- Piping and system pressure integrity testing to confirm that leak tightness and leak rates comply with out-leakage specification.
- Demineralizer resin bed load capacity.
- Proper types and amounts of adsorption and filtration media have been loaded into each demineralizer resin bed.
- Proper filter media, for the pre-filters, ultra-filters, and main filters, have been loaded into the filter housings.

11.2.2.6 Instrumentation Design

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Instrumentation readout is available in the main control room (MCR) and on a local control panel for major components. Instrumentation display for other components is available on a local control panel.

Releases to the environment are monitored using radiation sensors (refer to Section 11.5.3.2 and Table 11.5-1, Monitor R-32) and flow sensors to limit and control offsite releases. See Section 11.2.1.2.3 for a description of this instrumentation.

In accordance with the guidelines of RG 1.143, each tank has level instrumentation that actuates an alarm on detection of high liquid level, allowing action to be taken to divert the flow to a backup tank to avoid a tank overflow. A summary of the tank level indication and associated alarms is provided in Table 11.2-12.

11.2.3 Radioactive Effluent Releases

For the U.S. EPR, releases of radioactive effluent via the liquid pathway only occurs by discharges from the monitoring tanks in the liquid waste storage system. Most of the activity carried into the liquid waste storage and processing systems is removed from the waste stream by a combination of chemical treatments, evaporation, inertial separation, and demineralization and filtration. These treatments may be performed repeatedly, with continuing concentration and chemical treatment cycles, until the wastewater meets release limits. Contaminants removed from the wastewater are transferred to the solid waste management system (see Section 11.4).

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Treated wastewater held in the monitoring tanks must be sampled and analyzed in the laboratory before its release can be authorized. The laboratory analysis confirms that the activity of the wastewater in the monitoring tanks is within release limits. Once the laboratory results have been reviewed and confirmed to be within release limits, release is authorized. During the release, two radiation sensors (refer to Section 11.5.3.2 and Table 11.5-1, Monitor R-32) in the activity- measurement tank and two flow sensors downstream of the tank continually monitor and record the discharge. If the sensors detect activity or an activity release rate in excess of release limits, or if a significant discrepancy exists between the two activity measurements or the two flow measurements, the sensors signal automatic valve closure, which terminates the release. After the isolation valves of the liquid waste storage system, the treated wastewater travels through a double-walled pipe to the discharge canal. The treated waste water is diluted with water from the lined retention pond. The treated wastewater environmental interface occurs at the discharge structure. The discharges from the liquid waste storage system do not interact with the Circulating Water System (CWS).

The physical release location and discharge configuration for treated effluent are site-specific and plant-specific. Refer to Section 11.2.3.3 for the related COL item.

11.2.3.1 Discharge Requirements

Discharge requirements consist of liquid radioactive waste activity, flow monitor alarm settings, and automatic isolation settings. These requirements are established for each batch of monitoring tank treated wastewater to meet the ALARA design objectives.

11.2.3.2 Estimated Annual Releases

The GALE Code (Reference 1) was used to provide an estimate of annual releases from the U.S. EPR. Input parameters used in the GALE code model for the U.S. EPR are presented in Table 11.2-3—Liquid and Gaseous Effluent Input Parameters for the GALE Computer Code. Liquid releases (for a single plant unit) in units of Curies/year at the liquid effluent discharge point are presented in Table 11.2-4—Releases to Liquid Effluent Discharge Point (Ci/yr) Calculated by GALE Code.

11.2.3.3 Release Points and Dilution Factors

The liquid waste storage system has a single release point. The release is further diluted to meet the ALARA design objectives of 10 CFR Part 50, Appendix I. This regulation specifies maximum annual values for dose and dose commitment for individuals in an unrestricted area from the pathways of exposure. The U.S. EPR complies with these values with a dilution flow of 100 cubic feet per second (cfs) without additional downstream dilution. Since dilution is site dependent, discharge flow rates vary for each release.

The activity in the liquid effluent is diluted by two potential means prior to reaching a given dose receptor. The first is the mixing that occurs in the discharge canal, prior to the effluent reaching the plant outfall. The flowrate for this discharge dilution is site-specific, and may be provided by cooling tower blowdown, dilution pumps, and/or other plant discharges. The second dilution source is the mixing with, and subsequent dilution by, the receiving water body prior to reaching the dose receptor (e.g., fish, drinking water supply intake). The value of this dilution is also site-specific and varies with factors such as distance between the outfall and the dose receptor, hydrological mixing characteristics of the receiving body, and design and location of the outfall structure.

The combination of pre-outfall dilution from the discharge flowrate and the post-outfall mixing after the liquid effluent reaches the receiving water body determines the effective dilution of the radioactive effluents. For the generic design calculation of doses from liquid effluents, it is assumed that the discharge flow rate is 100 cfs and that no further mixing or dilution occurs beyond the plant outfall. However, equivalent effective dilution may be achieved by various combinations of pre-outfall dilution from the discharge flowrate and post-outfall mixing, where a reduction in discharge flowrate is offset by a proportional increase in post-outfall dilution.

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The physical release location and dilution factors for treated effluent are site-specific. A COL applicant that references the U.S. EPR design certification will provide site-specific information on the release pathway including a detailed description of the discharge path and plant sources of dilution, the discharge flow rate and dilution factors at or beyond the point of discharge.

11.2.3.4 Estimated Doses

11.2.3.4.1 Liquid Pathways

The LADTAP II computer program (Reference 2) was used to calculate doses to the maximally exposed individual (MEI) from liquid effluents. LADTAP II implements

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Table 11.2-13—Detailed Dose Commitment Results by Age Group and Organ due to Liquid Effluent Releases
Sheet 1 of 2

Pathway	Skin (mrem/yr)	Bone (mrem/yr)	Liver (mrem/yr)	Total Body (mrem/yr)	Thyroid (mrem/yr)	Kidney (mrem/yr)	Lung (mrem/yr)	GI-LLI (mrem/yr)
Fish								
Adult		2.10E-01	3.87E-01	2.90E-01	2.56E-01	1.46E-01	6.10E-02	6.74E-02
Teen		2.21E-01	3.92E-01	1.70E-01	2.37E-01	1.44E-01	6.35E-02	5.13E-02
Child		2.74E-01	3.42E-01	7.41E-02	2.45E-01	1.21E-01	5.07E-02	2.71E-02
Drinking								
Adult		6.61E-03	8.21E-01	8.18E-01	1.40E+00	8.20E-01	8.13E-01	8.68E-01
Teen		6.44E-03	5.80E-01	5.76E-01	1.08E+00	5.79E-01	5.73E-01	6.14E-01
Child		1.87E-02	1.12E+00	1.10E+00	2.35E+00	1.11E+00	1.10E+00	1.14E+00
Infant		2.20E-02	1.10E+00	1.08E+00	3.05E+00	1.09E+00	1.08E+00	1.10E+00
Shoreline								
Adult	1.75E-03	1.50E-03	1.50E-03	1.50E-03	1.50E-03	1.50E-03	1.50E-03	1.50E-03
Teen	9.79E-03	8.35E-03	8.35E-03	8.35E-03	8.35E-03	8.35E-03	8.35E-03	8.35E-03
Child	2.05E-03	1.75E-03	1.75E-03	1.75E-03	1.75E-03	1.75E-03	1.75E-03	1.75E-03
Irrigated Foods								
Vegetables								
Adult		6.99E-03	2.98E-01	2.96E-01	3.77E-01	2.94E-01	2.90E-01	3.56E-01
Teen		1.18E-02	3.69E-01	3.59E-01	4.84E-01	3.62E-01	3.55E-01	4.39E-01
Child		2.82E-02	5.86E-01	5.65E-01	8.19E-01	5.74E-01	5.62E-01	6.28E-01
Leafy Vegetables								
Adult		9.50E-04	3.69E-02	3.65E-02	6.96E-02	3.64E-02	3.57E-02	4.43E-02
Teen		8.69E-04	2.47E-02	2.40E-02	5.09E-02	2.43E-02	2.37E-02	2.96E-02
Child		1.56E-03	2.94E-02	2.84E-02	6.86E-02	2.89E-02	2.82E-02	3.16E-02

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Table 11.2-13—Detailed Dose Commitment Results by Age Group and Organ due to Liquid Effluent Releases
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Pathway	Skin (mrem/yr)	Bone (mrem/yr)	Liver (mrem/yr)	Total Body (mrem/yr)	Thyroid (mrem/yr)	Kidney (mrem/yr)	Lung (mrem/yr)	GI-LLI (mrem/yr)
Milk								
Adult		5.36E-03	1.82E-01	1.79E-01	3.35E-01	1.76E-01	1.73E-01	1.74E-01
Teen		9.57E-03	2.40E-01	2.31E-01	4.82E-01	2.31E-01	2.26E-01	2.26E-01
Child		2.27E-02	3.82E-01	3.61E-01	8.65E-01	3.66E-01	3.58E-01	3.57E-01
Infant				5.45E-01	1.78E+00			
Meat								
Adult		1.11E-02	6.22E-02	6.33E-02	6.68E-02	8.18E-02	6.13E-02	7.39E-01
Teen		9.30E-03	3.73E-02	3.79E-02	4.05E-02	5.38E-02	3.66E-02	4.59E-01
Child		1.75E-02	4.52E-02	4.65E-02	5.03E-02	6.70E-02	4.43E-02	3.02E-01
Total								
Adult	1.75E-03	2.43E-01	1.79E+00	1.68E+00	2.51E+00	1.56E+00	1.44E+00	2.25E+00
Teen	9.79E-03	2.67E-01	1.65E+00	1.41E+00	2.38E+00	1.40E+00	1.29E+00	1.83E+00
Child	2.05E-03	3.64E-01	2.51E+00	2.18E+00	4.40E+00	2.27E+00	2.14E+00	2.49E+00
Infant				1.63E+00	4.83E+00			

Figure 11.2-1—Liquid Waste Storage System

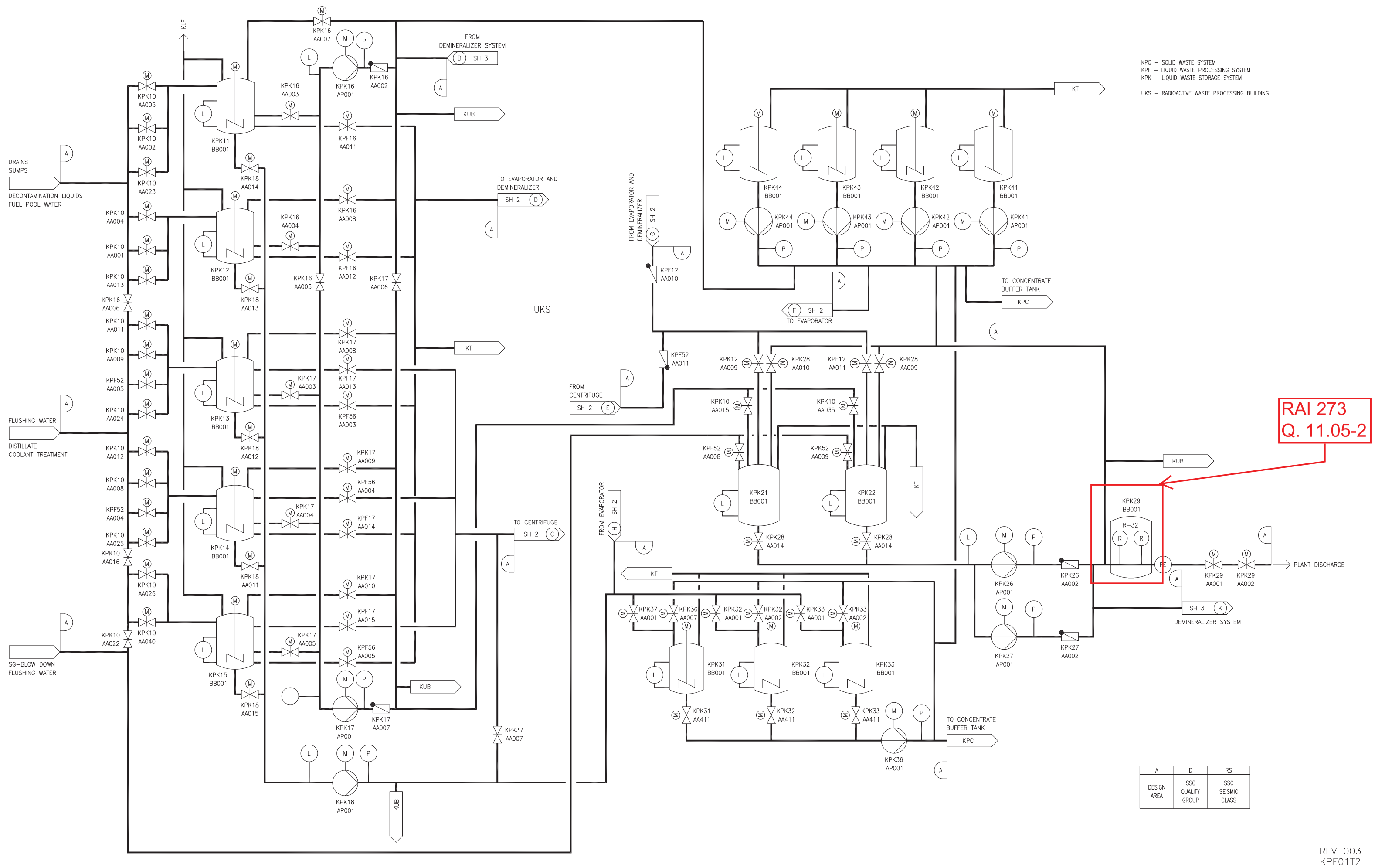

Figure 11.3-1—Gaseous Waste Processing System - Normal Operation

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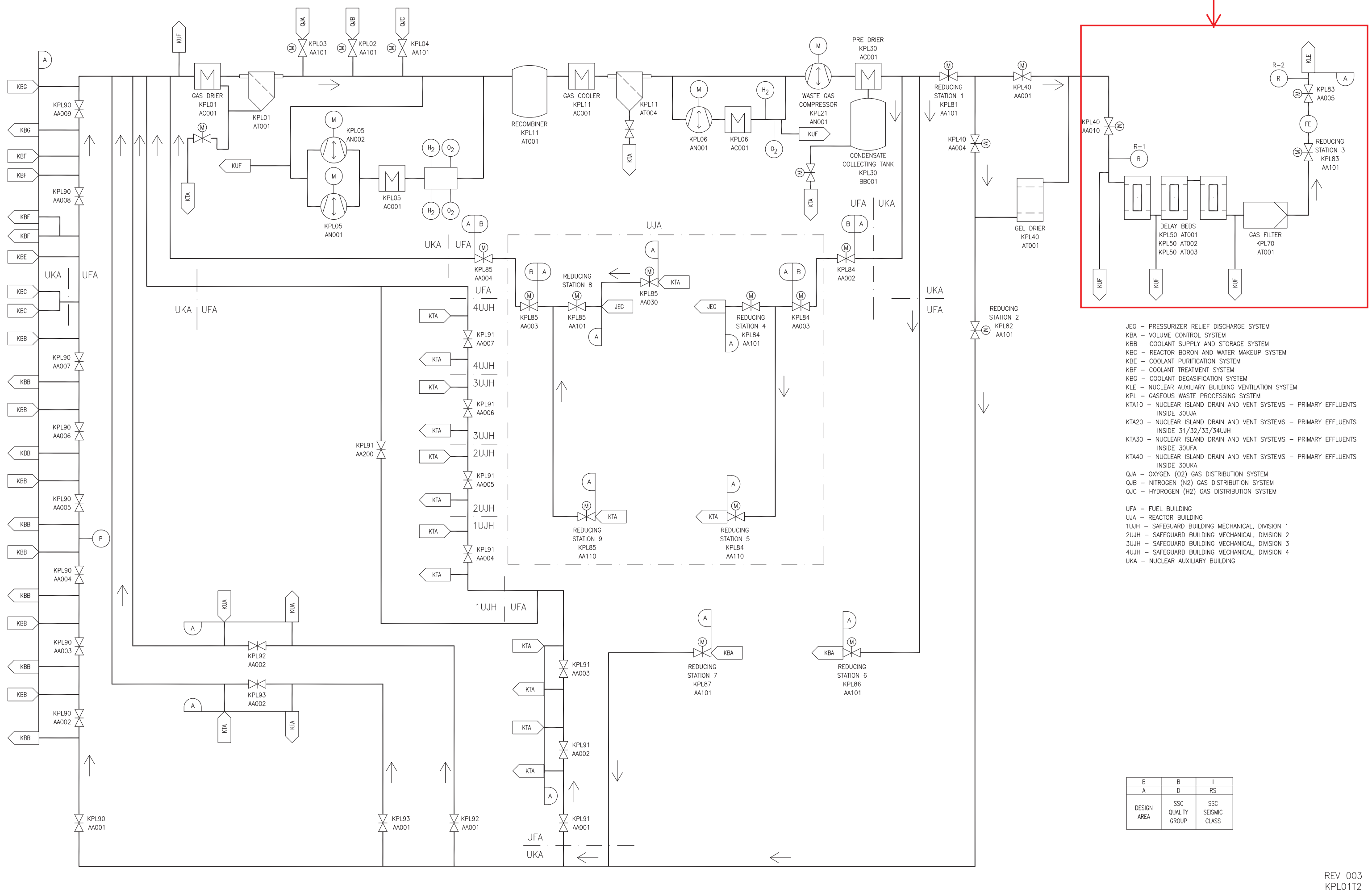


Figure 11.3-2—Gaseous Waste Processing System - Gaseous Waste Sources

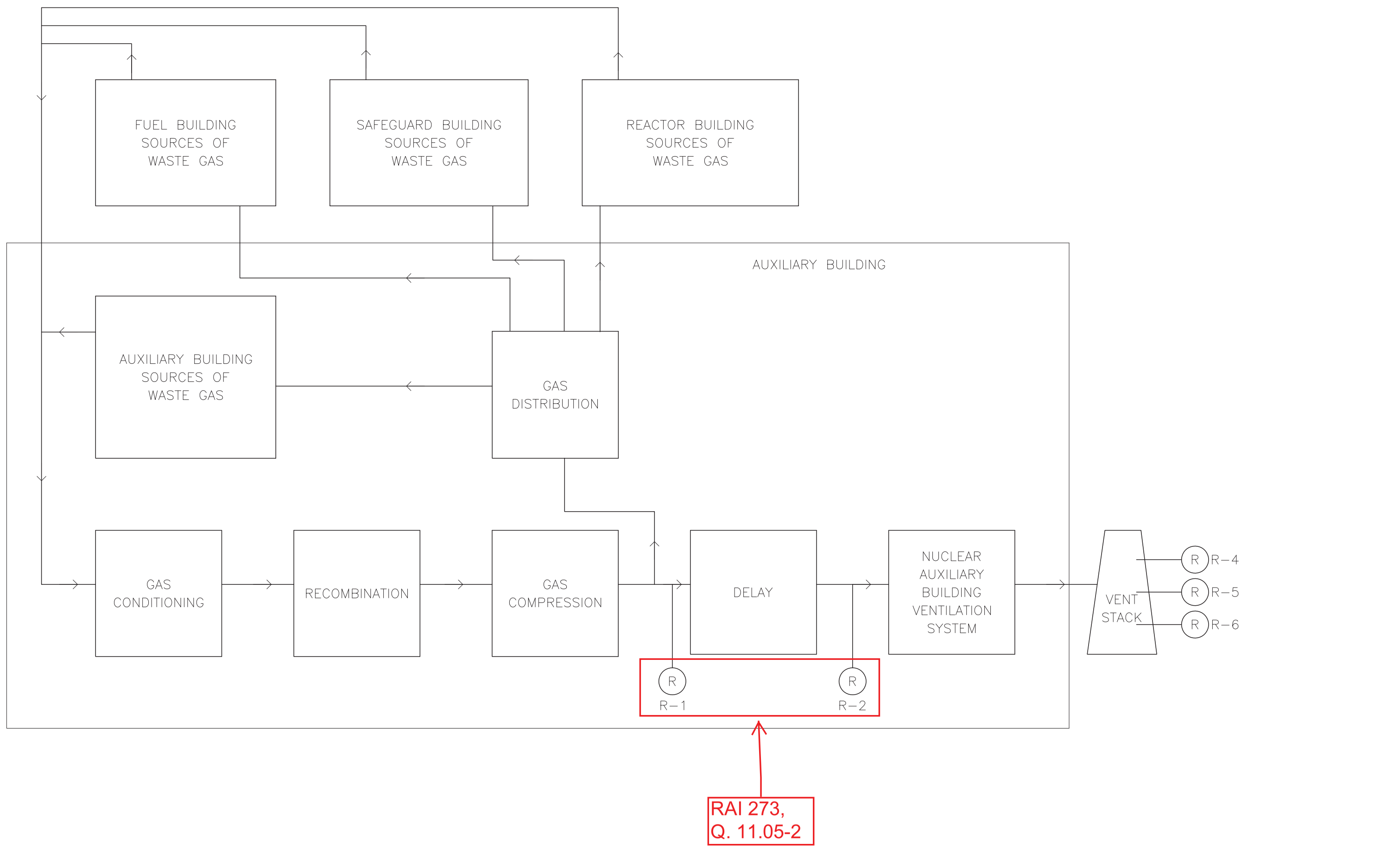
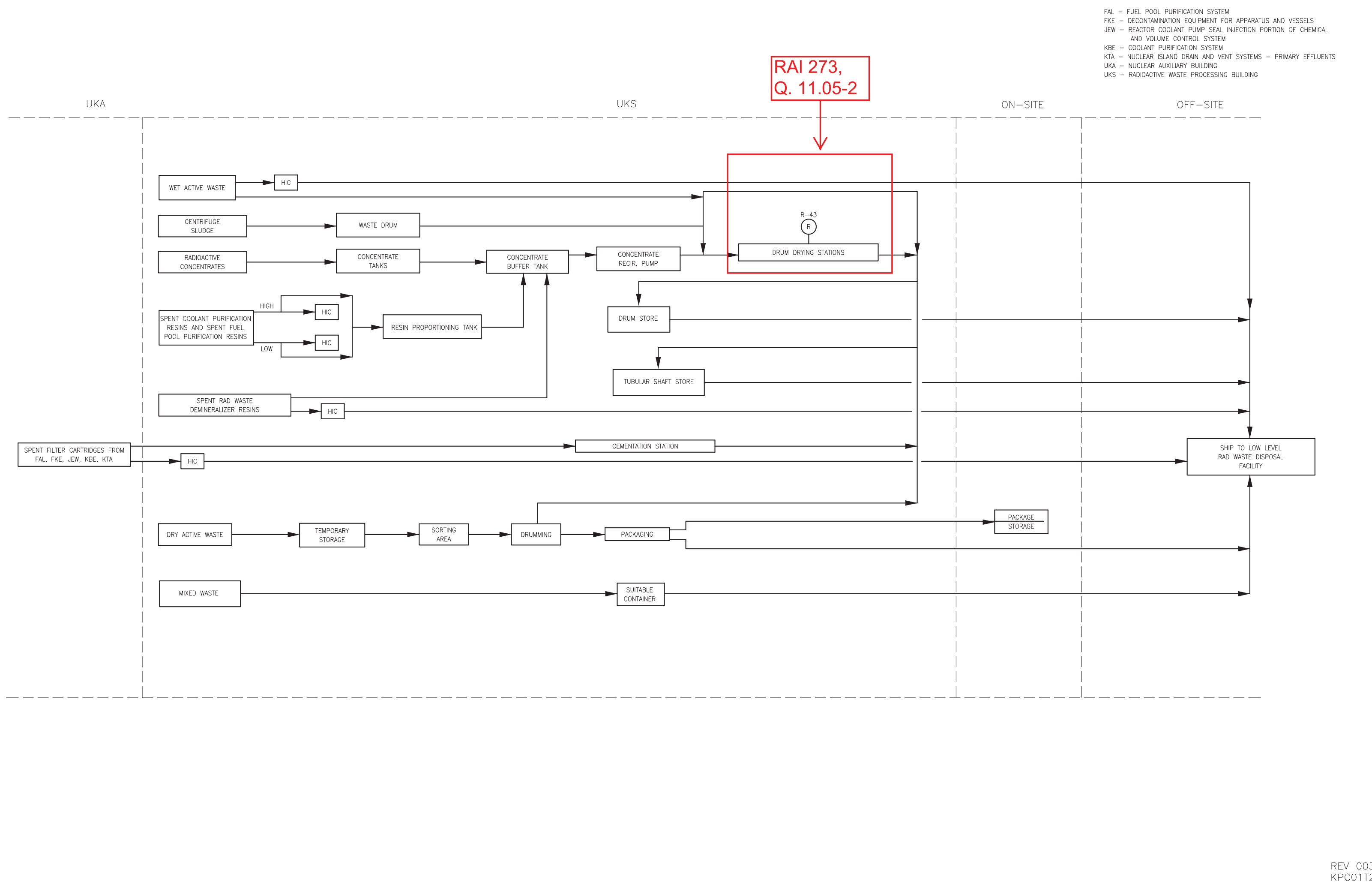


Figure 11.4-1—Solid Waste Management Flow Diagram



leakage to the containment atmosphere, cooling water systems, and the secondary side of the steam generators. Process monitors also provide alarm and gross indication of the extent of failed fuel. They also monitor radioactive waste systems and associated handling areas to detect and alarm conditions that may result in loss of residual heat removal capability and excessive radiation levels. The process monitors that provide a signal for the actuation of engineered safety feature (ESF) systems are designed and qualified to the same criteria as the ESF system. Similarly, the monitors that provide for the actuation of non-ESF systems are designed and qualified to the same criteria as the non-ESF system. This function complies with applicable portions of 10 CFR Part 50, Appendix A, GDC 63.

The effluent radiological monitoring and sampling systems operate continuously during both intermittent and continuous discharges of potentially radioactive plant effluents, in compliance with RG 1.21. The system allows verification of several discharge requirements:

- The most restrictive anticipated radionuclides are at effluent concentrations below the limits specified in Table 2 of Appendix B of 10 CFR Part 20.
- Effluents meet ALARA design objectives of 10 CFR Part, 50 Appendix I.
- Effluents comply with 10 CFR 20.1302 dose limits.
- Effluents comply with EPA environmental radiation standards contained in 40 CFR Part 190.

The effluent radiological monitoring and sampling systems alarm and automatically terminate the release of effluents when radionuclide concentrations exceed the specified limits. The effluent monitors that provide a signal for the actuation of ESF systems are designed and qualified to the same criteria as the ESF system. Similarly, the monitors that provide for the actuation of non-ESF systems are designed and qualified to the same criteria as the non-ESF system. This design complies with 10 CFR Part 50, Appendix A, GDC 60, which requires that "the design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents." Effluent radiological monitors provide sufficient radioactivity release data to prepare effluent release reports required by RG 1.21.

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For monitors that provide an automatic control feature, the isolation or diversion valves or ventilation dampers will be downstream of the monitor so that upon detection of elevated radioactivity levels in the effluent stream will terminate releases by closure of the valves or ventilation dampers in a timely manner to isolate the downstream process stream from further contamination.

11.5.2 System Description

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The effluent and process radiological monitoring and sampling systems consist of radiation detectors connected to local black boxes. Each black box processes the detector signal, performs alarm or control functions, and transmits the signal to a control room information and control system. Monitoring systems alarm when setpoint limits are exceeded or when the system becomes inoperable (see Section 7.1.1 for I&C architecture). Alarms are located both locally and in the control room. Information regarding subsystem checks, tests, and maintenance may be found in Section 7.1.1.5.5.

The detector procurement specifications will require purging or flushing taps for cleaning the sampling pathway within the detector with clean air or water. The design allows the detector to be taken offline and isolated. Then, either the detector is replaced or removed for maintenance. Maintenance technicians will bag and tag the detector as required, transport it to the equipment decontamination facility for decontamination and purging or flushing, and transport it to the instrument shop for repair, cleaning, calibration, and functional checks, as appropriate. The detector would then be reinstalled.

Purge or flush fluids will be captured in the equipment decontamination facility (see Section 12.3.1.6) and sent to the liquid waste management system (LWMS). Gases are routed through a monitored exhaust. Provisions to prevent cross-contamination of purge and flush supply systems are described in Section 12.3.6.5, which demonstrates isolations to prevent cross-contamination of distributed purge and flush supply systems.

AREVA NP Inc. has designed safety-related process and effluent radiological monitoring and sampling systems in accordance with the following criteria:

- Radiation detectors and black boxes are powered from the uninterruptible power supply system; sample pumps and heat-tracing systems are powered from Class 1E power.
- Components are environmentally qualified as applicable. Section 3.11 addresses the environmental qualification of instrumentation.
- Components are seismically qualified as applicable. Sections 3.10 and 3.11 address the qualification of instrumentation.
- Systems comply with the fire protection criteria addressed in Section 9.5.
- Multiple (redundant) systems are used and are physically separated in accordance with criteria addressed in Section 8.3.2.

Process and effluent radiological monitoring and sampling systems that sample airborne radioactive materials are designed in accordance with the general principles and guidance contained in ANSI Standard N.13.1-1999 (Reference 1). Use of this ANSI standard is in accordance with RG 1.21.

Refer to Section 12.3.6.5.4 for radioactive waste management system design features which demonstrate compliance with the requirements of 10 CFR 20.1406.

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A COL applicant that references the U.S. EPR design certification will fully describe, at the functional level, elements of the process and effluent monitoring and sampling programs required by 10 CFR Part 50, Appendix I and 10 CFR 52.79(a)(16). This program description, Offsite Dose Calculation Manual (ODCM), will specify how a licensee controls, monitors, and performs radiological evaluations of releases. The program will also document and report radiological effluents discharged to the environment. NEI 07-09A (Reference 10) is an alternate means of demonstrating compliance with GL 89-01 and SECY 05-0197 until a plant and site-specific ODCM is developed under a license condition. The lower limits of detection (LLD) for liquid and gaseous process monitors and detection sensitivities for liquid and gaseous process monitors will be calculated in accordance with the methodology provided in the ODCM.

A COL applicant that references the U.S. EPR design certification is responsible for deriving PERMSS subsystem's lower limits of detection or detection sensitivities, and set-points (alarms and process termination/diversion) for liquid and gaseous process radiation monitoring equipment not covered by the ODCM based on plant and site-specific conditions and operating characteristics of each installed radiation monitoring subsystem.

A COL applicant that references the U.S. EPR design certification is responsible for developing a plant-specific process and effluent radiological sampling and analysis plan for systems not covered by the ODCM, including provisions describing sampling and analytical frequencies, and radiological analyses for the expected types of liquid and gaseous samples and waste media generated by the LWMS, GWMS, and SWMS.

11.5.3 Effluent Monitoring and Sampling

Sections 12.1 and 12.3 describe how the ALARA provisions of RG 8.8 and RG 8.10 are implemented in system designs and operation to comply with occupational dose limits of 10 CFR 20.1201 and 10 CFR 20.1202 and occupational limits of Table 1, annual limit on intake (ALI) and derived air concentration (DAC), of Appendix B to 10 CFR Part 20.

11.5.3.1 Gaseous Effluents

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Compartment exhaust air from the controlled area and system exhaust air are discharged into the vent stack exhaust. The gaseous effluent monitoring and sampling system monitors the Reactor Containment Building (including the low flow purge subsystem and the internal filtration subsystem), the Fuel Building, the Annulus Ventilation System, the Nuclear Auxiliary Building, the mechanical area of the Safeguard Buildings, the controlled area of the Access and Radioactive Waste Processing Buildings and the vent stack. The U.S. EPR ventilation systems are described in Section 9.4.~~Compartment exhaust air from the controlled area and system exhaust air are discharged into the vent stack exhaust. The gaseous effluent monitoring and sampling system monitors the Reactor Containment Building, the Fuel Building, the Nuclear Auxiliary Building, the mechanical area of the Safeguard Buildings, the controlled area of the Access and Radioactive Waste Processing Buildings and the vent stack. The U.S. EPR ventilation systems are described in Section 9.4.~~

Continuously operating measurement devices monitor vent stack gaseous effluent for noble gases, aerosol, and iodine. Samples are also collected for laboratory analysis of tritium, noble gases, aerosols and iodine. Two permanently installed monitoring and sampling systems provide gaseous samples from the vent stack to the measurement devices. Each sampling system has separate sample lines and independent nozzle arrays, located within the vent stack, to retrieve gaseous samples. The system is designed in accordance with ANSI N13.1-1999 (Reference 1) to provide extraction of a representative gaseous sample. The vent stack gaseous effluent monitoring system functional location is shown in Figure 11.5-1—Radioactive Effluent Flow Paths With Process and Effluent Radiation Monitors, as well as Figure 11.3-1—Gaseous Waste Processing System - Normal Operation, and Figure 11.3-2—Gaseous Waste Processing System - Gaseous Waste Sources.

The ODCM (see Section 11.5.2) includes the following information for each location subject to routine gaseous effluent sampling: the sampling frequency and the analytical process and sensitivity for selected radioanalytical methods and types of sampling media.

The gaseous effluent monitoring and sampling system has the following general characteristics:

- Noble gas activity is monitored with gamma and beta-sensitive detectors. The gross output of the monitor is periodically normalized to the radionuclide composition by performing a gamma-spectroscopic analysis on a representative grab sample.

- Aerosol activity is monitored with the use of an aerosol filter through which sample flow is continuously maintained. Aerosol particles are removed by the filter, which is monitored by a gamma-sensitive detector.
- Iodine activity is monitored by a dual filter for organic and inorganic iodine. Gamma-sensitive detectors monitor each filter.

For both aerosol and iodine monitoring, the gross outputs of the monitors are normalized by laboratory analysis of a duplicate set of filters installed in parallel with the primary ones. Measurement ranges of noble gas, aerosol, and iodine monitors are shown in Table 11.5-1—Radiation Monitor Detector Parameters. The gaseous effluent radiological monitoring and sampling for the vent stack does not perform automatic actions. The system monitors, records, and alarms in the control room if monitored radiation levels increase beyond specified setpoints.

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The ODCM (see Section 11.5.2) contains the standard radiological gaseous effluent controls for the plant. This includes a description of how effluent release rates will be derived and parameters used in setting instrumentation alarm setpoints to control or terminate effluent releases in unrestricted areas that are above the effluent concentrations in Table 2 of Appendix B to 10 CFR Part 20. In addition, the ODCM describes how the guidance of NUREG-1301 (Reference 8) and NUREG-0133 (Reference 9) were used in developing the bases of alarm setpoints. ~~NEI 07-09A (Reference 10) is an alternate means of demonstrating compliance with GL 89-01 and SECY 05-0197 until a plant and site-specific ODCM is developed under a license condition.~~

11.5.3.1.1 Gaseous Waste Processing System (GWPS)

Airborne releases can be limited by restricting reactor coolant leakage and by limiting the concentrations of radioactive gases in the reactor coolant system (RCS), however, these gases escape from reactor coolant during normal operations and require treatment in the gaseous waste processing system. The purpose of the monitoring system is to control the subsequent release of processed waste gases to the atmosphere in compliance with regulatory limits. See Section 11.3 for a complete discussion. Gamma-sensitive radiation detectors continuously monitor the gaseous waste delay beds. A gamma-sensitive radiation detector is located upstream of the beds (R-1) and a beta-sensitive radiation detector is located downstream (R-2). The upstream monitor measures the gamma radiation emitted by the radionuclides entering the beds, and provides station personnel with an indication of the amount of radioactivity entering the beds. The downstream monitor is a beta-sensitive instrument, because krypton-85 generally forms the main constituent (about 95 percent) of the radioactive noble gases. The gaseous waste disposal radiation monitoring system provides control room and local indication only. This system initiates an automatic action to close the discharge valve on high activity.

The radiation monitors in the gaseous waste processing system are shown on Figure 11.3-1 and Figure 11.3-2. In addition, Section 9.4 includes simplified diagrams of the ventilation systems that show the radiation monitors. Measurement ranges of the gaseous waste disposal radiation monitoring system are shown in Table 11.5-1. The safety classification for these instruments is non-safety, augmented quality. Portable check sources are used to check the detector and instrumentation circuitry.

11.5.3.1.2 Main Condenser Evacuation System (MCES)

Noncondensable gases (air and noble gases) in the secondary steam system are continuously removed during operation by the main condenser evacuation system (see Section 10.4.2). These gases discharge to the vent stack via the main condenser evacuation system air vent sub-system and via the nuclear auxiliary building ventilation system. To monitor noble gas radioactivity, the monitoring system extracts part of the flow from the vent line (see Figure 10.4.2-2 and Figure 11.5-1) and passes it through a measuring vessel with a beta-sensitive detector (R-3). If the monitoring system detects noble gas radioactivity in the secondary steam system, then it provides local and control room indication and alarms. The task of the monitor is to provide information and alarm about any increase of activity in the secondary system. This alarm is an indication of breach of fuel cladding, primary coolant boundary, or containment leak. Table 11.5-1 shows the measurement ranges of the main condenser evacuation system. This system does not initiate automatic actions.

Measurement ranges of the main condenser evacuation radiation monitoring system are shown in Table 11.5-1. The safety classification for these instruments is non-safety. Portable check sources are used to check the detector and instrumentation circuitry.

11.5.3.1.3 Sampling Activity System

The vent stack exhaust air is sampled by means of a redundant nozzle array which meets requirements for obtaining a representative sample to determine the release of radioactive substances in gaseous and aerosol-bound form. The sampling points are located down-stream of the last compartment air outlet into the vent stack at a point where there has been achieved an adequate mixture of the entire exhaust air flow.

The sampling activity system consists of three arrays of detectors at measurement points R-4, R-5, and R-6, which measure gaseous activity in the vent stack as shown in Figure 11.3-2 and Figure 11.5-1. The R-4 continuous effluent monitors consist of one noble gas, one iodine, and one aerosol monitors plus one iodine and one aerosol accident monitors in the vent stack. Effluent grab sample provisions consist of one noble gas, one iodine, one aerosol sample points, plus one noble gas, one iodine, and one aerosol accident sample points, and an H-3, C-14 sample point in the vent stack. The R-5 continuous effluent monitors consist of two noble gas monitors in the vent stack. Effluent grab sample provisions consist of one iodine and one aerosol sample points, plus one noble gas, one iodine, and one aerosol accident sample points, and an H-3, C-14 sample point in the vent stack. Tritium is monitored by continuously taking a sample from the exhaust air. The R-6 continuous effluent monitors consist of two noble gas accident monitors in the vent stack.

Coupled with a vent stack air flow rate instrument in the immediate vicinity of the detector arrays, the monitors provide the data necessary to determine the release of gaseous effluents through the vent stack. This monitoring system provides local and control room indication and alarms. This system does not initiate automatic actions.

Measurement ranges of the sampling activity system are shown in Table 11.5-1. The safety classification for these instruments is non-safety. Portable check sources are used to check the detector and instrumentation circuitry.

11.5.3.1.4 Containment Building Ventilation System (CBVS) - Low Flow Purge

Aerosol monitors, iodine monitors and noble gas monitors are used for monitoring airborne radioactivity in the cells exhaust air and in the exhaust air of the containment before filters. Especially for the containment atmosphere, an additional tritium monitor is used. Releases of radioactivity into compartment air and contamination of the compartment air in the course of plant operation are monitored by this equipment. Airborne radioactivity monitoring serves for personnel protection and overall plant monitoring.

The CBVS low flow purge is described in Section 9.4.7.2. The CBVS monitoring system consists of three sets of detectors at measurement points R-7, R-8, and R-9, which measure gaseous activity in the ventilation system as shown in Figure 9.4.7-2 and Figure 11.5-1. Measurement points R-7 and R-8 are upstream of the filtration trains and point R-9 is downstream of the filtration trains.

The R-7 continuous effluent monitors consist of one noble gas, one iodine, and one aerosol monitors. Effluent grab sample provisions consist of one aerosol sample point. The R-8 continuous effluent monitors consist of one tritium monitor with a beta-sensitive detector and effluent grab sample provisions. The R-9 continuous effluent monitors consist of two noble gas monitors. This monitoring system provides local and control room indication and alarms. This system automatic actions are described in Table 11.5-1.

Measurement ranges of the CBVS monitoring system are shown in Table 11.5-1. The safety classification for the R-7 and R-8 instruments is non-safety. The safety classification for the R-9 instruments is non-safety, augmented quality. Built-in check sources are used to check the detector and instrumentation circuitry to eliminate the need for special containment entries for circuit checks.

11.5.3.1.5 Containment Building Ventilation System (CBVS) - Internal Filtration

The internal filtration subsystem limits the release of radioactive material by reducing radioactive iodine contamination inside the equipment compartment. The CBVS internal filtration is described in Section 9.4.7.2. The CBVS monitoring system consists of three detectors and sample station at measurement point R-10 shown in Figure 9.4.7-3 and Figure 11.5-1. The R-10 continuous process monitors consist of one noble gas, one iodine, and one aerosol monitors. This aerosol monitor is used for RCS leakage detection to satisfy TS 16.3.4.14. Grab sample provisions consist of one aerosol sample point. This monitoring system provides local and control room indication and alarms. This system does not initiate automatic actions.

Measurement ranges of the CBVS monitoring system are shown in Table 11.5-1. The safety classification for the R-10 instruments is non-safety, augmented quality. A built-in check source is used to check the detector and instrumentation circuitry to eliminate the need for special containment entries for circuit checks.

11.5.3.1.6 Nuclear Auxiliary Building Ventilation System (NABVS)

The system provides conditioned air to the Nuclear Auxiliary Building (NAB), Fuel Building (FB), Containment Building, and the annulus area between the Containment Building and the Shield Building. The exhaust air from the NAB, FB, Safeguard Building (SB), Containment Building, and the annulus is processed through the NABVS filtration trains prior to release to the environment via the vent stack. Consequently, monitoring the exhaust air from each of these buildings also takes place in the NABVS. The NABVS is described in Section 9.4.3.

The NABVS monitoring system consists of six measurement points, R-11, R-12, R-13, R-14, R-15, and R-16. R-11 (NAB cell 1), R-12 (NAB cell 2), and R-13 (NAB cell 3) are shown in Figure 9.4.3-3 and Figure 11.5-1. R-14 (hot workshop) and R-15 (laboratory) are shown in Figure 9.4.3-2. R-16 (laboratory exhaust from iodine filter train) is shown in Figure 9.4.3-5.

The R-11, R-12, and R-13 measurement points continuous process monitors each consist of one noble gas, one iodine, and one aerosol monitors. These three measurement points grab sample provisions consist of one aerosol sample point at each measurement point. This monitoring system provides local and control room indication and alarms. The monitors for each of these measurement points initiate an automatic control feature that diverts the ventilation air flow to the NABVS iodine filter train on high activity. This is a non-safety function. The R-14 and R-15 measurement points continuous process monitors each consist of one aerosol monitor. These two measurement points grab sample provisions consist of one aerosol sample point at each measurement point. This monitoring system provides local and control room indication and alarms. The monitors for these measurement points do not initiate automatic actions. The R-16 measurement point has no monitors and the grab sample provisions consist of one iodine and one aerosol sample points in the laboratory exhaust that has been processed through the iodine filter train as shown in Figure 9.4.3-5.

Figure 9.4.3-3 depicts how the various building ventilation systems are combined into the NABVS, filtered, and passed through the NABVS iodine filter trains as required and sent to the vent stack. In addition to measurement points R-11, R-12, and R-13 described above, measurement points R-17 and R-18 (Fuel Building ventilation system) and measurement point R-25 (Safeguard Building ventilation system) are shown at the interface between the ventilation systems. An unmonitored line at the bottom of the figure shows the Reactor Building exhaust. This is the Reactor Building full flow purge exhaust which only operates during Mode 5. Fuel will have been off-loaded and the Reactor Building may be open to the outside atmosphere, and no monitoring is required.

Measurement ranges of the NABVS monitoring system are shown in Table 11.5-1. The safety classification for the R-11, R-12, R-13, R-14, R-15, and R-16 are provided in Table 11.5-1. Portable check sources are used to check the detector and instrumentation circuitry.

11.5.3.1.7 Fuel Building Ventilation System (FBVS)

The FBVS is described in Section 9.4.2. The monitoring system consists of three measurement points, R-17, R-18, and R-19. R-17 (FB cell 4) and R-18 (FB cell 5) are shown in Figure 9.4.3-3. R-19 (FB fuel handling hall) is shown in Figure 9.4.2-1 and Figure 11.5-1.

The R-17 and R-18 measurement points continuous process monitors each consist of one noble gas, one iodine, and one aerosol monitors. These two measurement points grab sample provisions consist of one aerosol sample point at each measurement point. This monitoring system provides local and control room indication and alarms. The monitors for each of these measurement points initiate an automatic control feature that diverts the ventilation air flow to the NABVS iodine filter train on high activity. This is a non-safety function. The R-19 measurement point continuous process monitor consists of two noble gas monitors. This monitoring system provides local and control room indication and alarms. The monitors for this measurement point initiate an automatic control feature that isolates the fuel handling area ventilation on high activity in the exhaust.

Measurement ranges of the FBVS monitoring system are shown in Table 11.5-1. The safety classification for the R-17 and R-18 instruments is non-safety augmented quality. The safety classification for the R-19 instruments is non-safety, augmented quality. Portable check sources are used to check the detector and instrumentation circuitry.

11.5.3.1.8 Radioactive Waste Processing Building Ventilation System (RWPBVS)

The RWPBVS monitoring system consists of five measurement points, R-20, R-21, R-22, R-23, and R-24. R-20 (RWPB cell 2) and R-22 (RWPB cell 1) are shown in Figure 9.4.8-2. R-21 (room exhaust) is shown in Figure 9.4.8-2 and Figure 11.5-1. R-23 (decontamination room) and R-24 (mechanical workshop) are shown in Figure 9.4.8-1. See Section 9.4.8 for a complete discussion of the radioactive waste processing building ventilation.

The R-20 and R-22 measurement points continuous process monitors each consist of one iodine and one aerosol monitor upstream of the filters. These two measurement points grab sample provisions consist of one aerosol sample point at each measurement point. The R-21 measurement point grab sample provisions consist of 1 aerosol sample point downstream of the filters. The R-23 and R-24 measurement points grab sample provisions each consist of one aerosol sample point. This monitoring system provides local and control room indication and alarms. The monitors R-20 and R-22 initiate automatic actions as described in Table 11.5-1.

Measurement ranges of the RWPBVS monitoring system are shown in Table 11.5-1. The safety classification for the R-20 R-21, R-22, R-23, and R-24 instruments are provided in Table 11.5-1. Portable check sources are used to check the detector and instrumentation circuitry.

11.5.3.1.9 Safeguard Building Ventilation System (SBVS)

The SBVS, through its interconnections to the SBVSE and the nuclear auxiliary building ventilation system (NABVS), provides conditioned air for ventilation to the mechanical part of the SBs. The purpose of the SBVS monitoring system is to monitor and record the radioactivity levels of noble gases, iodines and aerosols in the SBVS exhaust and to divert air flow to the NABVS iodine filter train on high activity. The SBVS monitoring system consists of two measurement points, R-25 and R-26. R-25 (SB cell 6) is shown in Figure 9.4.3-3. R-26 (downstream of accident iodine filters) is shown in Figure 9.4.5-2.

The R-25 measurement point continuous process monitors consists of one noble gas, one iodine, and one aerosol monitors. This measurement point grab sample provisions consist of one aerosol sample point at the measurement point. The monitors for this measurement point initiate an automatic control feature that diverts the ventilation air flow to the NABVS iodine filter train on high activity. This is a non-safety function. The R-26 measurement point continuous effluent monitor consists of two noble gas monitors on the exhaust of the accident iodine filtration trains. This monitoring system provides local and control room indication and alarms. The monitors for this measurement point initiate an automatic control feature that diverts SBVS to the iodine filtration train.

Measurement ranges of the SBVS monitoring system are shown in Table 11.5-1. The safety classification for the R-25 and R-66 instruments is non-safety. The safety classification for the R-26 instruments is non-safety augmented quality. Portable check sources are used to check the detector and instrumentation circuitry of measurement points R-25 and R-26.

11.5.3.1.10 Annulus Ventilation System (AVS)

The AVS monitoring system consists of two measurement points, R-27 and R-28. Both of these points are on the exhaust of the iodine filtration trains from the annulus as shown in Figure 6.2.3-2.

The R-27 measurement point continuous effluent monitors consist of two noble gas monitors that are designed for post accident operation. The monitors for this measurement point do not initiate automatic actions. The R-28 measurement point grab sample consists of an iodine and aerosol sampling on the exhaust of the iodine filtration trains that are designed for post accident operation.

Measurement ranges of the AVS monitoring system are shown in Table 11.5-1. The safety classification for the R-27 instruments is non-safety augmented quality. The safety classification for the R-28 instruments is non-safety. Built-in check sources are used to check the detector and instrumentation circuitry of measurement point R-27.

11.5.3.1.11 Control Room Air Conditioning System (CRACS)

The main control room air conditioning system (CRACS) is designed to maintain a controlled environment in the control room envelope (CRE) area for the comfort and safety of control room personnel and to support operability of the control room components during normal operation, anticipated operational occurrences and design basis accidents. See Section 9.4.1 for a complete discussion. The purpose of the CRACS monitoring system is to monitor the air intakes for incoming radioactivity during accident conditions and then when the activity reaches the set-point, to isolate the MCR ventilation from the outside atmosphere, and initiate emergency habitability and supplemental filtration. The CRACS monitoring system consists of two measurement points, R-29 and R-30. R-29 and R-30 are shown in Figure 9.4.1-1.

The R-29 and R-30 measurement points continuous process monitors each consist of two radiation monitors. This monitoring system provides control room indication and alarms. The monitors for each of these measurement points initiate an automatic control feature that isolates MCR ventilation from the outside atmosphere and initiates emergency habitability and supplemental filtration. See Section 7.3.1.2.16 for details regarding the actuation of this automatic control feature.

Measurement ranges of the CRACS monitoring system are shown in Table 11.5-1. The safety classification for the R-29 and R-30 instruments is safety. Built-in check sources are used to check the detector and instrumentation circuitry.

11.5.3.1.12 Access Building Ventilation System (ABVS)

The access building ventilation system (ABVS) maintains room ambient conditions inside the Access Building to permit personnel access to the Nuclear Island (NI), and to control the concentration of airborne radioactive material in the controlled areas of the building during normal operation, including maintenance and refueling shutdowns, and during anticipated operational occurrences. The purpose of the ABVS monitoring system is to measure and record the radioactivity level in the exhaust of the ABVS system (see Section 9.4.14). The ABVS monitoring system consists of one measurement point, R-31 which is shown in Figure 9.4.14-2. The R-31 measurement point effluent grab sample provisions consist of one iodine and one aerosol sample point at the measurement point. The safety classification for the R-31 instruments is non-safety.

11.5.3.2 Liquid Effluents

The liquid effluent radioactive waste monitoring and sampling system measures the concentration of radioactive materials in liquids released to the environment. Liquid radionuclide concentration levels are designed to comply with 10 CFR Part 20 and dose requirements specified in 10 CFR Part 50. Liquid radioactive waste is discharged in batches from waste monitoring tanks. Prior to release of a liquid radioactive waste from a monitoring tank, the system obtains a representative sample and the sample is radiochemically analyzed. The ODCM (see Section 11.5.2) includes the following information for each location subject to routine liquid effluent sampling: (1) the sampling frequency, and (2) the analytical process and sensitivity for selected radioanalytical methods and types of sampling media. Results of this analysis are used in conjunction with dilution factor data to determine a release setpoint for the liquid waste monitoring system. Two continuously operating radiation sensors monitor the release line from the monitoring tanks. If a set limit is exceeded or if the monitoring system is inoperable, the release is automatically terminated. To terminate a release, one of the radiation sensors closes both isolation valves.

The liquid effluent radioactive waste monitoring system functional location is shown on Figure 11.2-1. Measurement ranges of the liquid radioactive waste monitoring system are shown in Table 11.5-1. The ODCM contains the plant's standard radiological effluent controls. This includes a description of how liquid effluent release rates are derived and parameters used in setting instrumentation alarm setpoints to control or terminate effluent releases in unrestricted areas that are above the effluent concentrations in Table 2 of Appendix B to 10 CFR Part 20. In addition,

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the ODCM describes how the guidance of NUREG-1301 (Reference 8) and NUREG-0133 (Reference 9) were used in developing the bases of alarm setpoints.

The liquid radwaste monitoring system consists of one measurement point, R-32 which is shown in Figure 11.2-1 and Figure 11.5-1. The R-32 measurement point continuous effluent monitors consists of two liquid monitors on the liquid radwaste release line. This monitoring system provides local and control room indication and alarms. The monitors for this measurement point initiate an automatic control feature to close two downstream isolation valves, close the upstream isolation valve, and shut down the operating recirculation and discharge pump(s). The safety classification for the R-32 instruments is non-safety augmented quality.

11.5.4 Process Monitoring and Sampling

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Process radiation monitoring detects, at an early stage, the escape of radioactive materials from radioactivity-containing systems into systems that are normally free of activity. Process radiation monitors generally operate continuously and provide both local and control room indication and alarm. Certain systems automatically initiate isolation actions along with control room alarm upon the detection of high radiation levels. Information regarding subsystem checks, tests, and maintenance may be found in Section 7.1.1.5.5.

11.5.4.1 Main Steam Radiation Monitoring System

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Radioactivity releases from the reactor coolant system (RCS) to the main steam system ~~(nitrogen-16, noble gases)~~ can occur because of steam generator tube leakage. Radioactivity (nitrogen-16, noble gases) in the main steam system is monitored over a wide power range by four redundant detectors~~measuring arrangements~~ per main steam line, for a total of 16 detectors for the system. These detectors are located at measuring points R-55, R-56, R-57, and R-58 as shown on Figure 10.3-1. The gamma sensitive detectors are mounted adjacent to the main steam lines within the main steam and feedwater valve compartments. ~~At low power levels, radioactivity is detected in the main steam due to the presence of noble gas. At high power levels, radiation is detected from the strong gamma from nitrogen-16. Shielding of detectors helps to prevent false readings from the detectors on the other main steam lines. The redundant measurement signals are processed, and provide alarm in the control room upon detection of radioactivity.~~The detectors are placed within specially designed lead shields that limit the angle of view to the steam line being monitored. Such an arrangement minimizes the contribution of scatter radiation as well as direct radiation emanating from the adjacent steam lines. The redundant measurement signals are processed, and provide alarm in the control room upon detection of radioactivity.

At both low and high power levels, the detected radiation emanating from the main steam lines is primarily due to the presence of N-16 (at a concentration of 4.5E-06 µCi/

cc for full-power operation and the Technical Specification leakage rate of 150 gallons per day). The photon radiation from noble-gas decay is predominantly of low energy in comparison to the high-energy radiation emitted by N-16 and, as a result, undergoes significant attenuation in traversing the 1.9 inch steel wall of the main steam lines. Specifically, for the same N-16 and noble-gas concentration at the radiation monitor location, N-16 yields a radiation field at the detector which is about 50 times higher than that generated by the realistic noble-gas mix. The noble-gas concentration at the radiation monitor location is expected to range between 1.1E-06 μCi/cc at 10 percent power to 8.1E-07 μCi/cc at full power, based on the RCS realistic noble gas concentration of 2.0 μCi/gm (from Table 11.1-7—RCS and Secondary Coolant System Realistic Source Terms) and the Technical Specification limit of 150 gallons per day for primary-to-secondary leakage. The drop in the steam noble gas concentration with power level is due to the non-direct proportionality between steam volumetric flow and power level for the U.S. EPR design. Specifically, a 10-fold increase in the power level (from 10 to 100 percent) corresponds with a steam volumetric flow increase by a factor of 13.6.

Noble-gas radioactivity within the steam lines is relied upon for mitigation of a steam generator tube rupture accident (SGTR), after plant cooldown has already been initiated and the N-16 radiation field no longer exists. Following such an event, the noble-gas concentration at the MSL detectors of the affected steam generator would be sufficiently high to generate the needed post-accident signal for identification and automatic isolation of the affected SG.

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The main steam radiation monitoring system is also used in conjunction with the condenser air removal and steam generator blowdown radiation monitoring systems to identify a steam generator tube leak. The main steam radiation monitoring system provides alarms and signals to the protection system for automatic isolation of an affected steam generator. This monitoring system provides control room indication and alarms. However, this is not the only required signal for the automatic control feature to actuate. With an actuated partial cooldown signal, either a high steam generator water level or high main steam activity will actuate the signal to isolate the affected steam generator. See Section 7.3.1.2.14 for additional information. ~~Measurement ranges of the main steam radiation monitoring system are shown in Table 11.5-1—Radiation Monitor Detector Parameters.~~

Measurement ranges of the main steam radiation monitoring system are shown in Table 11.5-1. The safety classification for the R-55, R-56, R-57, and R-58 instruments is safety. Built-in check sources are used to check the detector and instrumentation circuitry of measurement points R-55, R-56, R-57, and R-58.

Quantification of the primary-to-secondary leakage rate by the MSL radiation monitors is based on correlations which account for the following:

Monitor Instrumentation

To avoid spurious trip and failure, the four detectors on each steam line are linked in a two-out-of-four logic circuit.

11.5.4.2 ~~Condenser Air Removal~~Main Condenser Evacuation Radiation Monitoring System

Noncondensable gases (air and noble gases) in the secondary steam system are continuously removed during operation by the main condenser evacuation system (see Section 10.4.2). These gases discharge to the vent stack via a vent line. To monitor noble gas radioactivity, the monitoring system extracts part of the flow from the vent line (see Figure 10.4.2-2) and passes it through a measuring vessel with a beta-sensitive detector. If the monitoring system detects noble gas radioactivity in the secondary steam system, then it provides local and control room alarm. This alarm is an indication of breach of fuel cladding, primary coolant boundary, or containment leak. Table 11.5-1, Monitor R-3 shows the measurement ranges of the condenser air removal radiation monitoring system. This system does not initiate automatic actions.

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11.5.4.3 Steam Generator Blowdown Radiation Monitoring System

The evaporation process within the steam generator results in the concentration of contaminants in the liquid phase. These contaminants include non-gaseous radioactive substances that have entered the secondary system from the RCS due to tube leakage in a steam generator.

Sampling lines extract blowdown water from the individual blowdown lines for chemical analysis. These lines are located ahead of the primary isolation valve within the reactor containment. Flow is continuously extracted from each of these lines and fed to gamma activity measurement equipment. This configuration allows each steam generator to be monitored separately and continuously for radioactivity carryover to the secondary side.

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The steam generator blowdown radiation monitoring system provides an additional qualitative means to identify and verify a steam generator with a tube leak. These monitors may provide an indication from trending of a steam generator tube leak when the activity level is too low at the main steam line radiation monitors to develop a response from N-16 or noble gases.

The steam generator blowdown monitoring system consists of four measurement points, one on each blowdown line, R-46, R-47, R-48, and R-49. The R-46, R-47, R-48, and R-49 measurement points grab sample provisions consist of one liquid sample point at each measurement point. This monitoring system provides control room indication and alarms. This system does not solely initiate automatic actions. However, when a partial cooldown signal exists, a high activity signal from a steam generator blowdown radiation monitor will initiate an automatic control feature to isolate steam generator blowdown in the affected steam generator.

The steam generator blowdown radiation monitoring system functional location is shown on Figure 11.5-1. Measurement ranges of the steam generator radiation monitoring system are shown in Table 11.5-1. The safety classification for the R-46, R-47, R-48, and R-49 instruments is non-safety. Portable check sources are used to check the detector and instrumentation circuitry.

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~~The steam generator blowdown radiation monitoring system provides a second means to identify and verify a steam generator with a tube leak. These monitors detect a steam generator tube leak if there is no response from the main steam line radiation monitors because the activity level is too low. Together with manually extracted samples analyzed in the laboratory, this system also allows the calculation of radioactivity discharged from the secondary system due to leakage into the Turbine Building atmosphere. This system does not initiate automatic actions.~~

~~The steam generator blowdown radiation monitoring system functional location is shown on Figure 11.5-1. Measurement ranges of the steam generator radiation monitoring system are shown in Table 11.5-1.~~

11.5.4.4 Component Cooling Water Radiation Monitoring System

The component cooling water system consists of a closed-loop system of coolers (heat exchangers) used to transfer heat from nuclear components to essential service water. The system consists of many subsystems, two of which are the general component cooling water radiation monitoring subsystem which cools the residual heat removal system and the high-pressure (HP) cooling water radiation monitoring subsystem. The HP cooling water radiation monitoring subsystem is described in Section 11.5.4.17. The general component cooling water radiation monitoring system functional location (including subsystems) is shown in Figure 11.5-1.

The general component cooling water radiation monitoring subsystem uses gamma-sensitive radiation detectors in each of its four separate safety-related trains to monitor the fluid for the escape of radioactivity from the various radioactivity-containing systems that make up the nuclear components served by the component cooling circuits. The general component cooling water radiation monitoring subsystem consists of four measurement points, R-35, R-36, R-37, and R-38, one for each train. The gamma-sensitive detectors are lead-shielded and are installed adjacent to the piping in this subsystem. This subsystem provides local and control room indication alarms in the event that component cooling water gamma radiation levels exceed the monitor setpoint. The monitors for this measurement point initiate an automatic control feature that isolates the CCWS train on high activity. The R-35, R-36, R-37, and R-38 measurement points grab sample monitor consists of one liquid grab sample point in each train.

Measurement ranges of the component cooling water radiation monitoring system are shown in Table 11.5-1. The safety classification for the R-35, R-36, R-37, and R-38 instruments is non-safety, augmented quality. Portable check sources are used to check the detector and instrumentation circuitry of measurement points R-35, R-36, R-37, and R-38.~~The component cooling water system consists of a closed-loop system of coolers (heat exchangers) used to transfer heat from nuclear components to service water. Since it is a closed-loop system, the component cooling water system does not release radioactivity to the service water (and subsequently to the environment) in the event of leaks in the associated coolers. The system consists of two subsystems: the general component cooling water radiation monitoring subsystem and the high-pressure (HP) cooling water radiation monitoring subsystem. The component cooling water radiation monitoring system functional location (including subsystems) is shown in Figure 11.5-1. Measurement ranges of the component cooling water radiation monitoring system are shown in Table 11.5-1.~~

~~The general component cooling water radiation monitoring subsystem uses gamma-sensitive radiation detectors in each of its four separate safety-related trains to monitor the fluid for the escape of radioactivity from the various radioactivity-containing systems that make up the nuclear components served by the component cooling circuits. The gamma-sensitive detectors are lead-shielded and are installed adjacent to the piping in this subsystem. This subsystem provides local and control room alarms in the event that component cooling water gamma radiation levels exceed the monitor setpoint, but does not initiate automatic actions.~~

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~~The HP cooling water radiation monitoring subsystem consists of two gamma-sensitive radiation detectors upstream and two gamma-sensitive radiation detectors downstream on the component cooling water lines feeding/exiting the two high-pressure coolers of the volume control system. In the event of a leak in an HP cooler, in which high-activity primary coolant leaks into the component cooling water system, the radiation detector downstream of the defective cooler indicates the entry of radioactivity from this HP cooler into the component cooling loop that is running at the time. If the radioactivity exceeds a pre-determined limit, the defective HP cooler is automatically isolated on the primary side and an associated control room alarm is activated. This automatic action is suppressed if the limit value of the radiation detector at the inlet of the cooler has already triggered a high activity signal. This action is also suppressed during inservice inspection of the measuring points.~~

11.5.4.5 Gaseous Waste Disposal Radiation Monitoring System

Gamma-sensitive radiation detectors continuously monitor the gaseous waste delay beds. A gamma-sensitive radiation detector is located upstream of the beds (R-1) and a

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beta-sensitive radiation detector is located downstream (R-2). The upstream monitor measures the gamma radiation emitted by the radionuclides entering the beds, and provides station personnel with an indication of the amount of radioactivity entering the beds. The downstream monitor is a beta-sensitive instrument, since krypton-85 generally forms the main constituent (about 95 percent) of the radioactive noble gases. The gaseous waste disposal radiation monitoring system provides control room and local indication only. This system does not initiates automatic actions to close discharge valves on high activity.

The radiation monitors in the gaseous waste processing system are shown on Figures 11.3-1 and 11.3-2. In addition, Section 9.4 includes simplified diagrams of the ventilation systems that show the radiation monitors. Measurement ranges of the gaseous waste disposal radiation monitoring system are shown in Table 11.5-1.

11.5.4.6 Reactor Coolant Radiation Monitoring andNuclear Sampling Sampling System

The nuclear sampling and severe accident sampling monitoring system consists of one measurement point, R-41. R-41 measures the noble gas activity concentration in the gaseous volume flow subsequent to the degasifier for primary coolant of the Nuclear Sampling System as shown in Figure 9.3.2-1 and Figure 11.5-1.

The R-41 measurement point continuous process monitor consists of one noble gas monitor. This measurement point grab sample provisions consist of one liquid sample point at the measurement point. This monitoring system provides local and control room indication and alarms. The monitors for this measurement point do not initiate automatic actions.

Measurement ranges of the nuclear sampling and severe accident sampling monitoring system are shown in Table 11.5-1. The safety classification for the R-41 instrument is non-safety, augmented quality. Portable check sources are used to check the detector and instrumentation circuitry of measurement points R-41.

The severe accident sampling system is a separate system described in Section 9.3.2.2.1.3.

The reactor coolant radiation monitoring system uses a beta-sensitive radiation detector to provide continuous monitoring of the integral noble gas activity concentration of the coolant. The system monitors the noble gas activity concentration in the gaseous flow to the de-gasifier for primary coolant of the nuclear sampling system and allows in the early detection of fuel element failures. The measuring range of the system allows monitoring of noble gas activity concentrations in the RCS coolant up to the TS value. The reactor coolant radiation monitoring system provides indication only and does not initiate action. A diagram of the reactor coolant radiation monitoring and sampling system is shown in Figure 11.5-3.

~~Measurement ranges of the reactor coolant radiation monitoring system are shown in Table 11.5-1.~~

~~The reactor coolant radiation and sampling system provides representative samples of reactor coolant to the radiochemistry laboratory for analysis. Results of these analyses are used to demonstrate compliance with reactor coolant chemistry limits. These analyses are also used to provide information to detect failed fuel.~~

11.5.4.7 Chilled Water Supply for the Gaseous Waste Disposal Sampling System

The closed-loop chilled water system serves various components in the gaseous waste disposal system and the coolant degasification system. The radioactive sides of these operational components are separated from the chilled water system by means of both material and pressure barriers. The higher pressure on the non-contaminated side prevents radioactivity from escaping to the chilled water system except in the event of coincident failure of both of these barriers. The sampling point for extracting samples from this system for radiochemistry laboratory evaluation is provided in the return manifold of the chilled water system.

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The chilled water supply monitoring system consists of one measurement point, R-61. The R-61 measurement point grab sample provisions consists one liquid sample point at the measurement point. The safety classification for the R-61 sample point is non-safety.

11.5.4.8 Radiation Monitoring System for RCS Leakage Detection

Containment atmosphere particulate radioactivity monitoring is one of the systems used in the U.S. EPR design for RCS leakage detection described in Section 5.2.5. The particulate radiation monitoring system continuously monitors airborne radioactivity in the containment equipment area. Radiation levels are indicated in the MCR. Alarms alert the operators of elevated levels of radioactivity to allow for prompt identification of RCS leakage into the equipment area. The monitor is located in the service area of the containment, which is accessible during normal operation. The system draws air from the containment building ventilation system which filters airborne radioactivity within the equipment area. The sampled flow is returned to the equipment area.

The particulate monitor is a low-range monitor capable of detecting 3E-10 to 1E-6 μCi/cc (Radiation Monitoring Point R-10 in Table 11.5-1). The monitor sensitivity requirement is to be able to detect a leakage increase of one gpm within one hour based on a realistic RCS source term (Table 11.1-7) consistent with RG 1.45 and RIS-2009-02 (Reference 11). Typical radionuclides of interest are as follows:

1-member decay chains:	Na-24	Y-93
	Te-129	

- Placement of Radiation Monitoring Instrumentation

  The radiation level in the area housing the monitoring instrumentation is less than 25 mrem/hr (refer to Figure 12.3-13—Reactor Building Cross-Section Radiation Zones). Shielding is used to minimize the interferences from ambient external radiation levels, including shine from the potential accumulation of radioactivity on the KLA-5 filtration system.

RCS leakage quantification can also be based on the detection of activation products, such as F-18, which is generated by the O^{18} (proton, neutron) F^{18} reaction as the RCS coolant passes through the core. This particular isotope is of particular interest for RCS leakage detection because it is not a fission product (and, therefore, not dependent on fuel clad defects), has a reasonable half life (about 110 min), decays by positron emission (followed by annihilation radiation, which is readily detectable), and combines with lithium in the water to form LiF (a particulate). It is not addressed in the U.S. EPR design for RCS leakage detection due to the unavailability of analytical values for its equilibrium concentration within the RCS as LiF, and the loss mechanisms it is subjected to when released to the containment atmosphere as a result of leakage.

11.5.4.9 Essential Service Water System

The essential service water system (ESWS) is designed to remove heat from plant components. This is accomplished by providing cooling water from the essential service water (ESW) cooling tower basins to the component cooling water system (CCWS) heat exchangers (HX), emergency diesel generator (EDG) HXs, and ESW pump room coolers. The interface between the ESWS and CCWS is monitored for cross-contamination by monitoring points R-66 through R-70.

11.5.4.10 Fuel Pool Purification System

The fuel pool purification monitoring system consists of one measurement point, R-39. The R-39 measurement point grab sample provisions consists of one liquid sample point at the measurement point. The safety classification for the R-39 sample point is non-safety.

11.5.4.11 Nuclear Island Vent and Drain System

The Nuclear Island vent and drain monitoring system consists of two measurement points, R-40 and R-44. The R-40 and R-44 measurement points grab sample provisions consist of one liquid sample point at each measurement point. The safety classification for the R-40 and R-44 sample points is non-safety.

11.5.4.12 Laundry Handling Room and Decontamination System

The laundry and decontamination monitoring system consists of one measurement point, R-42. R-42 measures the radioactive contamination on clothing. Although an onsite laundry facility is not in the U.S. EPR design, the monitor may be used to spot check laundry from an offsite facility to verify the cleanliness of the protective clothing. This monitor is also suitable to measure the contamination of the clothes before washing to divide the clothes into very high, high, and low contamination categories for selection of individual washing programs at the offsite facility.

The R-42 measurement point process monitor consists of one protective clothing monitor. This measurement point grab sample provision consists of one liquid sample point of the decontamination waste at the measurement point. The monitors for this measurement point do not initiate automatic actions.

The safety classification for the R-42 instrument is non-safety. Portable check sources are used to check the detector and instrumentation circuitry of measurement points R-42.

11.5.4.13 Solid Radwaste System

The radiation monitoring of waste packages will be performed in the drum measuring station. The solid radwaste monitoring equipment, R-43, is comprised of seven dose rate monitors, one gamma spectroscopy system with a multi-channel analyzer, and drum identification, weighing, rotation, registration and archiving-equipment. R-43 is shown in Figure 11.4-1. The dose rate is measured at several positions while the drum is turning. The background dose rate level is measured at a representative position close to the drum measuring equipment. Simultaneously, the activity of gamma emitting nuclides will be detected and measured. A liquid grab sample may be drawn from the drum and analyzed in the laboratory.

Measurement ranges of the solid radwaste monitoring system are shown in Table 11.5-1. The safety classification for the R-43 instruments is non-safety. Portable check sources are used to check the detector and instrumentation circuitry.

11.5.4.14 Reactor Boron and Water Makeup System

The reactor boron and water makeup monitoring system consists of one measurement point, R-45. The R-45 measurement point grab sample provision consists of one liquid sample point at the measurement point. The safety classification for the R-45 sample point is non-safety.

11.5.4.15 Turbine Building Drains and Vents System

The Turbine Building drains and vents monitoring system consists of one measurement point, R-50 on the common release line as shown in Figure 11.5-1. The task for this measurement point is to monitor the liquid effluent from the Plant Drainage System before it is discharged to the retention pond.

The R-50 measurement point continuous effluent monitor consists of one liquid monitor. This measurement point effluent grab sample provisions consist of one liquid sample point at the measurement point. This monitoring system provides local and control room indication and alarms. The monitor for this measurement point does not initiate automatic actions.

The safety classification for the R-50 instrument is non-safety. Portable check sources are used to check the detector and instrumentation circuitry of measurement points R-50.

11.5.4.16 Clean Drains System

The clean drains monitoring system consists of one sampling point, R-65. The clean drains are collected and routed to the main condenser. The safety classification for the R-65 sample point is non-safety.

11.5.4.17 Chemical and Volume Control System (CVCS) High Pressure Coolers

The component cooling water system consists of a closed-loop system of coolers (heat exchangers) used to transfer heat from nuclear components to essential service water. The system consists of many subsystems, two of which are the general component cooling water radiation monitoring subsystem which cools the residual heat removal system and the high-pressure (HP) cooling water radiation monitoring subsystem. The general component cooling water radiation monitoring subsystem is described in Section 11.5.4.4.

The HP cooling water radiation monitoring subsystem consists of five measurement points, R-51, R-52, R-53, R-54, and R-64. The R-51 and R-52 monitoring points are on the CCWS inlet and outlet of CVCS HP cooler 1 as shown on Figure 9.2.2-2. The R-53 and R-54 monitoring points are on the CCWS inlet and outlet of CVCS HP cooler 2 as shown on Figure 9.2.2-3. The R-64 monitoring point is downstream of the severe accident heat removal system heat exchangers. The gamma-sensitive detectors are lead-shielded and are installed adjacent to the piping in this subsystem. This subsystem provides local and control room alarms in the event that component cooling water gamma radiation levels exceed the monitor setpoint. The monitors for this measurement point initiate an automatic control feature that isolates the CCWS train on high activity.

Measurement ranges of the component cooling water radiation monitoring system are shown in Table 11.5-1. The safety classification for the R-51, R-52, R-53, and R-54 instruments is non-safety, augmented quality. The safety classification for the R-64 instrument is non-safety. Built-in check sources are used to check the detector and instrumentation circuitry of measurement points R-51, R-52, R-53, R-54, and R-64.

11.5.4.18 Safety Chilled Water System

The safety chilled water system (SCWS) supplies refrigerated chilled water to the safety-related heating, ventilation and air conditioning (HVAC) systems and the low head safety injection system (LHSI) pumps and motors in Safeguard Buildings (SB) 1 and 4 and the fuel building ventilation system (FBVS). See Section 9.2.8 for a complete discussion. The purpose for radiation monitoring is to detect leakage of radioactive fluid from the reactor coolant side of the LHSI heat exchanger to the safety chilled water side and to alert the operator that a leak exists.

The safety chilled water monitoring system consists of two measurement points, R-59 and R-60 downstream of the LHSI sealing fluid cooler as shown in Figure 9.2.8-1 and Figure 11.5-1. The task for this measurement point is to monitor the residual heat removal heat exchanger for leakage.

The R-59 and R-60 measurement point continuous process monitor consists of one liquid monitor. This measurement point grab sample provision consists of one liquid sample point at the measurement point. This monitoring system provides local and control room indication and alarms. The monitor for this measurement point does not initiate automatic actions.

The safety classification for the R-59 and R-60 instruments is non-safety. Portable check sources are used to check the detector and instrumentation circuitry of measurement points R-59 and R-60.

11.5.5 References

1. ANSI/HPS N13.1-1999, “Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities,” American National Standards Institute/Health Physics Society, 1999.

2. ANSI N42.18-2004, "Specifications and Performance of On-site Instrumentation for Continuously Monitoring Radioactivity in Effluents," American National Standards Institute, 2004.

3. NUREG-0800, BTP 7-10, “Guidance on Application of Regulatory Guide 1.97,” Revision 5, U.S. Nuclear Regulatory Commission, March 2007.

4. NUREG-0737, “Clarification of TMI Action Plan Requirements,” U.S. Nuclear Regulatory Commission, November 1980.

5. NUREG-0718, “Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses,” U.S. Nuclear Regulatory Commission, March 1981.

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Table 11.5-1—Radiation Monitor Detector Parameters[1]

Process System[2]	Monitor Provisions			Sample Provisions			Range
	In-Process Continuous	ACF	In-Effluent Continuous	In-Process Grab sample	In-Effluent Grab sample	Continuous	
Gaseous Streams							
Gaseous Waste Processing System (Waste Gas Holdup System)	1 noble gas monitor upstream of the delay beds	—	—	—	—	—	1E+0 - 1E+4 cps (Kr-85, Xe-133)
	—	—	1 noble gas monitor downstream of the delay beds	—	—	—	1E-06 - 1E+2 µCi/cc (Kr-85, Xe-133)
Condenser Evacuation System	1 noble gas monitor on the condenser exhaust	—	—	—	—	—	3E-6 - 1E-2 µCi/cc (Kr-85, Xe-133)
Sampling Activity Monitoring System (Vent & Stack Release Point System)	—	—	5 noble gas monitors in the stack	—	3 noble gas sample points in the stack	3 noble gas sample points in the stack	3E-6 - 1E-2 µCi/cc (Kr-85, Xe-133)
	—	—	2 iodine monitors, 2 aerosol monitors in the stack	—	4 iodine and 4 aerosol, and 2 H-3, C14 sample points in the stack	4 iodine, 4 aerosol and 2 H-3, C-14 sample points in the stack	—

Process System[2]	Monitor Provisions			Sample Provisions			Range
	In-Process Continuous	ACF	In-Effluent Continuous	In-Process Grab sample	In-Effluent Grab sample	Continuous	
Containment Building Ventilation System —Containment Purge Subsystem (Containment Purge System)	1* noble gas, 1* aerosol, 1* iodine, and 1* H-3 monitor		2* noble gas monitors	1 aerosol sample point		1 aerosol sample point	3E-7 – 1E-2 µCi/cc (Kr-85, Xe-133)
Containment Building Ventilation System —Internal Filtration Subsystem	1* noble gas, 1* aerosol, 1* iodine monitor			1 aerosol sampler		1 aerosol sampler	3E-7 – 1E-2 µCi/cc (Kr-85, Xe-133)
Nuclear Auxiliary Building Ventilation System	3* noble gas, 3* iodine, and 5* aerosol monitors on ventilation exhaust	—	—	5 aerosol sample points in the ventilation exhaust	1 iodine and 1 aerosol sample points in the ventilation exhaust	1 iodine and 6 aerosol sample points in the ventilation exhaust	3E-7 – 1E-2 µCi/cc (Kr-85, Xe-133)
Fuel Building Ventilation System (Fuel Storage Area Ventilation System)	4* noble gas, 2* aerosol, and 2* iodine monitors on ventilation exhaust	isolate fuel handling area ventilation on high exhaust activity, 1 noble gas monitor supplies the signal	—	2 aerosol sample points in the ventilation exhaust	—	2 aerosol sample points in the ventilation exhaust	3E-7 – 1E-2 µCi/cc (Kr-85, Xe-133)

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Process System[2]	Monitor Provisions			Sample Provisions			Range
	In-Process Continuous	ACF	In-Effluent Continuous	In-Process Grab sample	In-Effluent Grab sample	Continuous	
Radioactive Waste Processing Building Ventilation System (Radwaste Area Vent Systems)	2* iodine and 4* aerosol monitors upstream of ventilation filters	—	—	4 aerosol sample points in the ventilation exhaust	1 aerosol and 1 iodine sample points in the ventilation exhaust	5 aerosol and 1 iodine sample points in the ventilation exhaust	
Turbine Gland Seal Condenser Vent System	The design provides monitors for the main steam system presented below. These instruments provide sufficient monitoring of the steam system to warn the operator of primary to secondary leakage and measure the effluent as would similar monitors on the turbine gland seal condenser vent system.						
Mech. Vacuum Pump Exhaust (Hogging System)	This system is used only during plant startup when an accident that could release gaseous radioactive material that could reach this system is unlikely and is therefore not monitored.						
Evaporator Vent Systems	The boron recovery evaporator in the coolant treatment system vents to the ventilation cell in which it is located in the Nuclear Auxiliary Building. The exhaust from the cell is monitored as shown above in the auxiliary building ventilation system. The evaporator in the Radwaste Building is monitored by the radwaste building ventilation system, which is described above.						
Liquid Waste Processing System (Pretreatment Liquid Radwaste Tank Vent Gas Systems)	All liquid radwaste processes are monitored by the radwaste building ventilation, which is described above.						
Steam Generator Blowdown System (Flash Tank and Steam Generator Blowdown Vent Systems)	There are no vents to atmosphere in the steam generator blowdown system. The steam generator blowdown flash tank is vented to the feedwater tank where the vented vapor is mixed with the incoming feedwater and is conveyed to the steam generator.						

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Process System²	Monitor Provisions			Sample Provisions			Range
	In-Process Continuous	ACF	In-Effluent Continuous	In-Process Grab sample	In-Effluent Grab sample	Continuous	
Pressurizer & Boron Recovery Vent Systems	The pressurizer is vented to the nuclear island drains and vents system, which routes gaseous radioactive wastes to the gaseous waste processing system. The coolant treatment system, which contains the boron recovery subsystem is vented to the gaseous waste processing system.						
Solid Waste System (Waste compactors, shredders, etc. (as permanently installed or mobile systems))	Solid waste components that could generate airborne wastes are monitored by the radwaste building ventilation system, which is described above.						
Safeguard Building Controlled Area Ventilation System	1* noble gas, 1* aerosol and 1* iodine monitors on ventilation exhaust	—	2* noble gas monitors on ventilation exhaust	1 aerosol sample points in the ventilation exhaust	1 iodine and 1 aerosol sample points in the ventilation exhaust	1 iodine and 2 aerosol sample points in the ventilation exhaust	3E-7 – 1E-2 μCi/cc (Kr-85, Xe-133)
Annulus Ventilation System	—	—	2* noble gas monitors in the ventilation exhaust	—	2 iodine sample points in the ventilation exhaust	2 iodine sample points in the ventilation exhaust	—
Control Room Air Conditioning System	4* radiation monitors	isolates MCR ventilation intakes and initiates emergency habitability and supplemental filtration	—	—	—	—	1E-5 – 1E+1 rad/hr

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Process System[2]	Monitor Provisions			Sample Provisions			Range
	In-Process Continuous	ACF	In-Effluent Continuous	In-Process Grab sample	In-Effluent Grab sample	Continuous	
Access Building Ventilation System	—	—	—	1 iodine and 1 aerosol in the ventilation exhaust	—	—	
Liquid Streams							
Liquid Radwaste (Batch) Effluent System	—	close the discharge valve on high activity	2 redundant monitors on the liquid radwaste release line	sample and analysis	sample and analysis, H-3 analysis	—	5E-6 – 1E-3 µCi/ml
Liquid Radwaste (Continuous) Effluent System	The U.S. EPR uses a batch liquid radwaste effluent discharge, which is monitored as described above.						
Service Water System and/or Circulating Water System	For U.S. EPR, the CWS has no contact with the TBPD except that the TBPD and the cooling tower blowdown each discharge separately to the "lined retention pond."						
Component Cooling Water System	1 radiation monitor on each loop	—	—	—	sample and analysis, H-3 analysis	—	1E-6 – 1E-3 µCi/ml
Fuel Pool Purification System (Spent Fuel Pool Treatment System)	—	—	—	—	sample and analysis, H-3 analysis	—	—

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Process System²	Monitor Provisions			Sample Provisions			Range
	In-Process Continuous	ACF	In-Effluent Continuous	In-Process Grab sample	In-Effluent Grab sample	Continuous	
Nuclear Island Drain and Vent Systems (Equipment & Floor Drain Collection and Treatment Systems)	—	—	—	—	sample and analysis, H-3 analysis	—	—
Phase Separator Decant & Holding Basin Systems	For the U.S. EPR, this is handled by the solid radwaste system.						
Chemical & Regeneration Solution Waste Systems	For the U.S. EPR, these waste streams are collected by the nuclear island drains and vents system, which is, routed to the liquid radwaste storage tanks.						
Nuclear Sampling and Severe Accident Sampling (Laboratory & Sample System Waste Systems)	—	—	—	—	sample and analysis, H-3 analysis	—	—
Laundry Room (Laundry & Decontamination Waste Systems)	1 protective clothing monitor (no laundry processing on site)	—	—	—	decontamination waste sample and analysis, H-3 analysis	—	—

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Process System[2]	Monitor Provisions			Sample Provisions			Range
	In Process Continuous	ACF	In Effluent Continuous	In Process Grab sample	In Effluent Grab sample	Continuous	
Solid Radwaste System (Resin Slurry, Solidification, & Baling Drain Systems)	—	—	—	—	sample and analysis, H-3 analysis	—	—
Radwaste Liquid Tanks (outside the buildings)	Not applicable to the U.S. EPR.						
Rainwater Collection and Drainage System (Storm & Underdrain Water System)	Potentially radioactive systems are segregated from non-radioactive systems, such as rainwater collection and drainage systems, to minimize the migration of radioactive material across systems. Monitoring is not required.						
Nuclear Island Drains and Vents System (Tanks and Sumps Inside Reactor Building)	—	—	—	—	sample and analysis, H-3 analysis	—	—
Reactor Boron and Makeup Water System (Boron Recovery System Liquid Effluent)	—	—	—	—	sample and analysis, H-3 analysis	—	—
Steam Generator Blowdown System (Steam Generator Blowdown (Batch) Liquid Effluent System)	1 radiation monitor on each steam generator blowdown line	diverts blowdown to radwaste	—	—	sample and analysis, H-3 analysis	—	3E-6–1E-2 µCi/ml

Process System[2]	Monitor Provisions			Sample Provisions			Range
	In-Process Continuous	ACF	In-Effluent Continuous	In-Process Grab sample	In-Effluent Grab sample	Continuous	
Steam Generator Blowdown System (Steam Generator Blowdown (Continuous) Liquid Effluent System)	1 radiation monitor on each steam generator blowdown line	diverts blowdown to radwaste	—	—	sample and analysis, H-3 analysis	—	3E-6–1E-2 µCi/ml
Turbine Drains and Vents System (Secondary Coolant Treatment Waste & Turbine Building Drain Systems)	—	—	—	—	sample and analysis, H-3 analysis	—	—
Clean Drains System (Noncontaminated Wastewater & PWR Turbine Building Clean Drain System)	—	—	1 radiation monitor on the common release line	—	sample and analysis, H-3 analysis	—	3E-6–1E-2 µCi/ml
Reverse Osmosis Systems	Not applicable to the U.S. EPR.						
Mobile Liquid and Wet Waste Processing Systems	Both the liquid waste storage and liquid waste processing systems are located entirely within the Radioactive Waste Processing Building. Interfacing system piping delivers influent liquid wastes from the adjacent Nuclear Auxiliary Building. The Radioactive Waste Processing Building is also sized to provide space and support services to optional, site-specific mobile or vendor supplied processing equipment. However, such optional mobile or vendor-supplied systems would be site-specific design features and are outside design certification scope.						

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Process System[2]	Monitor Provisions			Sample Provisions			Range
	In-Process Continuous	ACF	In-Effluent Continuous	In-Process Grab sample	In-Effluent Grab sample	Continuous	
Chemical and Volume Control System High Pressure Coolers	radiation monitors are on the component cooling inlet and outlet of each cooler	—	—	—	—	—	3E-5 - 3E+0 µCi/ml
Main Steam System	4 N-16 monitors on each main steam line	—	—	—	—	—	1E-1 - 1E+4 cps (N-16)

NOTES:

1. The instruments with an * asterisk are duplicated in Table 12.3-4.
2. Where a difference exists between the name of a system in the U.S. EPR and that listed in NUREG-0800, SRP 11.5, Tables 1 and 2 (Reference 7), the U.S. EPR system name is listed first, followed by the SRP name in parentheses.

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Table 11.5-1—Radiation Monitor Detector Parameters
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Process System and Initial Test Program[10]	Rad Meas Pt	Monitor Provisions[11]			Sample Provisions[12, 14]		noble gas H-3 or N-16 monitor range[13, 18]	aerosol monitor range[13, 18]	iodine monitor range[13, 18]	liquid monitor range[13, 18]	Figure	Text	safety grade	check source
		In-Process continuous	ACF	In-Effluent continuous	In-Process sample	In-Effluent sample								
Gaseous Waste Processing System (KPL) Test #144 Test #99 Test #153 Test #216	R-1	1 noble gas monitor upstream of the delay beds KPL40CR001	none	none	grab sample	none	3E-7 - 1E-2 µCi/cc (Kr-85, Xe-133)	n/a	n/a	n/a	11.3-1 11.3-2	11.3.1.2.3 11.5.3.1.1 11.5.4.5	NS-AQ	portable
	R-2	none	close the discharge valve on high activity	1 noble gas monitor downstream of the delay beds discharged into NABVS (cell 1) KPL83CR001	grab sample	none	3E-7 - 1E+2 µCi/cc (Kr-85, Xe-133)	n/a	n/a	n/a	11.3-1 11.3-2 11.5-1 9.4.3-2	11.3.1.2.3 11.3.2.3.12 11.5.3.1.1 11.5.4.5	NS-AQ	portable
Vent System for Air Removal (MAQ) Test #144 Test #204	R-3	none	none	1 noble gas monitor on the condenser exhaust MAQ90CR001	none	none	3E-7 - 1E-2 µCi/cc (Kr-85, Xe-133)	n/a	n/a	n/a	11.5-1 10.4.2-2	11.5.4 11.5.3.1.2	Non-safety	built-in

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Table 11.5-1—Radiation Monitor Detector Parameters
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Process System and Initial Test Program[10]	Rad Meas Pt	Monitor Provisions[11]			Sample Provisions[12, 14]		noble gas H-3 or N-16 monitor range [13, 18]	aerosol monitor range[13, 18]	iodine monitor range[13, 18]	liquid monitor range[13, 18]	Figure	Text	safety grade	check source
		In-Process continuous	ACF	In-Effluent continuous	In-Process sample	In-Effluent sample								
Sampling Activity Monitoring System (Vent Stack Release Point System (KLK)) Test #144 Test #92	R-4	none	none	1 noble gas monitor, 1 iodine monitor, 1 iodine accident monitor, 1 aerosol monitor and 1 aerosol accident monitor in the vent stack KLK70CR001 KLK70CR071 KLK70CR072 KLK70CR031 KLK70CR032	none	1 noble gas, 1 noble gas accident, 1 iodine, 1 iodine accident, 1 aerosol, 1 aerosol accident, and 1 H-3, C-14 grab sample points in the vent stack KLK70CR581 KLK70CR582 KLK70CR571 KLK70CR572 KLK70CR561 KLK70CR562 KLK70CR551	3E-7 - 1E+4 µCi/cc (Kr-85, Xe-133)	1E-8- 1E+2 µCi/cc (Cs-137)	5E-11 - 3E-7 µCi/cc (I-131)	n/a	11.3-2 11.5-1 9.4.3-3	11.5.3.1.3	Non-safety	built-in
	R-5	none	none	2 noble gas monitors in the vent stack KLK90CR001 KLK90CR002	none	1 noble gas accident, 1 iodine, 1 iodine accident, 1 aerosol, 1 aerosol accident, and 1 H-3, C-14 grab sample points in the vent stack KLK90CR582 KLK90CR571 KLK90CR572 KLK90CR561 KLK90CR562 KLK90CR551	3E-7 - 1E+4 µCi/cc (Kr-85, Xe-133)	n/a	n/a	n/a	11.3-2 11.5-1 9.4.3-3	11.5.3.1.3	Non-safety	built-in
	R-6	none	none	2 noble gas accident monitors in the vent stack KLK95CR001 KLK95CR002	none	none	1E-4 - 1E+4 rad/hr	n/a	n/a	n/a	11.3-2 11.5-1 9.4.3-3	11.5.3.1.3	Non-safety	built-in

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Table 11.5-1—Radiation Monitor Detector Parameters
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Process System and Initial Test Program[10]	Rad Meas Pt	Monitor Provisions[11]			Sample Provisions[12, 14]		noble gas H-3 or N-16 monitor range[13, 18]	aerosol monitor range[13, 18]	iodine monitor range[13, 18]	liquid monitor range[13, 18]	Figure	Text	safety grade	check source
		In-Process continuous	ACF	In-Effluent continuous	In-Process sample	In-Effluent sample								
Containment Building Low Flow Purge Subsystem (KLA2) Test #144 Test #76	R-7	none	none	1 noble gas, 1 aerosol, 1 iodine monitor KLK10CR001 KLK10CR031 KLK10CR071	none	1 aerosol sample point KLK10CR561	3E-7 - 1E-2 µCi/cc (Kr-85, Xe-133)	3E-10 - 1E-6 µCi/cc (Cs-137)	3E-10 - 5E-8 µCi/cc (I-131)	n/a	11.5-1 9.4.7-2	11.5.3.1.4	Non-safety	built-in
	R-8	none	none	1 H-3 monitor KLK12CR041	none	H-3 sample and analysis	3E-9 - 3E-4 µCi/cc	n/a	n/a	n/a	11.5-1 9.4.7-2	11.5.3.1.4	Non-safety	built-in
	R-9	none	Close KLA supply air damper to the Fuel Building hatch area, when the equipment hatch is open Close KLL exhaust air damper to the area in front of the emergency airlock Opens KLA2 iodine filtration isolation dampers, starts KLA2 iodine filtration unit fans	2 noble gas monitors KLK13CR001 KLK13CR002	none	none	1E-5 - 1E0 rad/hr	n/a	n/a	n/a	11.5-1 9.4.7-2	11.5.3.1.4	NS-AQ	built-in
Containment Building Internal Filtration Subsystem (KLA5) Test #144 Test #75 Test #187	R-10	1 noble gas, 1 aerosol[6], 1 iodine monitor KLK05CR001 KLK05CR031 KLK05CR071	none	none	1 aerosol grab sample KLK05CR561	none	3E-7 - 1E-2 µCi/cc (Kr-85, Xe-133)	3E-10 - 1E-6 µCi/cc (Te-129, Ru-106/ Rh-106)	3E-10 - 5E-8 µCi/cc (I-131)	n/a	11.5-1 9.4.7-3	11.5.3.1.5 11.5.4.8	NS-AQ	built-in

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Table 11.5-1—Radiation Monitor Detector Parameters
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Process System and Initial Test Program[10]	Rad Meas Pt	Monitor Provisions[11]			Sample Provisions[12, 14]		noble gas H-3 or N-16 monitor range[13, 18]	aerosol monitor range[13, 18]	iodine monitor range[13, 18]	liquid monitor range[13, 18]	Figure	Text	safety grade	check source
		In-Process continuous	ACF	In-Effluent continuous	In-Process sample	In-Effluent sample								
Nuclear Auxiliary Building Ventilation System (KLE) Test #144 Test #79	R-11	1 noble gas, 1 iodine, and 1 aerosol monitors on NABVS (cell 1) ventilation exhaust KLK30CR001 KLK30CR071 KLK30CR031	NABVS (cell 1) diverts exhaust to iodine filtration on high activity[1]	none	1 aerosol grab sample point in the ventilation exhaust KLK30CR561	none	3E-7 - 1E-2 µCi/cc (Kr-85, Xe-133)	3E-10 - 1E-6 µCi/cc (Cs-137)	3E-10 - 5E-8 µCi/cc (I-131)	n/a	11.5-1 9.4.3-3	11.5.3.1.6 9.4.3.2.1	NS-AQ	portable
	R-12	1 noble gas, 1 iodine, and 1 aerosol monitors on NABVS (cell 2) ventilation exhaust KLK31CR001 KLK31CR071 KLK31CR031	NABVS (cell 2) diverts exhaust to iodine filtration on high activity[1]	none	1 aerosol grab sample point in the NABVS (cell 2) ventilation exhaust KLK31CR561	none	3E-7 - 1E-2 µCi/cc (Kr-85, Xe-133)	3E-10 - 1E-6 µCi/cc (Cs-137)	3E-10 - 5E-8 µCi/cc (I-131)	n/a	9.4.3-3	11.5.3.1.6 9.4.3.2.1	NS-AQ	portable
	R-13	1 noble gas, 1 iodine, and 1 aerosol monitors on NABVS (cell 3) ventilation exhaust KLK32CR001 KLK32CR071 KLK32CR031	NABVS (cell 3) diverts exhaust to iodine filtration on high activity[1]	none	1 aerosol grab sample point in the ventilation exhaust KLK32CR561	none	3E-7 - 1E-2 µCi/cc (Kr-85, Xe-133)	3E-10 - 1E-6 µCi/cc (Cs-137)	3E-10 - 5E-8 µCi/cc (I-131)	n/a	9.4.3-3	11.5.3.1.6 9.4.3.2.1	NS-AQ	portable
	R-14	1 aerosol monitor on ventilation exhaust in the hot workshop KLK33CR031	none	none	1 aerosol grab sample point in the ventilation exhaust in the hot workshop KLK33CR561	none	n/a	3E-10 - 1E-6 µCi/cc (Cs-137)	n/a	n/a	9.4.3-2	11.5.3.1.6	Non-safety	portable
	R-15	1 aerosol monitor on ventilation exhaust in laboratory room upstream of NABVS iodine filtration train KLK41CR031	none	none	1 aerosol grab sample point in the ventilation exhaust KLK41CR561	none	n/a	3E-10 - 1E-6 µCi/cc (Cs-137)	n/a	n/a	9.4.3-2	11.5.3.1.6	Non-safety	portable

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Table 11.5-1—Radiation Monitor Detector Parameters
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Process System and Initial Test Program[10]	Rad Meas Pt	Monitor Provisions[11]			Sample Provisions[12, 14]		noble gas H-3 or N-16 monitor range[13, 18]	aerosol monitor range[13, 18]	iodine monitor range[13, 18]	liquid monitor range[13, 18]	Figure	Text	safety grade	check source
		In-Process continuous	ACF	In-Effluent continuous	In-Process sample	In-Effluent sample								
Nuclear Auxiliary Building Ventilation System (KLE) (continued)	R-16[9]	none	none	none	1 iodine and 1 aerosol sample points in the ventilation exhaust KLK42CR571 KLK42CR561	none	n/a	n/a	n/a	n/a	9.4.3-5	11.5.3.1.6	Non-safety	portable
Fuel Building Ventilation System (KLL) Test #144 Test #81	R-17	1 noble gas, 1 aerosol, and 1 iodine monitors on FBVS (cell 4) ventilation exhaust KLK34CR001 KLK34CR031 KLK34CR071	FBVS (cell 4) exhaust feeds into NABVS which diverts exhaust to iodine filtration on high activity[13]	none	1 aerosol grab sample point in the ventilation exhaust KLK34CR561	none	3E-7 - 1E-2 µCi/cc (Kr-85, Xe-133)	3E-10 - 1E-6 µCi/cc (Cs-137)	3E-10 - 5E-8 µCi/cc (I-131)	n/a	9.4.3-3	9.4.2 11.5.3.1.7	NS-AQ	portable
	R-18	1 noble gas, 1 aerosol, and 1 iodine monitors on FBVS (cell 5) ventilation exhaust KLK35CR001 KLK35CR031 KLK35CR071	FBVS (cell 5) exhaust feeds into NABVS which diverts exhaust to iodine filtration on high activity[13]	none	1 aerosol grab sample point in the FB ventilation exhaust KLK35CR561	none	3E-7 - 1E-2 µCi/cc (Kr-85, Xe-133)	3E-10 - 1E-6 µCi/cc (Cs-137)	3E-10 - 5E-8 µCi/cc (I-131)	n/a	9.4.3-3	9.4.2 11.5.3.1.7	NS-AQ	portable
	R-19	2 noble gas monitors on ventilation exhaust of the fuel handling area KLK38CR001 KLK38CR002	isolate fuel handling area ventilation (cell 5) feeds into SBVS on high exhaust activity, 2 noble gas monitors supply the signal	none	none	none	1E-5 - 1E+0 rad/hr	n/a	n/a	n/a	9.4.2-1 11.5-1	9.4.2.2.3 11.5.3.1.7	NS-AQ	portable

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Table 11.5-1—Radiation Monitor Detector Parameters
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Process System and Initial Test Program[10]	Rad Meas Pt	Monitor Provisions[11]			Sample Provisions[12, 14]		noble gas H-3 or N-16 monitor range [13, 18]	aerosol monitor range[13, 18]	iodine monitor range[13, 18]	liquid monitor range[13, 18]	Figure	Text	safety grade	check source
		In-Process continuous	ACF	In-Effluent continuous	In-Process sample	In-Effluent sample								
Radioactive Waste Building Ventilation System (KLF) Test #144 Test #80	R-20	1 iodine and 1 aerosol monitors upstream of ventilation filters in Cell 1 KLK50CR071 KLK50CR031	Close KLF room exhaust air normal carbon bypass isolation dampers Opens KLF carbon filtration unit isolation dampers and the air is directed through the carbon filter and exhausts this air through the plant vent stack	none	1 aerosol sample point upstream of the ventilation filters KLK50CR561	none	none	3E-10 - 1E-6 µCi/cc (Cs-137)	3E-10 - 5E-8 µCi/cc (I-131)	n/a	9.4.8-2	11.5.3.1.8	NS-AQ	portable
	R-21[9]	none	none	none	none	1 aerosol and 1 iodine sample points in the RWB ventilation exhaust KLK51CR561 KLK51CR571	none	none	none	n/a	9.4.8-2 11.5-1	11.5.3.1.8	non-safety	n/a

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Table 11.5-1—Radiation Monitor Detector Parameters
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Process System and Initial Test Program[10]	Rad Meas Pt	Monitor Provisions[11]			Sample Provisions[12, 14]		noble gas H-3 or N-16 monitor range [13, 18]	aerosol monitor range[13, 18]	iodine monitor range[13, 18]	liquid monitor range[13, 18]	Figure	Text	safety grade	check source
		In-Process continuous	ACF	In-Effluent continuous	In-Process sample	In-Effluent sample								
Radioactive Waste Building Ventilation System (KLF) (continued)	R-22	1 iodine and 1 aerosol monitors upstream of ventilation filters in Cell 2 KLK52CR071 KLK52CR031	Close KLF room exhaust air normal carbon bypass isolation dampers Opens KLF carbon filtration unit isolation dampers and the air is directed through the carbon filter and exhausts this air through the plant vent stack	none	1 aerosol sample point upstream of the ventilation filters KLK52CR561	none	none	3E-10 - 1E-6 µCi/cc (Cs-137)	3E-10 - 5E-8 µCi/cc (I-131)	n/a	9.4.8-2	11.5.3.1.8	NS-AQ	portable
	R-23	1 aerosol monitor upstream of ventilation filters in decontamination room KLK53CR031	none	none	1 aerosol sample point upstream of the ventilation filters KLK53CR561	none	none	3E-10 - 1E-6 µCi/cc (Cs-137)	n/a	n/a	9.4.8-1	11.5.3.1.8	non-safety	portable
	R-24	1 aerosol monitor upstream of ventilation filters in workshop KLK54CR031	none	none	1 aerosol sample point upstream of the ventilation filters KLK54CR561	none	none	3E-10 - 1E-6 µCi/cc (Cs-137)	n/a	n/a	9.4.8-1 11.5-1	11.5.3.1.8	non-safety	portable
Turbine Gland Sealing Condenser Vent System (MAW)	n/a	The turbine gland seal condenser vent joins the exhaust line from the main condenser evacuation system upstream of radiological measuring point R-3. Therefore, R-3 monitors both the main condenser evacuation system and the turbine gland seal condenser vent system. See R-3 for details.					n/a	n/a	n/a	n/a	n/a	n/a	n/a	n/a
Main Condenser Evacuation System (MAJ)	n/a	This system is used only during plant startup when an accident that could release gaseous radioactive material that could reach this system is unlikely and is therefore not monitored.					n/a	n/a	n/a	n/a	n/a	n/a	n/a	n/a

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Table 11.5-1—Radiation Monitor Detector Parameters
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Process System and Initial Test Program[10]	Rad Meas Pt	Monitor Provisions[11]			Sample Provisions[12, 14]		noble gas H-3 or N-16 monitor range [13, 18]	aerosol monitor range[13, 18]	iodine monitor range[13, 18]	liquid monitor range[13, 18]	Figure	Text	safety grade	check source
		In-Process continuous	ACF	In-Effluent continuous	In-Process sample	In-Effluent sample								
Evaporator Vent Systems	n/a	The boron recovery evaporator in the coolant treatment system vents to the ventilation cell in which it is located in the Nuclear Auxiliary Building. The exhaust from the cell is monitored as shown above in the Auxiliary Building ventilation system. The evaporator in the Radwaste Building is monitored by the Radwaste Building ventilation which is described above.					n/a	n/a	n/a	n/a	n/a	n/a	n/a	n/a
Liquid Waste Processing System (Pretreatment Liquid Radwaste Tank Vent Gas Systems (KPF))	n/a	All liquid radwaste processes are monitored by the Radwaste Building ventilation which is described above.					n/a	n/a	n/a	n/a	n/a	n/a	n/a	n/a
Steam Generator Blowdown System (Flash Tank and Steam Generator Blowdown Vent Systems (LCQ))	n/a	There are no vents to atmosphere in the steam generator blowdown system. The steam generator blowdown flash tank is vented to the feedwater tank where the vented vapor is mixed with the incoming feedwater and is conveyed to the steam generator.					n/a	n/a	n/a	n/a	n/a	n/a	n/a	n/a
Pressurizer & Boron Recovery Vent Systems (KT)	n/a	The pressurizer is vented to the Nuclear Island Drains and Vents System which routes gaseous radioactive wastes to the gaseous waste processing system. The coolant treatment system which contains the boron recovery sub-system is vented to the gaseous waste processing system.					n/a	n/a	n/a	n/a	n/a	n/a	n/a	n/a
Solid Waste Management System (Waste compactors, shredders, etc. (as permanently installed or mobile systems (KPE)	n/a	Solid waste components that could generate airborne wastes are monitored by the Radwaste Building ventilation which is described above.					n/a	n/a	n/a	n/a	n/a	n/a	n/a	n/a
Safeguard Building Controlled Area Ventilation System (KLC) Test #144 Test #83	R-25	1 noble gas, 1 aerosol and 1 iodine monitors on NABVS cell 6 ventilation exhaust KLK36CR001 KLK36CR031 KLK36CR071	SBVS (cell 6) exhaust feeds into NABVS which diverts exhaust to the NABVS iodine filtration on high activity[13]	none	1 aerosol sample points in the NABVS ventilation exhaust KLK36CR561	none	3E-7 - 1E-2 µCi/cc (Kr-85, Xe-133)	3E-10 - 1E-6 µCi/cc (Cs-137)	3E-10 - 5E-8 µCi/cc (I-131)	n/a	9.4.3-3	11.5.3.1.9	NS-AQ	portable
	R-26	none	none	2 noble gas accident monitors on SBVS controlled area accident exhaust train[2] KLK37CR001 KLK37CR002	none	1 iodine and 1 aerosol sample points in the SBVS controlled area accident exhaust train KLK37CR571 KLK37CR561	1E-4 - 1E+4 rad/hr	n/a	n/a	n/a	9.4.5-2	11.5.3	Non-safety	portable

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Table 11.5-1—Radiation Monitor Detector Parameters
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Process System and Initial Test Program[10]	Rad Meas Pt	Monitor Provisions[11]			Sample Provisions[12, 14]		noble gas H-3 or N-16 monitor range[13, 18]	aerosol monitor range[13, 18]	iodine monitor range[13, 18]	liquid monitor range[13, 18]	Figure	Text	safety grade	check source
		In-Process continuous	ACF	In-Effluent continuous	In-Process sample	In-Effluent sample								
Annulus Ventilation System (KLB) Test #144 Test #77	R-27	none	none	2 noble gas accident monitors in the ventilation exhaust KLK21CR001 KLK21CR002	none	none	1E-4 - 1E+4 rad/hr	n/a	n/a	n/a	6.2.3-2	6.2.3.2.2.2 11.5.3.1.10	NS-AQ	built-in
	R-28[9]	none	none	none	none	1 aerosol and 1 iodine accident sample points in the RWB ventilation exhaust KLK20CR561 KLK20CR571	n/a	n/a	n/a	n/a	6.2.3-2 11.5-1	6.2.3.2.2.2 11.5.3.1.10	Non-safety	n/a

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Table 11.5-1—Radiation Monitor Detector Parameters
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Process System and Initial Test Program[10]	Rad Meas Pt	Monitor Provisions[11]			Sample Provisions[12, 14]		noble gas H-3 or N-16 monitor range[13, 18]	aerosol monitor range[13, 18]	iodine monitor range[13, 18]	liquid monitor range[13, 18]	Figure	Text	safety grade	check source
		In-Process continuous	ACF	In-Effluent continuous	In-Process sample	In-Effluent sample								
Main Control Room Air Conditioning System (SAB) Test #144 Test #82 Test #143	R-29[19]	2 radiation monitors KLK65CR001 KLK65CR002	Opens CREF iodine filtration unit isolation dampers Closes the CREF iodine filtration unit bypass dampers Initiates CREF fans and the inlet air is directed through the carbon filters to the CRE	none	none	none	1E-5 - 1E+1 rad/hr	n/a	n/a	n/a	9.4.1-1	9.4.1.1 11.5.3.1.11	Safety	built-in
	R-30[19]	2 radiation monitors KLK66CR001 KLK66CR002	Opens CREF iodine filtration unit isolation dampers Closes the CREF iodine filtration unit bypass dampers Initiates CREF fans and the inlet air is directed through the carbon filters to the CRE	none	none	none	1E-5 - 1E+1 rad/hr	n/a	n/a	n/a	9.4.1-1	9.4.1.1 11.5.3.1.11	Safety	built-in

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Table 11.5-1—Radiation Monitor Detector Parameters
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Process System and Initial Test Program[10]	Rad Meas Pt	Monitor Provisions[11]			Sample Provisions[12, 14]		noble gas H-3 or N-16 monitor range[13, 18]	aerosol monitor range[13, 18]	iodine monitor range[13, 18]	liquid monitor range[13, 18]	Figure	Text	safety grade	check source
		In-Process continuous	ACF	In-Effluent continuous	In-Process sample	In-Effluent sample								
Access Building Ventilation System (KLD) Test #224	R-31[2]	none	none	none	1 aerosol and 1 iodine in the ventilation exhaust KLK40CR561 KLK40CR571	none	n/a	n/a	n/a	n/a	9.4.14-2	11.5.3.1.12	Non-safety	n/a
Liquid Radwaste (Batch) Effluent System (KPF) Test #144 Test #95 Test #215 Test #153	R-32	none	close the discharge valve on high activity	2 redundant monitors on the liquid radwaste release line KPK29CR001 KPK29CR002	none	H-3 grab sample and analysis	n/a	n/a	n/a	5E-6 - 1E-3 µCi/ml (Cs-137)	11.2-1 11.5-1	11.2.1.2.3 11.5.3.2	NS-AQ	portable
Liquid Radwaste (Continuous) Effluent System (KPF)	n/a	The US EPR uses a batch liquid radwaste effluent discharge which is monitored as described above.					n/a	n/a	n/a	n/a	n/a	n/a	n/a	n/a
Circulating Water System (PA)	n/a	For the US EPR, the CWS has no contact with the Turbine Building Plant Drainage.					n/a	n/a	n/a	n/a	n/a	n/a	n/a	n/a
Component Cooling Water System (KA) Test #144 Test #153	R-35	1 radiation monitor on each loop KAA10CR001	isolate the CCWS train on high activity	none	grab sample and analysis, H-3 analysis	none	n/a	n/a	n/a	1E-6 - 1E-3 µCi/ml (Cs-137)	9.2.2-1 Sh.1 11.5-1	11.5.4.4 9.2.2.2.1	NS-AQ	portable
	R-36	1 radiation monitor on each loop KAA20CR001	isolate the CCWS train on high activity	none	grab sample and analysis, H-3 analysis	none	n/a	n/a	n/a	1E-6 - 1E-3 µCi/ml (Cs-137)	9.2.2-1 Sh.1 11.5-1	11.5.4.4 9.2.2.2.1	NS-AQ	portable
	R-37	1 radiation monitor on each loop KAA30CR001	isolate the CCWS train on high activity	none	grab sample and analysis, H-3 analysis	none	n/a	n/a	n/a	1E-6 - 1E-3 µCi/ml (Cs-137)	9.2.2-1 Sh.1 11.5-1	11.5.4.4 9.2.2.2.1	NS-AQ	portable
	R-38	1 radiation monitor on each loop KAA40CR001	isolate the CCWS train on high activity	none	grab sample and analysis, H-3 analysis	none	n/a	n/a	n/a	1E-6 - 1E-3 µCi/ml (Cs-137)	9.2.2-1 Sh.1 11.5-1	11.5.4.4 9.2.2.2.1	NS-AQ	portable
Fuel Pool Cooling and Fuel Pool Purification System (FAK, FAL) Test #153	R-39[2]	none	none	none	grab sample and analysis, H-3 analysis	none	none	n/a	n/a	n/a	n/a	9.1.3.4 11.5.4.10	Non-safety	n/a

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Table 11.5-1—Radiation Monitor Detector Parameters
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Process System and Initial Test Program[10]	Rad Meas Pt	Monitor Provisions[11]			Sample Provisions[12, 14]		noble gas H-3 or N-16 monitor range [13, 18]	aerosol monitor range[13, 18]	iodine monitor range[13, 18]	liquid monitor range[13, 18]	Figure	Text	safety grade	check source
		In-Process continuous	ACF	In-Effluent continuous	In-Process sample	In-Effluent sample								
Nuclear Island Drain and Vent Systems (KT) Test #153	R-40[9]	none	none	none	grab sample and analysis, H-3 analysis	none	n/a	n/a	n/a	n/a	n/a	11.5.4.11	Non-safety	n/a
Phase Separator Decant & Holding Basin Systems (KPE)	n/a	For the US EPR, this is handled by the solid radwaste system.					n/a	n/a	n/a	n/a	n/a	n/a	n/a	n/a
Chemical & Regeneration Solution Waste Systems (KT)	n/a	For the US EPR these waste stream are collected by the Nuclear Island Drains and Vents System which is routed to the liquid radwaste storage tanks.					n/a	n/a	n/a	n/a	n/a	n/a	n/a	n/a
Nuclear Sampling System (KU) Test #144 Test #204 Test #205 Test #153 Test #100	R-41	1 noble gas monitor KUA66CR001	none	none	grab sample and analysis, H-3 analysis	none	3E-7 - 3E+3 µCi/cc (Kr-85, Xe-133)	n/a	n/a	n/a	9.3.2-1 Sh.2 11.5-1	11.5.4.6	NS-AQ	portable
Decontamination Waste Systems & Laundry Handling Room (FK) n/a	R-42	1 protective clothing monitor (no laundry processing on site) SRP10CR051 (Frisker)	none	none	none	none	n/a	n/a	n/a	n/a	n/a	11.5.4.1.2	Non-safety	portable

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Table 11.5-1—Radiation Monitor Detector Parameters
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Process System and Initial Test Program[10]	Rad Meas Pt	Monitor Provisions[11]			Sample Provisions[12, 14]		noble gas H-3 or N-16 monitor range[13, 18]	aerosol monitor range[13, 18]	iodine monitor range[13, 18]	liquid monitor range[13, 18]	Figure	Text	safety grade	check source
		In-Process continuous	ACF	In-Effluent continuous	In-Process sample	In-Effluent sample								
Solid Waste Management System (KPE) Test #93 Test #94	R-43	Process Monitor 7 dose rate detectors and 1 gamma spectroscopy system on the drum drying station KPC90CR501 KPC90CR502 KPC90CR503 KPC90CR504 KPC90CR505 KPC90CR506 KPC90CR507 KPC90CR508	none	none	none	none	n/a	n//a	n/a	1E-4 - 1E+0 rad/hr	11.4-1 11.5-1	11.5.4.13 11.4.1.2.4	Non-safety	portable
Rainwater Collection and Drainage System (Storm & Underdrain Water System)	n/a	Systems that are potentially radioactive are segregated from non-radioactive systems, such as rainwater collection and drainage systems, to minimize the migration of radioactive material across systems. Monitoring is not required.					n/a	n/a	n/a	n/a	n/a	n/a	n/a	n/a
Nuclear Island Drain and Vent System (KT)	R-44[9]	none	none	none	grab sample and analysis, H-3 analysis	none	n/a	n/a	n/a	n/a	n/a	11.5.4.11	Non-safety	n/a
Reactor Boron and Water Makeup System (KBC) Test #153	R-45[9]	none	none	none	grab sample and analysis, H-3 analysis	none	n/a	n/a	n/a	n/a	n/a	11.5.4.14	Non-safety	n/a

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**Table 11.5-1—Radiation Monitor Detector Parameters
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Process System and Initial Test Program[10]	Rad Meas Pt	Monitor Provisions[11]			Sample Provisions[12, 14]		noble gas H-3 or N-16 monitor range [13, 18]	aerosol monitor range[13, 18]	iodine monitor range[13, 18]	liquid monitor range[13, 18]	Figure	Text	safety grade	check source
		In-Process continuous	ACF	In-Effluent continuous	In-Process sample	In-Effluent sample								
Steam Generator Blowdown System (LCQ) Test #144 Test #67 Test#71	R-46	1 radiation monitor on each steam generator blowdown line QUC11CR001	none[7]	none	grab sample and analysis, H-3 analysis	none	n/a	n/a	n/a	3E-6 -1E-2 µCi/ml (Cs-137)	11.5-1	Table 9.3.2-2 11.5.4.3	Non-safety	portable
	R-47	1 radiation monitor on each steam generator blowdown line QUC12CR001	none[7]	none	grab sample and analysis, H-3 analysis	none	n/a	n/a	n/a	3E-6 -1E-2 µCi/ml (Cs-137)	11.5-1	Table 9.3.2-2 11.5.4.3	Non-safety	portable
	R-48	1 radiation monitor on each steam generator blowdown line QUC13CR001	none[7]	none	grab sample and analysis, H-3 analysis	none	n/a	n/a	n/a	3E-6 -1E-2 µCi/ml (Cs-137)	11.5-1	Table 9.3.2-2 11.5.4.3	Non-safety	portable
	R-49	1 radiation monitor on each steam generator blowdown line QUC14CR001	none[7]	none	grab sample and analysis, H-3 analysis	none	n/a	n/a	n/a	3E-6 -1E-2 µCi/ml (Cs-137)	11.5-1	Table 9.3.2-2 11.5.4.3	Non-safety	portable
Plant Drainage System (GM) Test #144	R-50	none	none	1 radiation monitor on the common release line	none	grab sample and analysis, H-3 analysis	n/a	n/a	n/a	3E-6 - 1E-2 µCi/ml (Cs-137)	11.5-1	11.5.4.15	Non-safety	built-in
Clean Drains System [4] (Noncontaminated Wastewater & Turbine Building Clean Drain System (LCM)) Test #144	R-65[9]	none	none	none	none	grab sample and analysis, H-3 analysis	n/a	n/a	n/a	n/a	n/a	11.5.4.16	Non-safety	n/a

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Table 11.5-1—Radiation Monitor Detector Parameters
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Process System and Initial Test Program[10]	Rad Meas Pt	Monitor Provisions[11]			Sample Provisions[12, 14]		noble gas H-3 or N-16 monitor range[13, 18]	aerosol monitor range[13, 18]	iodine monitor range[13, 18]	liquid monitor range[13, 18]	Figure	Text	safety grade	check source
		In-Process continuous	ACF	In-Effluent continuous	In-Process sample	In-Effluent sample								
Chemical and Volume Control System (High Pressure Coolers) Detection on CCWS Common Loop (KBA, KBD) Test #144 Test #153	R-51	radiation monitors are on the component cooling inlet of each cooler KAB60CR001	isolate the CCWS train on high activity	none	none	none	n/a	n/a	n/a	3E-5 - 3E+0 µCi/ml (Cs-137)	9.2.2-2 Sh.2	9.2.2.2.1 9.2.2.3.1 11.5.4.17	NS-AQ	built-in
	R-52	radiation monitors are on the component cooling outlet of each cooler KAB60CR002	isolate the CCWS train on high activity	none	none	none	n/a	n/a	n/a	3E-5 - 3E+0 µCi/ml (Cs-137)	9.2.2-2 Sh. 2	9.2.2.2.1 9.2.2.3.1 11.5.4.17	NS-AQ	built-in
	R-53	radiation monitors are on the component cooling inlet of each cooler KAB70CR001	isolate the CCWS train on high activity	none	none	none	n/a	n/a	n/a	3E-5 - 3E+0 µCi/ml (Cs-137)	9.2.2-3 Sh .2	9.2.2.2.1 9.2.2.3.1 11.5.4.17	NS-AQ	built-in
	R-54	radiation monitors are on the component cooling outlet of each cooler KAB70CR002	isolate the CCWS train on high activity	none	none	none	n/a	n/a	n/a	3E-5 - 3E+0 µCi/ml (Cs-137)	9.2.2-3 Sh .2	9.2.2.2.1 9.2.2.3.1 11.5.4.17	NS-AQ	built-in
Component Cooling Water System (KA) Test #144 Test #153	R-64	1 radiation monitor on the dedicated CCWS loop KAA50CR001	none	none	none	sample and analysis, H-3 analysis	n/a	n/a	n/a	1E-6 - 1E-3 µCi/ml (Cs-137)	9.2.2-4	9.2.2.2.1 9.2.2.3.1 11.5.4.17	Non-safety	built-in

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Table 11.5-1—Radiation Monitor Detector Parameters Sheet 16 of 17

Process System and Initial Test Program[10]	Rad Meas Pt	Monitor Provisions[11]			Sample Provisions[12, 14]		noble gas H-3 or N-16 monitor range [13, 18]	aerosol monitor range[13, 18]	iodine monitor range[13, 18]	liquid monitor range[13, 18]	Figure	Text	safety grade	check source
		In-Process continuous	ACF	In-Effluent continuous	In-Process sample	In-Effluent sample								
Main Steam lines (LB)	R-55[15]	4 N-16 monitors on each main steam line LBA10CR811 LBA10CR821 LBA10CR831 LBA10CR841	none[8]	none	none	none	1E-8 - 1E-2 µCi/cc (N-16)[17]	n/a	n/a	n/a	10.3-1 Sh. 1	11.5.4.1	Safety	built-in
	R-56[15]	4 N-16 monitors on each main steam line LBA20CR811 LBA20CR821 LBA20CR831 LBA20CR841	none	none	none	none	1E-8 - 1E-2 µCi/cc (N-16)[17]	n/a	n/a	n/a	10.3-1 Sh. 1	11.5.4.1	Safety	built-in
	R-57[15]	4 N-16 monitors on each main steam line LBA30CR811 LBA30CR821 LBA30CR831 LBA30CR841	none	none	none	none	1E-8 - 1E-2 µCi/cc (N-16)[17]	n/a	n/a	n/a	10.3-1 Sh. 1	11.5.4.1	Safety	built-in
	R-58[15]	4 N-16 monitors on each main steam line LBA40CR811 LBA40CR821 LBA40CR831 LBA40CR841	none	none	none	none	1E-8 - 1E-2 µCi/cc (N-16)[17]	n/a	n/a	n/a	10.3-1 Sh. 1	11.5.4.1	Safety	built-in
Safety Chilled Water System (QKA) Test #144	R-59	1 radiation monitor on train 1 downstream of the LHSI sealing fluid cooler	none	none	none	none	n/a	n/a	n/a	1E-6 - 1E-3 µCi/ml (Cs-137)	9.2.8-1 Sh. 3 11.5-1	9.2.8.4 11.5.4.18	Non-safety	portable
	R-60	1 radiation monitor on train 4 downstream of the LHSI sealing fluid cooler	none	none	none	none	n/a	n/a	n/a	1E-6 - 1E-3 µCi/ml (Cs-137)	9.2.8-1 Sh. 3 11.5-1	9.2.8.4 11.5.4.18	Non-safety	portable

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Table 11.5-1—Radiation Monitor Detector Parameters
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Process System and Initial Test Program[10]	Rad Meas Pt	Monitor Provisions[11]			Sample Provisions[12, 14]		noble gas H-3 or N-16 monitor range[13, 18]	aerosol monitor range[13, 18]	iodine monitor range[13, 18]	liquid monitor range[13, 18]	Figure	Text	safety grade	check source
		In-Process continuous	ACF	In-Effluent continuous	In-Process sample	In-Effluent sample								
Chilled Water for Gaseous Waste (QNA, QNB)	R-61[9]	none	none	none	sample and analysis, H-3 analysis	none	n/a	n/a	n/a	1E-6 - 1E-3 µCi/ml (Cs-137)	none	11.5.4.7	Non-safety	n/a
Essential Service Water System – Train Monitoring (PE) Test #144	R-66	1 monitor on Train 1 CCWS HX Outlet PEB10CR001	Alarm in the Control Room on high activity	none	sample and analysis, H-3 analysis	none	n/a	n/a	n/a	1E-6 - 1E-3 µCi/ml (Cs-137)	9.2.1-1, Sh.2 11.5-1	9.2.1.2 11.5.4	Non-safety	portable
	R-67	1 monitor on Train 2 CCWS HX Outlet PEP20CR001	Alarm in the Control Room on high activity	none	sample and analysis, H-3 analysis	none	n/a	n/a	n/a	1E-6 - 1E-3 µCi/ml (Cs-137)	9.2.1-1, Sh.2 11.5-1	9.2.1.2 11.5.4	Non-safety	portable
	R-68	1 monitor on Train 3 CCWS HX Outlet PEP30CR001	Alarm in the Control Room on high activity	none	sample and analysis, H-3 analysis	none	n/a	n/a	n/a	1E-6 - 1E-3 µCi/ml (Cs-137)	9.2.1-1, Sh.2 11.5-1	9.2.1.2 11.5.4	Non-safety	portable
	R-69	1 monitor on Train 4 CCWS HX Outlet PEP40CR001	Alarm in the Control Room on high activity	none	sample and analysis, H-3 analysis	none	n/a	n/a	n/a	1E-6 - 1E-3 µCi/ml (Cs-137)	9.2.1-1, Sh.2 11.5-1	9.2.1.2 11.5.4	Non-safety	portable
Essential Service Water System – Dedicated Train Test #144	R-70	1 monitor on Dedicated Train CCWS HX Outlet PEB80CR001	Alarm in the Control Room on high activity	none	sample and analysis, H-3 analysis	none	n/a	n/a	n/a	1E-6 - 1E-3 µCi/ml (Cs-137)	9.2.1-1, Sh.4 11.5-1	9.2.1.2 11.5.4	Non-safety	portable

Notes:

1. This is a non-safety Automatic Control Feature which diverts flow to the KLE iodine filtration train.

2. Note that a containment isolation signal or high radiation from the in-containment high range monitors (i.e., an accident has occurred) will automatically divert SBVS and FBVS exhaust flow to the KLC accident exhaust filtration trains (Refer to Figure 9.4.5-2 for the SBVS and Figure 9.4.2-1 and Figure 9.4.5-2 for the FBVS). This is a separate iodine filtration train from the KLE iodine filtration train which is shown on Figure 9.4.3-4.

3. The Fuel Building exhaust (cells 4 & 5) and the Safeguard Building exhaust (cell 6) feed into the NABVS as shown on Figure 9.4.3-3. The radiation detector is very close to the boundary between the FBVS and NABVS and the boundary between the SBVS and NABVS. While the exhaust comes from the FBVS and SBVS, the radiation detectors and the automatic control features are within the NABVS. Hence, the diversion of exhaust to the NABVS iodine filtration train on high activity.

4. The clean drains are collected and routed to the main condenser.

5. Footnote 5 not used.

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6. This monitor is used for RCS leakage detection to satisfy TS 16.3.4.14.
7. Steam generator blowdown high activity does not alone constitute a signal to isolate SGBS, however when coupled with a partial cooldown signal, the SGBS of the affected steam generator will automatically be isolated.
8. The main steam activity does not alone constitute a signal for an automatic control feature. The electronic logic is that a partial cooldown signal AND either a main steam high activity OR a steam generator high level will generate a signal to automatically isolate the affected steam generator.
9. This is a sample point that has no monitoring functionality.
10. Initial Test Program (ITP) Test #144 is the primary test for Process and Effluent Radiological Monitoring System (PERMS). Other ancillary ITP tests are also listed for systems utilizing radiation monitors.
11. Radiation monitor designations are as follows:
    - CR001/CR002: Noble Gas or liquid gamma or beta activity monitors.
    - CR031/CR032: Normal operations / post-accident aerosol/particulate monitors.
    - CR071/CR072: Normal operations / post-accident iodine monitors.
12. Sampler designations are as follows:
    - CR551: H-3 and C-14 air sampler evaluated in laboratory.
    - CR561/CR562: Aerosol sampler with particle filter (normal operations / post-accident) evaluated in laboratory.
    - CR571/CR572: Iodine sampler with common filter bed (normal operations / post-accident) evaluated in laboratory.
    - CR581/CR582: Air grab samples (normal operations / post-accident).
13. A COL applicant that references the U.S. EPR design certification is responsible for deriving PERMSS subsystem's lower limits of detection or detection sensitivities, and setpoints (alarms and process termination/ diversion) for liquid and gaseous process radiation monitoring equipment not covered by the ODCM based on plant and site-specific conditions and operating characteristics of each installed radiation monitoring subsystem (COL 11.5-3).
14. A COL applicant that references the U.S. EPR design certification is responsible for developing a plant-specific process and effluent radiological sampling and analysis plan for systems not covered by the ODCM, including provisions describing sampling and analytical frequencies, and radiological analyses for the expected types of liquid and gaseous samples and waste media generated by the LWMS, GWMS, and SWMS (COL 11.5-4).
15. This monitor is used for primary to secondary leak redetection to satisfy TS 16.3.4.12 and industry guidance in NEI-97-06.
16. Radionuclide concentrations for the RCS leakage monitor test will be selected to be consistent with the design basis description in Section 11.5.4.8.
17. Radionuclide concentrations for the primary to secondary leakage monitor test will be selected to be consistent with the design basis description in Section 11.5.4.1
18. The radionuclide concentrations for testing monitors will be selected to be a fraction of the monitoring range.
19. Radionuclide concentration for testing this monitor will be selected to be above three times the nominal background at full power (See Table 15.0-8).

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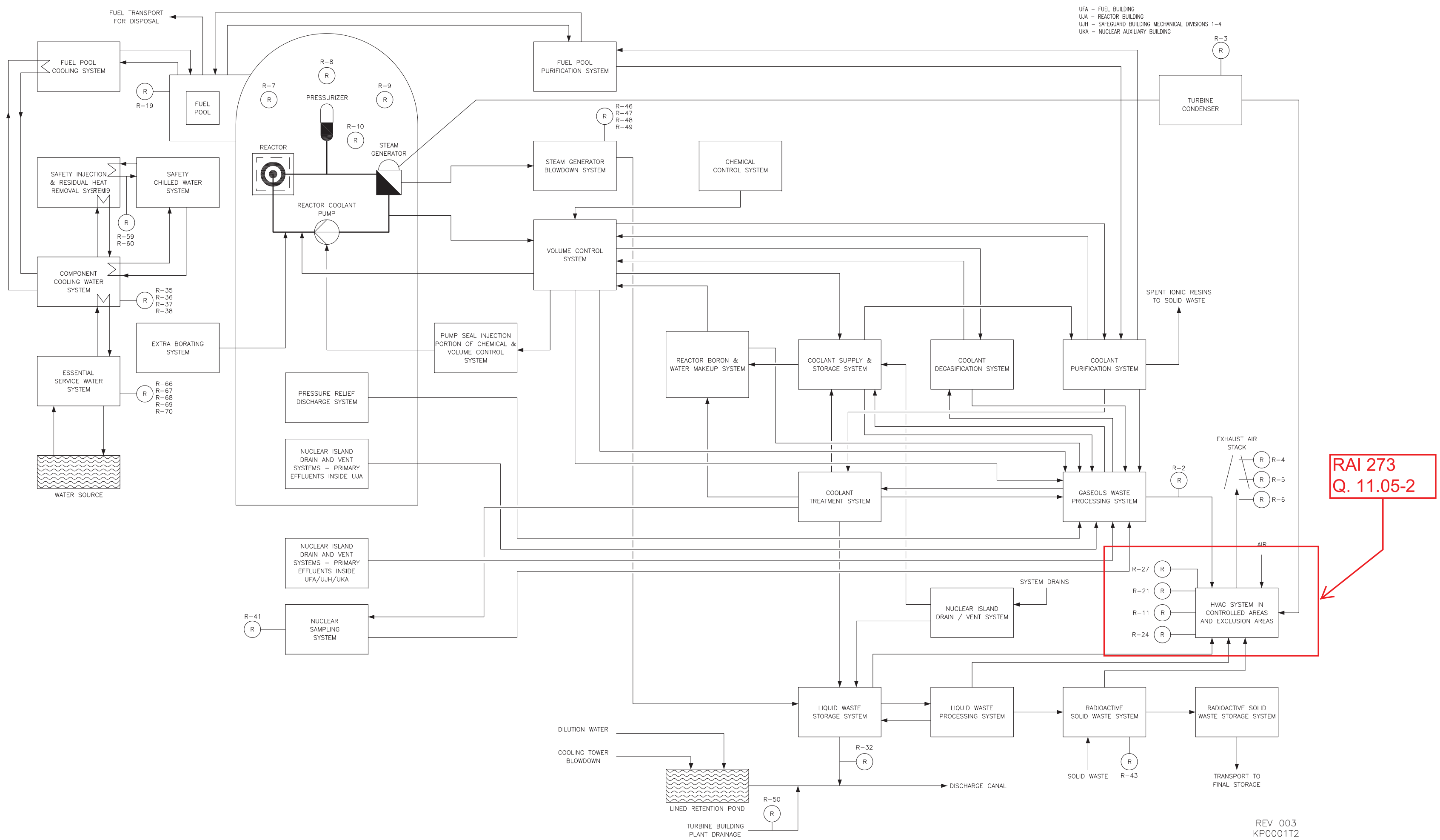
Figure 11.5-1—Radioactive Effluent Flow Paths With Process and Effluent Radiation Monitors



- Provide a local readout, an audible alarm, and visual alarms outside of the room in which the detector is located and are visible to operating personnel prior to entry.

The accident area radiation monitors have usable ranges that include the maximum calculated accident levels and are designed to operate effectively under the environmental conditions caused by an accident. These monitors follow the guidance of RG 1.97 (refer to Section 7.5). This instrumentation is powered by the Class 1E uninterruptible power supply (EUPS), described in Section 8.3.1, which is served by a two-hour battery backup with diesel generators as the auxiliary power to provide continuous indication.

Table 12.3-3 includes area radiation monitoring instrumentation.

Refer to Section 7.5 for information regarding specific post-accident monitoring instrumentation.

12.3.4.1.3 In-containment High-Range Monitoring

The in-containment monitoring instrumentation used during postulated accidents is provided to:

- Measure gamma radiation, primarily from airborne gaseous radioactivity.
- Deliver a signal to the MCR to alert operators when predetermined setpoints are reached.
- Record data from the monitors to maintain a record of the gamma radiation after an accident as a function of time so that the inventory of radioactive materials in the containment volume can be estimated.

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- Initiate Stage 1 Containment Isolation on high range radiation monitor signal inside the Reactor Building.
- ~~Initiate Safeguard Building controlled-area ventilation system isolation and exhaust filtering on high radiation inside the Reactor Building.~~
- ~~Initiate Fuel Building ventilation system exhaust filtering on high radiation in the Reactor Building, concurrent with Safeguard Building controlled area ventilation system alignment.~~

These safety-related instruments and the associated network are environmentally qualified (refer to Section 3.11) to survive an accident and perform their design functions. The instruments are designed to respond to gamma radiation over the energy range of at least 60 keV to 3 MeV, with a dose rate response accuracy within a factor of two over the entire range. These monitors conform to the criteria set forth in 10 CFR 50.34(f)(2)(xvii), NUREG-0737, II.F.1 (Reference 3), and RG 1.97 (refer to Section 7.5). These monitors also meet the applicable requirements of IEEE Std 497-2002 (Reference 7). This instrumentation is powered by the EUPS (refer to

Section 8.3.1), which is served by a two-hour battery backup with diesel generators as the auxiliary power to provide continuous indication.

The in-containment high-range monitoring instrumentation consists of four independent high-range monitors located in widely separated areas in the service compartment of the containment. The locations are chosen to allow the detectors to be exposed to a significant volume of the containment atmosphere without obstruction so that the readouts are representative of the containment atmosphere, yet permitting easy access for calibration and maintenance activities.

Table 12.3-3 includes the high-range monitoring instrumentation.

12.3.4.2 Airborne Radioactivity Monitoring Instrumentation

12.3.4.2.1 Normal Operations

The airborne radioactivity monitoring instrumentation for use during normal operation and AOOs is provided to:

- Continuously monitor for the presence of airborne radioactivity at selected locations of the plant that are normally occupied and may contain airborne radioactivity.
- Detect derived air concentrations in air (DAC) of the most restrictive particulate and iodine radionuclides in any compartment or room served by lowest ventilation rate within 10 hours (i.e., 10 DAC-hours) in accordance with Section 12.3-12.4 in NUREG-0800 (Reference 1).

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- Verify the integrity of systems that contain radioactive material.
- Warn of unexpected releases of airborne radioactive material.
- Initiate automatic air isolation of NABVS and exhaust the fuel handling area by SBVS when a high exhaust activity setpoint is reached or instrument failure is detected (refer to Table 12.3-4 and Table 11.5-1, Monitor R-19).
- Initiates automatic ~~isolation~~closure of the ~~MCR intakes~~CREF bypass inlet isolation dampers, opens CREF inlet isolation dampers, and initiates CREF fans when a high activity setpoint (refer to Table 11.5-1, Monitors R-29 and R-30 for automatic control functions) is reached or instrument failure is detected. This is an ESF system (refer to Section 7.3).

This instrument category, other than ESF system detectors, is designated only for routine monitoring and is powered by the non-1E power supply (refer to Section 8.3.1), which has no auxiliary power. The ESF system detectors are powered by the EUPS (refer to Section 8.3.1) which is served by a two-hour battery backup with diesel generators as the auxiliary power to provide continuous indication.

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coolant storage tanks are installed is designed with a leak retention capability equivalent to the complete drainage of one tank. A steam generator tube leak will be detected by continuous process radiation monitors or radiochemical grab sampling on the secondary side of the steam generator as described in Section 5.4.2.5.2.5.

12.3.6.1.6 Maintenance Activities - Reactor Coolant Pumps

To reduce facility contamination due to maintenance activities involving the reactor coolant pumps, design features such as a removable shaft and permanently installed decontamination equipment are provided (see Sections 12.3.1.3 and 12.3.1.9.2). In addition, a dedicated room is provided for reactor coolant pump maintenance. For portions of the piping system potentially requiring inspections, maintenance, or repairs, bolted flanges are provided. For the tanks in the CSSS, inspection and maintenance can be conducted during plant operations as any one tank may be isolated, drained, purged, and opened independently of other tanks while maintaining the normal functions of the system. Prior to performing maintenance activities, the coolant storage tanks can be decontaminated. The monitoring instruments of the CSSS are located in accessible rooms for ease of inspection and maintenance.

12.3.6.1.7 Access Building - Eyewash and Shower Waste-Water

Eyewash stations and shower wastewater in the Access Building are routed to a tank in the nuclear island drain and vent system. Liquid effluent from the decontamination facilities (e.g., showers, floor washing) will be collected and stored in the storage tanks of the nuclear island drain and vent system.

12.3.6.2 Contamination Control for the Environment

Tanks that contain potentially radioactive liquids are located inside the NI structures. These tanks are all above the floor level and can be inspected and repaired in the event of a leak. The liquid from potential tank leaks is contained by berms and collected by the plant drain system for processing in the liquid waste management system. The only tank-like structures that are below grade are the UHS cooling tower basins, which is not a radiological system.

NI floor drains, sumps and piping that transfer potentially radioactive liquids to the liquid waste management system are designed with barriers and leakage detection instrumentation. These barriers and detection instrumentation minimize the introduction of uncontrolled radioactive effluent into the environment.

The only pathway allowed for the discharge of a liquid effluent is after treatment from the liquid waste management systems. Liquid effluent activity and volumetric flow are recorded continuously in the Radioactive Waste Building during discharges to allow for immediate intervention in case any limit is exceeded. A vertical U-bend trap in the piping ahead of the Radioactive Waste Processing Building outlet serves to radioactivity from the atmosphere above the spent fuel pool and processes it through high efficiency particulate air filters and charcoal adsorber units to the unit vent.

There are no portions of the spent fuel pool system handling potentially contaminated material that are buried or routed through exterior boundaries. Leak detection under the spent fuel pool provides full coverage in case of a leak, and leak detection equipment in channels aid in identifying the location of the leak. Sumps that collect potential spent fuel pool leakage are double lined with non-porous material. In addition, walls and curbs are used in areas with potential leaks of contaminated fluids to prevent the spread of these fluids.

12.3.6.5.2 Process Sampling System

The process sampling systems of the U.S. EPR are designed to minimize contamination of the facility and the environment as described in the general protective design features listed in Sections 12.3.6.1 and 12.3.6.2. Nuclear sampling, sample activity monitoring, and radiation monitoring comprise the process sampling system.

The process sampling system is described in Section 9.3.2.

Design Provisions for Minimizing Contamination of the Facility

To minimize potential contamination of the facility, the process sampling systems are designed to:

- Monitor for potential higher than normal levels of radiation in the facility, and thereby, provide a means to mitigate it from spreading to other parts of the facility.
- Monitor variables and systems over their anticipated ranges to assure adequate safety, including those variables and systems that can affect the integrity of the reactor core and the reactor coolant pressure boundary.
- Provide a confinement boundary against any releases from the sampling system.
- Confirm that contaminated fluids are not transferred to non-contaminated fluids.

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The process sampling systems monitor radioactivity levels in plant process streams and atmospheres, indicate and alarm excessive radioactivity levels, and in some cases automatically initiate protective isolation actions via radiation monitors (refer to Table 12.3-4 and Table 11.5-1) to minimize potential contamination of the facility. The systems consist of permanently installed, continuous monitoring devices together with a program of, and provisions for, specific sample collections and laboratory analyses. For example, area radiation monitors located in the Safeguard and Radioactive Waste Processing Buildings are provided to continually monitor radiation levels in the spaces which contain components for recirculation of loss of coolant accident (LOCA) fluids and components for processing radioactive wastes. In case of high levels of radiation, both local alarms and signals to the MCR are provided as well as automatic control functions (refer to Table 12.3-4 and Table 11.5-1). Additional process monitoring functions are detailed in Section 11.5.4. The process sampling systems also provide information regarding the release of radioactive materials during normal operations, anticipated operational occurrences, and postulated accidents to provide an early indication of the need to initiate other protective actions to minimize potential facility contamination. For example, under accident conditions samples of the containment atmosphere can be taken via the sampling activity monitoring system to provide data on airborne radioactive concentrations within the containment.

The process sampling system obtains and analyzes key chemistry parameters such as chloride, hydrogen, and oxygen concentrations in the reactor coolant. The control of corrosive chemicals decreases the potential for facility contamination by decreasing the probability that the reactor coolant pressure boundary or fuel cladding are compromised due to degradation from corrosive chemical attack.

To minimize the potential for facility contamination due to a leak from the process sampling systems, sample lines penetrating the containment are capable of isolation upon receipt of a containment isolation signal from the reactor protection system. In addition, the portion of the process sampling system that includes the reactor coolant pressure boundary is designed, fabricated, erected, and tested to have a low probability of abnormal leakage, rapidly propagating failure and gross rupture. Motor-operated isolation valves in the three nuclear sampling lines connected to the reactor coolant system (RCS) maintain the reactor coolant pressure boundary integrity. Sample (glove) boxes are used to collect active liquid grab samples to confine any spills. Safety-related portions of the process sampling systems are designed to withstand the effects of natural phenomena. Non-safety-related portions of the process sampling systems are designed to have provisions for a leakage detection and control program to minimize the leakage from those portions of the process sampling systems outside of the containment that contain or may contain radioactive material following an accident.

The design of the process sampling system prevents the inadvertent transfer of contaminated fluids to non-contaminated drainage systems. This is accomplished by transferring contaminated fluids either back to the system being sampled or to an appropriate radwaste system.

The components are designed to permit periodic testing and in-service inspections during plant operation and are designed for the life of the plant. The piping connections and joints in these systems are welded except where flanged or screwed connections are required to facilitate equipment removal for inspection, maintenance, or pressure testing.The pipes inside containment are routed with a continuous slope without low points and each sample line is equipped with an inner and outer containment isolation valve. In addition, there is a sampling isolation valve in each line that belongs to the RCS. Sample lines are flushed for a sufficient period of time prior to sample extraction to remove sediment deposits and air and gas pockets. In addition, samples from tanks are taken from the bulk volume to avoid low points and sediment traps. Decontamination fluid can be injected via dedicated nozzles to vessels in the process sampling systems.

Design Provisions for Minimizing Contamination of the Environment

The process sampling systems minimize contamination of the environment by: (1) monitoring the atmosphere in various locations of the facility and taking actions to minimize potential releases from the facility; (2) monitoring the effluents discharged from the various ventilation systems addressed in Section 9.4 and taking actions to minimize potential releases from the facility; (3) controlling and potentially reducing the concentration and quality of fission products potentially released following postulated accidents; and (4) providing protection against leaks from sampling equipment.

The process sampling systems provide radiation and airborne monitors at various plant locations to assist in the detection of abnormal operational conditions. Upon detection of contamination, these monitors provide indication and alarms in the MCR and health physics office to initiate actions to minimize environmental contamination. For example, the containment atmosphere is monitored during normal and transient operations by containment gaseous radiation monitors and under accident conditions, can initiate RB containment isolation, and thereby, minimize any releases to the environment. In addition, sampling points are located on the process radiological monitoring and sampling systems to permit representative sampling for radiochemical analysis to indicate the existence of and, to the extent possible, the magnitude of reactor coolant and reactor auxiliary system leakage to the containment atmosphere, cooling water systems, and the secondary side of the steam generators. Process monitors also provide an alarm and gross indication of the extent of failed fuel. They also monitor radioactive waste systems and associated handling areas to detect and alarm under conditions that may result in a loss of residual heat removal capability and excessive radiation levels. In each of these cases, the process sampling system has monitors that provide indications and alarms to the MCR to initiate actions to minimize potential environmental contamination.

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The process sampling systems also continuously monitor facility radioactivity levels in the effluent discharge paths during normal and accident conditions. The gaseous effluent monitoring and sampling system monitors the Reactor Containment Building, the FB, the Nuclear Auxiliary Building, the mechanical area of the Safeguard Buildings, the controlled area of the Access and Radioactive Waste Processing Buildings and the vent stack. Sampling points are located on effluent radiological monitoring and sampling systems to permit representative sampling for radiochemical analysis. The gaseous effluent radiological monitoring and sampling systems alarm but perform ~~no automatic actions~~various automatic control functions (refer to

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Table 12.3-4 and Table 11.5-1), when radionuclide concentrations exceed the specified limits. As stated in Section 11.5.2, a COL applicant will describe in the Offsite Dose Calculation Manual (ODCM) how a gaseous radiological release will be controlled.

The liquid effluent radioactive waste monitoring and sampling system measures the concentration of radioactive materials in liquids to be discharged to the environment in batches from waste monitoring tanks. Prior to release of a liquid radioactive waste from a monitoring tank, the system obtains a representative sample which is radiochemically analyzed. If the sample is acceptable, flow from these tanks to the environment is permitted. The flow is monitored and if radionuclide concentrations exceed the specified limits, the discharge to the environment is automatically isolated upstream prior to any unacceptable release to the environment (refer to Table 11.5-1, Monitor R-32).

The non-safety-related portions of the process sampling systems are designed to control fission products, chloride, hydrogen, oxygen, and other substances that may be released into the reactor containment and also, reduce the concentration and quality of fission products released to the environment following postulated accidents. The control of corrosive chemicals decreases the potential for a release to the environment by decreasing the probability that the reactor coolant pressure boundary is compromised due to degradation from corrosive chemical attack. Non-safety-related portions of the process sampling systems include means to suitably control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences.

There are no buried pipes in the process sampling systems that handle potentially contaminated liquids, and hence, no means to contaminate the environment from a leaking pipe. There are also no by-pass lines around the radiation monitors for the liquid effluents released from the waste monitoring tanks. Gases that may potentially leak from these process sampling systems are collected by one of the HVAC systems described in Section 9.4 and subsequently filtered prior to a release from the vent stack.

12.3.6.5.3 Coolant Supply and Storage System

The coolant supply and storage system is designed to minimize contamination of the facility and the environment as described in the general protective design features listed in Section 12.3.6.1 and 12.3.6.2.

Design Provisions for Minimizing Contamination of the Facility

To minimize the potential to contaminate the facility, the coolant supply and storage system is designed so that any leakage is detected, quantified, and the location of the leak determined by the leakage detection system. The RCS pressure boundary provides the first barrier against reactor coolant contaminating the facility and the second barrier against the release of radioactivity from the fission products of the fuel into the facility. The coolant supply and storage system is designed to provide a secure envelope for the retention of reactor coolant and associated gases. This system uses vessels and welded piping (including for the local sampling system) to provide a barrier against leakage and is equipped with manual valves to provide system isolation from non-contaminated support systems such as the demineralized water distribution system. The piping and equipment exposed to coolant are austenitic stainless steel to avoid corrosion issues. In addition, level measurements in the vessels of this system prevent high levels by automatically isolating inlet supplies to the vessels and preventing cross contamination of interfacing support systems. In case of a leak, these level measurements also identify the vessel that is leaking by a low level measurement. The compartment where the coolant storage tanks are installed is designed with a leak retention capability equivalent to the complete drainage of one tank. If the leak is due to a steam generator tube leak, it is detected by continuous process radiation monitors or radiochemical grab sampling.

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The leakage detection systems, in combination with instrumentation from other interconnected systems, detect, quantify, and determine the location of leakage from the reactor coolant pressure boundary. These systems provide a method of collecting and quantifying reactor coolant pressure boundary leakage. These leakage detection systems include diverse measurement methods such as sump level and discharge flow monitors, containment atmosphere radiation monitors, containment air cooler condensate flow monitors, containment humidity monitors, temperature monitors of the reactor vessel closure joint, and reactor coolant inventory monitors at the pressurizer, volume control tank, and coolant drain collection tank. Provisions are also incorporated into the U.S. EPR to isolate, capture, and quantify leakage from known potential sources, such as flanges and relief valves, so that such leakage may be monitored separately from unidentified leakage. Each of these monitoring systems provide indications of leak rates and leak locations to the plant operators in the MCR.

Leakage of reactor coolant into the component cooling water system (CCWS) from a residual heat removal heat exchanger tube, reactor coolant pressure seal thermal barrier, or other source is identified by increased activity in the CCWS fluid as detected by a continuous monitor or routine sampling, and is also indicated by an unexpected increasing level in the surge tank. The dedicated CCWS surge tank is charged by nitrogen over-pressurization, resulting in potential component coolant leakage into, rather than out of, most interfaces with contaminated fluids (e.g., the severe accident heat removal system). For potentially contaminated systems operating at pressures greater than the CCWS, radiation and flow monitors in the CCWS detect and allow actions to be taken to limit the leakage into the system. For example, the chemical volume and control system (CVCS) high pressure coolers operate at pressures greater than the CCWS, and upon a high pressure cooler tube rupture results in a leak

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of reactor coolant into the CCWS. This leakage into the CCWS is detected by the CCWS flow meters (increased flow) or radiation monitors (increased radioactivity) and the high activity measurement generates a signal to automatically close the cooler isolation valves isolating the CVCS HP cooler from the CCWS to minimize the leakage into CCWS (refer to Table 11.5-1, Monitors R-51 through R-54).

In addition to maintaining the confinement barriers of the reactor coolant, facility contamination is minimized by the compartmentalization of buildings that contain portions of the reactor coolant pressure boundary. The Containment Building is divided into two compartments: an inner equipment compartment and an outer service compartment. The inner compartment contains the steam generators, reactor coolant pumps, and primary loop piping. The outer compartment contains support equipment. In the event of a reactor coolant leak, facility contamination is minimized by containing the majority of the contamination in the inner compartment. Similarly, the portion of the Safeguard Buildings that coolant passes through is in the radiological controlled areas of these buildings which are separated from the non-radiological areas (i.e., uncontrolled areas) that contain items such as instrumentation, control equipment, and switchgear. To minimize the spread of contamination, these two areas of the Safeguard Buildings are served by separate ventilation systems with the radiological controlled area ventilated by the safeguards building controlled-area ventilation system described in Section 9.4.5. The reactor coolant storage tanks that reside in the Nuclear Auxiliary Building are located in a similarly compartmentalized area within this building. This potentially contaminated area is ventilated by the nuclear auxiliary building ventilation system described in Section 9.4.3.

To minimize facility contamination caused by leaks of contaminated fluids, coatings such as sealers or special paint are used on walls, ceilings, and floors potentially exposed to these leaks to permit easy decontamination. The coating-sealed first level of the concrete compartments housing the coolant storage tanks contain elevated access doors, sealed floor penetrations, and floor drains that are normally closed so that each compartment has the capability of retaining the complete drainage of one coolant storage tank. This feature minimizes the spread of contamination in the unlikely event of such a leak.

To minimize facility contamination due to maintenance activities involving the reactor coolant pumps, design features such as a removable shaft and permanently installed decontamination equipment are provided. A dedicated room for maintenance of the reactor coolant pumps is provided in the U.S. EPR. In addition, bolted flanges are provided on the piping system only where removal is required for inspection, maintenance, or repair. For the tanks in the coolant supply and storage system, system inspection, and maintenance can be conducted during plant operations as any one tank may be isolated, drained, purged, and opened independently of other tanks while maintaining the normal functions of the system. Prior to performing maintenance activities, the coolant storage tanks can be decontaminated. The

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tank continuously measure and record the volume and flow rate as the treated wastewater is released. If the total activity indicated by sensors exceeds predetermined limits, control signals are generated automatically to close two redundant downstream isolation valves, close the upstream isolation valve, and shut down the operating recirculation and discharge pump(s) (refer to Table 12.3-4 and Table 11.5-1, Monitor R-32). The content of the monitoring tanks is then sent back to the processing system for further treatment.

For gases, the U.S. EPR liquid waste management system receives degasified liquids in the storage tanks. These tanks are continuously vented to the radioactive waste processing building ventilation system (refer to Section 9.4.8) so that any generation of gaseous activity is continually removed. The gaseous waste processing system is primarily designed to collect radioactive waste gases from the various systems in which they are released, to process these waste gases and provide sufficient holdup time for radioactive decay to reduce the activity present, and to control the subsequent release of processed waste gases to the environment in compliance with regulatory limits. This system maintains a negative system pressure to prevent the escape of radioactive gases from the components connected to it.

Releases from the gaseous waste processing system are continuously monitored by radiation sensors in the delay system discharge line. The system also provides grab sample collections for analysis from several different points on the process stream, and from each of the delay beds along the discharge line. Gaseous waste processing system releases are routed through the filtration system of the nuclear auxiliary building ventilation system (refer to Section 9.4.3 for information on this HVAC system).

For spills from liquid tanks outside of containment, the U.S. EPR provides design features to control and collect radioactive material spills. The tanks for these systems are contained in rooms with drains to collect any spills and to prevent any uncontrolled release to the environment (refer to Section 9.3.3). If a leak escapes into the room containing a waste system vessel, then the room contains the leak or drains the leak to a nearby sump. The floor drain from a room can be opened to drain the leakage into a sump. From the sump, the liquid is pumped into a storage tank in the liquid waste storage system.

For the gaseous release associated with spills from these systems, the U.S. EPR provides the radioactive waste building ventilation system which is addressed in Section 9.4.8. Other portions of the solid waste treatment system contain sorting boxes used to sort the various dry active wastes produced in the controlled areas of the plant. These sorting boxes contain hand holes with rubber gloves for sorting the wastes and are connected to the radioactive waste processing building ventilation system through a filling hood. Any airborne contaminants created during the sorting, shredding, or compaction processes are captured by the filling hood and subsequently treated in the radioactive waste building ventilation system.

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Table 12.3-3—Radiation Monitor Detector Parameters (Sheet 2 of 3)

Monitor Location	Monitor Provisions		Range
	Continuous	ACF	
	~~2 monitors in air leaving containment next to air duct (downstream KLA2 low flow purge exhaust)~~	~~—~~	~~1E-5 – 1E+0 rad/hr~~
	~~2 monitors in exhaust air from exhaust cell (downstream KLB accident exhaust filter)~~	~~—~~	~~1E-4 – 1E+4 rem/hr (Kr-85, Xe-133)~~
	~~2 monitors in exhaust air from exhaust cell (downstream KLC accident exhaust filter)~~	~~—~~	~~1E-4 – 1E+4 rem/hr (Kr-85, Xe-133)~~
Safeguard Building (Mechanical)	~~1 monitor – Service Corridor near Containment Heat Removal System (elevation -31' Division 1)~~	~~—~~	~~1E-4 – 1E+4 rem/hr~~
	1 monitor – Service Corridor near Safety Injection System (elevation -31' Division 2)	---	1E-4 – 1E+4 rem/hr
	1 monitor – Service Corridor near Safety Injection System (elevation -31' Division 3)	---	1E-4 – 1E+4 rem/hr
	1 monitor – Service Corridor near ~~Containment~~Severe Accident Heat Removal System (elevation -31' Division 4)	---	1E-4 – 1E+4 rem/hr
	1 monitor – Personnel Air Lock Area (elevation 0' Division 2)	---	1E-4 – E+4 rem/hr
Safeguard Building (Electrical)	1 monitor – MCR (+53' elevation Division 2/3)	---	1E-4 – 1E+4 rem/hr

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Table 12.3-4—Airborne Radioactivity Detector Parameters (Sheet 1 of 5)

Monitor Location	Monitor Provisions		Range[1]
	In-Process Continuous	ACF	
Reactor Building	1 noble gas monitor at (R-10) refueling machine (used during spent fuel movement only)	---	1E-6 – 1E-2 μCi/cc (Kr-85, Xe-133)
	1 noble gas monitor (R-10) in exhaust containment ventilation (upstream KLA05 filters)	---	3E-7 – 1E-2 μCi/cc (Kr-85, Xe-133)
	1 aerosol monitor (R-10) in exhaust from containment ventilation (upstream KLA05 filters)	---	5E-4 – 3E+0 μCi 3E-10 – 1E-6 μCi/cc Must be capable of detecting 10 DAC-hours
	1 gaseous iodine monitors (R-10) in exhaust from containment ventilation (upstream KLA05 filters)	---	5E-4 – 3E+0 μCi 3E-10 – 5E-8 μCi/cc (I-131) Must be capable of detecting 10 DAC-hours
Fuel Building (Figure 12.3-73)	~~1 noble gas monitor at spent fuel mast bridge (used during spent fuel movement only)~~	~~—~~	~~1E-6 – 1E-2 μCi/cc (Kr-85, Xe-133)~~
	1 noble gas monitor (R-7) in exhaust air of containment ventilation (upstream KLA2 filters)	---	3E-7 – 1E-2 μCi/cc (Kr-85, Xe-133)
	1 aerosol monitor (R-7) in exhaust air of containment ventilation (upstream KLA2 filters)	---	5E-4 – 3E+0 μCi 3E-10 – 1E-6 μCi/cc Must be capable of detecting 10 DAC-hours

Table 12.3-4—Airborne Radioactivity Detector Parameters (Sheet 2 of 5)

Monitor Location	Monitor Provisions		Range[1]
	In-Process Continuous	ACF	
Fuel Building (Figure 12.3-73) (continued)	1 gaseous iodine monitor (R-7) in exhaust air of containment ventilation (upstream KLA2 filters)	---	5E-4 – 3E+0 μCi 3E-10 – 5E-8 μCi/cc (I-131) Must be capable of detecting 10 DAC-hours
	1 tritium monitor (R-8) in exhaust air of containment ventilation (upstream KLA2 filters)	---	3E-9 – 3E-4 μCi/cc (H-3)
	2 noble gas monitors (R-19) air leaving fuel handling area adjacent to monitored air duct (Fuel handling area)	~~Initiates automatic isolation of the Fuel Handling Area~~Refer to Table 11.5-1, Monitor R-19	1E-5 – 1E+0 rad/hr Must be capable of detecting 10 DAC-hours
	2 noble gas monitors (R-9) in air leaving containment (next to air duct) downstream KLA2 low flow purge exhaust	Refer to Table 11.5-1, Monitor R-9	1E-5 – 1E+0 rad/hr
	2 noble gas accident monitors (R-27) in exhaust air from exhaust cell (downstream KLB accident exhaust filter)	Refer to Table 11.5-1, Monitor R-27	1E-4 – 1E+4 rad/hr
	2 noble gas accident monitors (R-26) in exhaust air from exhaust cell (downstream KLC accident exhaust filter)	Referto Table 11.5-1, Monitor R-26	1E-4 – 1E+4 rad/hr

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Table 12.3-4—Airborne Radioactivity Detector Parameters (Sheet 3 of 5)

Monitor Location	Monitor Provisions		Range[1]
	In-Process Continuous	ACF	
Safeguard Building (Figure 12.3-72)	4 monitors (R-29 and R-30) intake air of the MCR	~~Initiates automatic iodine filtration of the MCR ventilation inlet air~~Refer to Table 11.5-1, Footnote 19 for Monitors R-29 and R-30	1E-5 – 1E+1 rad/hr Must be capable of detecting 10 DAC-hours

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Table 12.3-4—Airborne Radioactivity Detector Parameters
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Monitor Location	Monitor Provisions		Range[1]
	In-Process Continuous	ACF	
Nuclear Auxiliary Building (Figure 12.3-74)	6 aerosol monitors in exhaust air from exhaust cells of Safeguard Building and Nuclear Auxiliary Building, Fuel Building (upstream KLE Filtration)	—See Table 11.5-1, Monitors R-11, R-12, and R-13 (NABVS Cells 1-3), R-25 (SBVS Cell 6), R-17 and R-18 (FBVS Cells 4 and 5), and R-19 (Fuel Handling Area Ventilation (Cell 5))	5E-4 – 3E+0 µCi 3E-10 – 1E-6 µCi/cc Must be capable of detecting 10 DAC-hours
	6 noble gas monitors in exhaust air from exhaust cells of Safeguard Building and Nuclear Auxiliary Building, Fuel Building (upstream KLE Filtration)	—See Table 11.5-1, Monitors R-11, R-12, and R-13 (NABVS Cells 1-3), R-25 (SBVS Cell 6), R-17 and R-18 (FBVS Cells 4 and 5), and R-19 (Fuel Handling Area Ventilation (Cell 5))	3E-7 – 1E-2 µCi/cc (Kr-85, Xe-133)
	6 gaseous iodine monitors in exhaust air from exhaust cells of Safeguard Building and Nuclear Auxiliary Building, Fuel Building (upstream KLE Filtration)	—See Table 11.5-1, Monitors R-11, R-12, and R-13 (NABVS Cells 1-3), R-25 (SBVS Cell 6), R-17 and R-18 (FBVS Cells 4 and 5), and R-19 (Fuel Handling Area Ventilation (Cell 5))	5E-4 – 3E+0 µCi 3E-10 – 5E-8 µCi/cc (I-131) Must be capable of detecting 10 DAC-hours
	1 aerosol monitor in laboratory exhaust air (KLE Laboratory Exhaust)	---	5E-4 – 3E+0 µCi 3E-10 – 1E-6 µCi/cc Must be capable of detecting 10 DAC-hours
	1 aerosol monitors in exhaust air of hot workshop (KLE Cell 3)	---	5E-4 – 3E+0 µCi 3E-10 – 1E-6 µCi/cc Must be capable of detecting 10 DAC-hours

Table 12.3-4—Airborne Radioactivity Detector Parameters (Sheet 5 of 5)

Monitor Location	Monitor Provisions		Range[1]
	In-Process Continuous	ACF	
	4 radiation monitors (R-46 through R-49); 1 radiation monitor for each steam generator blowdown line	See Table 11.5-1, Footnote 7, Radiation Monitors R-46 through R-49	3E-6 – 1E-2 ìCi/cc
Radioactive Waste Processing Building (Figure 12.3-74)	1 aerosol monitor (R-23) in exhaust air of decontamination room (KLF Rooms Cell 2)	---	5E-4 – 3E+0 μCi 3E-10 – 1E-6 μCi/cc Must be capable of detecting 10 DAC-hours
	1 aerosol monitor (R-24) in exhaust air of mechanical workshop (KLF System Rooms)	---	5E-4 – 3E+0 μCi 3E-10 – 1E-6 μCi/cc Must be capable of detecting 10 DAC-hours
	2 aerosol monitors (R-20 and R-22) in exhaust air from exhaust cells (Upstream KLF Room Exhaust Filters)	~~—~~See Table 11.5-1, Monitor R-20 (Cell 1) and R-22 (Cell 2)	5E-4 – 3E+0 μCi 3E-10 – 1E-6 μCi/cc Must be capable of detecting 10 DAC-hours
	2 gaseous iodine monitors (R-20 and R-22) in exhaust air from exhaust cells (Upstream KLF Room Exhaust Filters)	~~—~~See Table 11.5-1, Monitor R-20 (Cell 1) and R-22 (Cell 2)	5E-4 – 3E+0 μCi 3E-10 – ~~1E-6~~5E-8 μCi/cc (I-131) Must be capable of detecting 10 DAC-hours

Note:

1. Only particulate and iodine monitors are required to detect 10 DAC-hrs (see Section 12.3.4.2.1).

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3.7.4 The blowdown system meets design requirements in response to a partial cooldown signal coupled with either high secondary activity (refer to Table 11.5-1, Monitors R-46 through R-49) or high SG level.

3.7.5 The blowdown system meets design requirements in response to a safety injection signal plus loss of offsite power.

3.8 Verify SG wet layup system operations.

3.9 Check electrical independence and redundancy of power supplies for safety-related functions by selectively removing power and determining loss of function.

4.0 DATA REQUIRED

4.1 Valve performance data, where required.

4.2 Valve position indication.

4.3 Position response of valves to loss of motive power.

4.4 Setpoints at which alarms and interlocks occur.

4.5 Response of CIVs to CIS and signals described in Section 10.4.8.2.2.

4.6 SG blowdown flow path flow rates.

5.0 ACCEPTANCE CRITERIA

5.1 The SGBS operates as designed (refer to Section 10.4.8):

5.1.1 SGBS alarms, interlocks, protective devices, and controls (manual and automatic) respond as required.

5.1.2 SGBS instrumentation performs as designed.

5.1.3 SGBS valves perform as designed (i.e., thrust, opening times, closing times, ability to initiate and terminate SGBS flow without introducing water hammers).

5.1.4 SGBS responds as designed to isolation signals.

5.1.5 SGBS flow rates meet design requirements.

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5.2 Verify that safety-related components meet electrical independence and redundancy requirements.

5.3 The SGBS meet design requirements to monitor radiation (refer to Table 11.5-1, Monitors R-46 through R-49).

14.2.12.7.10 Steam Turbine (Test #068)

1.0 OBJECTIVE

1.1 To demonstrate functional performance of the steam turbine controls.

1.2 To demonstrate functional performance of the steam turbine support system.

2.2 The secondary sampling system has been flushed with demineralized water to remove residues of chemical agents used during construction cleaning phases.

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2.3 Systems being sampled are at or near normal operating pressure and temperature.

2.4 Calibrating gases and solutions are available for radioactive and non-radioactive analyses as referenced in Table 9.3.2-2 and Table 11.5-1, Monitors R-46 through R-49 and R-50.

2.5 Test instrumentation is available and calibrated.

2.6 SECSS instrumentation has been calibrated and is functional for performance of the following test.

3.0 TEST METHOD

3.1 Withdraw fluid at each sample point, verifying adequate sample flow.

3.2 Verify that operation of alarms and interlocks meets design requirements.

3.3 Verify that operation of pump and heat exchangers in normal operation using normal flow paths meets design requirements.

3.4 Verify the analytical instrumentation provides indication and response that meet the design requirements.

3.5 Activate power-operated valves remotely while:

a. Observing each valve operation and position indication.

b. Measuring valve performance data (e.g., thrust, opening and closing times).

3.6 Verify power-operated valves fail upon loss of motive power as designed (refer to Section 9.3.2).

3.7 Verify that continuous monitors and sample flow rate meets design requirements.

3.8 Check electrical independence and redundancy of power supplies for safety-related functions by selectively removing power and determining loss of function.

4.0 DATA REQUIRED

4.1 Setpoints at which alarms and interlocks occur.

4.2 Sampling flow rate from each sample point.

4.3 Analytical instrument data.

4.4 Valve performance data, where required.

4.5 Valve position indication.

4.6 Position response of valves to loss of motive power.

5.0 ACCEPTANCE CRITERIA

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5.1 The SECSS meets design requirements (refer to Section 9.3.2 and Table 11.5-1, Monitors R-46 through R-49 and R-50):

5.1.1 SECSS alarms, interlocks, and controls (manual and automatic) function as designed.

5.1.2 SECSS valves perform as designed (i.e., opening times, closing times, and pressure/temperature controls).

5.1.3 SECSS meet design requirements for representative samples.

5.1.4 Continuous ~~and chemistry lab~~ instrumentation used to analyze SECSS parameters described in Table 9.3.2-2 and Table 11.5-1, Monitors R-46 through R-49 and R-50 will meet the design requirements for the measurements. This includes, but is not limited to, the following (that could adversely impact the ability to accurately measure the parameters described in Table 9.3.2-2 and Table 11.5-1, Monitors R-46 through R-49 and R-50):

- Range.
- Response time.
- Sensitivity.
- Maximum anticipated drift between calibrations.

5.2 Verify that safety-related components meet electrical independence and redundancy requirements.

14.2.12.7.14 Steam Generator Blowdown Demineralizing System (Test #072)

1.0 OBJECTIVE

1.1 To verify the ability of the steam generator blowdown (SGB) demineralizing system to clean the SG blowdown by a combination of filtration and ion exchange.

2.0 PREREQUISITES

2.1 Construction activities on the systems to be tested have been completed.

2.2 Systems being sampled are at or near normal operating pressure and temperature.

2.3 Calibrating gases and solutions are available for radioactive and non-radioactive analyses as referenced in Table 9.3.2-2.

2.4 Test instrumentation is available and calibrated.

2.5 Secondary sampling system instrumentation has been calibrated and is functional for performance of the following test.

2.6 Verify that steam generator blowdown system demineralizers are loaded with the proper type and amount of ion exchange resins.

3.8 Verify that operation of the equipment compartment ventilation units meets design requirements.

3.9 Verify HEPA filter efficiency, carbon adsorber efficiency, and air flow capacity.

3.10 Verify the system rated air flow and air balance.

3.11 Verify that operation of protective devices, controls, interlocks instrumentation, and alarms using actual or simulated inputs meet design requirements.

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3.12 Verify that operation of the ~~reactor containment building cooling and ventilation system to associated~~CBVS radiation monitors meets design requirements (refer to Table 11.5-1, Monitors R-7 through R-10).

4.0 DATA REQUIRED

4.1 Air balancing verification.

4.2 Fan and damper operating data.

4.3 Temperature data of building areas.

4.4 Setpoints of alarms, interlocks, and controls

4.5 Reactor Containment Building negative pressurization data.

4.6 HEPA filter and carbon adsorber data.

4.7 CBVS performance data in response to radiation monitors.

5.0 ACCEPTANCE CRITERIA

5.1 The CBVS operate as designed (refer to Section 9.4.7):

5.1.1 CBVS alarms, interlocks, protective devices, and controls (manual and automatic) function as designed.

5.1.2 CBVS fan performance meets design requirements.

5.1.3 CBVS dampers/valve performance (i.e., thrust, opening times, closing times, and ability to control flow) meets design requirements.

5.2 The CBVS meets design requirements to monitor radiation (refer to ~~Section 7.3.1~~Table 11.5-1, Monitors R-7 through R-10).

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14.2.12.8.4 Containment Purge (Test #076)

1.0 OBJECTIVE

1.1 To demonstrate the capability of the containment purge systems, both low-flow and full-flow, to maintain the containment air quality and cleanliness at the required value during normal operation (low-flow), inspection, testing, maintenance, and refueling operations.

1.2 To demonstrate electrical independence and redundancy of power supplies.

1.3 Demonstrate containment purge system response to protection system (PS) signals.

2.0 PREREQUISITES

2.1 Construction activities in the containment have been completed and acceptable levels of cleanliness established.

2.2 Construction activities on the containment purge systems have been completed.

2.3 Containment purge system instrumentation has been calibrated and is operating satisfactorily prior to performing the following test.

2.4 Support systems required for operation of the containment purge systems are complete and functional.

2.5 Test instrumentation is available and calibrated.

3.0 TEST METHOD

3.1 Demonstrate manual and automatic system controls.

3.2 Verify alarms, indicating instruments and status lights are functional.

3.3 Verify design air flows for high purge, low purge.

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3.4 Perform HEPA filters and carbon adsorber efficiency tests.

3.5 Demonstrate system responses to a high radiation signal to perform automatic control functions (refer to Section 11.5.3.1.4 and Table 11.5-1, Monitor R-9) and high relative humidity signal.

3.6 Operate control valves remotely while:

a. Observing each valve operation and position indication.

b. Measuring valve performance data (e.g., thrust, opening and closing times).

3.7 Verify power-operated valves fail upon loss of motive power as designed (refer to Section 9.4.7).

3.8 Simulate the following and observe isolation valve response:

3.8.1 CIAS.

3.8.2 High humidity actuation signal.

3.8.3 High radiation actuation signal.

3.9 Verify that operation of containment purge system radiation monitors meets design requirements (refer to Table 11.5-1, Monitor R-7 through R-9).

3.10 Check electrical independence and redundancy of power supplies for safety-related functions by selectively removing power and determining loss of function.

3.11 Verify that the containment purge system functions as designed.

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3.12 Verify containment purge isolation signal from PS upon detection of containment high activity at radiation monitor (refer to Table 11.5-1, Monitor R-9).

4.0 DATA REQUIRED

4.1 Air balancing report, including fan operating data for low purge and high purge fans.

4.2 HEPA filter and carbon adsorber data for exhaust filter trains.

4.3 Valve performance data, where required.

4.4 Valve position indication.

4.5 Position response of valves to loss of motive power.

4.6 Setpoints at which alarms and interlocks occur.

4.7 Temperature of air supply (outside) to high purge supply and discharge into containment.

4.8 Valves respond to the following simulated signals:

4.8.1 CIS.

4.8.2 High humidity actuation signal.

4.8.3 High radiation actuation signal.

4.9 Containment purge system radiation monitors performance data.

5.0 ACCEPTANCE CRITERIA

5.1 The containment purge system meets design requirements (refer to Section 9.4.7):

5.1.1 The containment purge alarms, remote indications, interlocks, and controls (manual and automatic) respond as designed.

5.1.2 The containment purge valves meet the design requirements (i.e., thrust opening speed, closing speed, failure mode upon loss of motive power).

- Table 14.3-2 Item 2-10.

5.1.3 The containment purge flow rate meets design requirements.

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5.2 The containment purge system responds to the radiation monitors as designed (refer to ~~Section 7.3.1~~Section 11.5.3.1.4 and Table 11.5-1, Monitor R-9):

5.2.1 The containment purge system isolates purge flow as designed upon detection of high activity

5.2.2 The radiation monitors perform as designed in response to high activity levels.

5.3 Verify that safety-related components meet electrical independence and redundancy requirements.

4.2 Valve and damper operating data.

4.3 Air balancing report, including fan operating data.

4.4 HEPA filter and carbon adsorber efficiency data.

4.5 Annulus negative pressurization data: Annulus pressure and drawdown time response curve.

5.0 ACCEPTANCE CRITERIA

5.1 The AVS operates as designed (refer to Section 6.2.3):

5.1.1 Verify that the response of alarms, interlocks, and control logic meets design requirements.

5.1.2 Verify that operation of valves and dampers meet design requirements.

5.1.3 Verify that system response to simulated accident signal meets design requirements.

- Table 14.3-2 Item 2-1.

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5.2 Verify that safety-related components meet electrical independence and redundancy requirements.

5.3 The AVS meet design requirements to monitor radiation (refer to Table 11.5-1, Monitor R-27).

14.2.12.8.6 Electrical Division of Safeguard Building Ventilation System (Test #078)

1.0 OBJECTIVE

1.1 To demonstrate the operation of the electrical division of safeguard building ventilation system (SBVSE):

1.1.1 Vital instrument and equipment room ventilation subsystems.

1.1.2 Electrical and mechanical equipment room air handling units, recirculation fans, battery rooms/safety chilled water room exhaust fans.

1.1.3 Component cooling water/heat exchanger rooms fan coil units.

1.1.4 Emergency feedwater pump rooms fan coil units.

1.2 To demonstrate electrical independence and redundancy of power supplies.

2.0 PREREQUISITES

2.1 Construction activities in the Safeguard Building controlled area are complete with penetrations sealed.

2.2 Construction activities on the SBVSE have been completed.

2.3 SBVSE instrumentation has been calibrated and is operating satisfactorily prior to performing the following test.

2.4 Support systems required for operation of the SBVSE are functional.

14.2.12.8.7 Nuclear Auxiliary Building Ventilation System (Test #079)

1.0 OBJECTIVE

1.1 To demonstrate the operation of the nuclear auxiliary building ventilation system (NABVS).

1.2 To demonstrate electrical independence and redundancy of power supplies.

2.0 PREREQUISITES

2.1 Construction activities in the nuclear auxiliary building are complete with penetrations sealed.

2.2 Construction activities on the NABVS have been completed.

2.3 NABVS instrumentation has been calibrated and is operating satisfactorily prior to performing the following test.

2.4 Support systems required for operation of the NABVS are functional.

2.5 Test instrumentation is available and calibrated.

3.0 TEST METHOD

3.1 Verify control logic.

3.2 Verify the operation of the air handling units or fans or both.

3.3 Verify alarms, indicating lights and status lights are functional.

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3.4 Perform air flow balancing of the NABVS.

3.5 Verify that operation of dampers meets design requirements.

3.6 Perform HEPA filter and carbon adsorber efficiency tests.

3.7 Verify operation of the ~~vent stack~~NABVS radioactivity monitors (refer to Table 11.5-1, Monitors R-11 through R-15).

3.8 Check electrical independence and redundancy of power supplies for safety-related functions by selectively removing power and determining loss of function.

4.0 DATA REQUIRED

4.1 Damper operating data.

4.2 Air flow and balancing verification.

4.3 Setpoints at which alarms and control occur.

4.4 Temperature data for each of the NABVS.

4.5 HEPA filter and carbon adsorber efficiency data.

5.0 ACCEPTANCE CRITERIA

5.1 The NABVS operates as designed (refer to Section 9.4.3):

5.1.1 NABVS alarms, interlocks, protective devices, and controls (manual and automatic) function as designed.

5.1.2 NABVS fan performance meets design requirements.

5.1.3 NABVS dampers/valve performance (i.e., thrust, opening times, closing times, and ability to control flow) meets design requirements.

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5.1.4 NABVS air balance meets design requirements.

5.1.5 The NABVS meets design requirements to monitor radiation (refer to Table 11.5-1, Monitors R-11 through R-15).

14.2.12.8.8 Radioactive Waste Building Ventilation System (Test #080)

1.0 OBJECTIVE

1.1 To demonstrate the proper operation of the radioactive waste building ventilation system (RWBVS) to maintain design condition.

2.0 PREREQUISITES

2.1 Construction activities on the RWBVS have been completed.

2.2 RWBVS instrumentation has been calibrated and is operating satisfactorily prior to performing the following test.

2.3 Support systems required for operation of the RWBVS are complete and functional.

2.4 Test instrumentation is available and calibrated.

3.0 TEST METHOD

3.1 Verify control logic.

3.2 Verify that operation, stroking speed and position indication of dampers meets design requirements.

3.3 Verify the capacity of the HVAC system to maintain the area temperature.

3.4 Verify the system maintains the Radioactive Waste Processing Building at a negative pressure.

3.5 Verify that operation of the general ventilation supply units and fans meets design requirements.

3.6 Verify that operation of the general ventilation exhaust units and fans meets design requirements.

3.7 Perform HEPA filter and carbon adsorber efficiency tests.

3.8 Verify the systems rated air flow and air balance.

3.9 Verify that operation of protective devices, controls, interlocks instrumentation and alarms using actual or simulated inputs meets design requirements.

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3.10 Verify that operation of the RWBVS response to high radiation monitor signal meets design requirements (refer to Table 11.5-1, Monitors R-20 and R-22).

4.0 DATA REQUIRED

4.1 Air balancing verification.

4.2 Fan and damper operating data.

4.3 Temperature data.

4.4 Setpoints of alarms interlocks and controls.

4.5 The Radioactive Waste Building negative pressure readings.

4.6 RWBVS performance data in response to radiation monitor signals.

4.7 HEPA filter and carbon adsorber efficiency data.

5.0 ACCEPTANCE CRITERIA

5.1 The RWBVS operates as designed (refer to Section 9.4.8):

5.1.1 RWBVS alarms, interlocks, protective devices, and controls (manual and automatic) function as designed.

5.1.2 RWBVS fan performance meets design requirements.

5.1.3 RWBVS dampers/valve performance (i.e., thrust, opening times, closing times, and ability to control flow) meets design requirements.

5.1.4 RWBVS air balance meets design requirements.

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5.2 The RWBVS responds as designed to radiation monitor signals designed (refer to ~~Section 9.4.8~~Table 11.5-1, Monitors R-20 through R-22).

14.2.12.8.9 Fuel Building Ventilation System (Test #081)

1.0 OBJECTIVE

1.1 To demonstrate the proper operation of the fuel building ventilation system (FBVS) to maintain design conditions.

1.2 To demonstrate electrical independence and redundancy of power supplies.

2.0 PREREQUISITES

2.1 Construction activities on the FBVS have been completed.

2.2 FBVS instrumentation has been calibrated and is operating satisfactorily prior to performing the following test.

2.3 Support systems required for operation of the FBVS are complete and functional.

2.4 Test instrumentation is available and calibrated.

3.0 TEST METHOD

3.1 Verify control logic.

3.2 Verify that operation, stroke speed and position indication of dampers meet design requirements.

3.3 Verify the system maintains the Fuel Building at a negative pressure.

3.4 Verify the NABVS supplies and exhausts air to the Fuel Building.

3.5 Verify that the operation of the fuel handling area ventilation exhaust units and fans meet design requirements.

3.6 Verify that operation of the heating and cooling units meet design requirements.

3.7 Verify HEPA filter efficiency, carbon adsorber efficiency, and air flow capacity.

3.8 Verify the systems rated air flow and air balance.

3.9 Verify that operation of protective devices, controls, interlocks instrumentation, and alarms using actual or simulated inputs.

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3.10 Verify system response to a high radiation signal (refer to Table 11.5-1, Monitors R-17, R-18, and R-19).

3.11 Verify that operation of the FBVS radiation monitors meets design requirements (refer to Table 11.5-1, Monitors R-17, R-18, and R-19).

3.12 Check electrical independence and redundancy of power supplies for safety-related functions by selectively removing power and determining loss of function.

4.0 DATA REQUIRED

4.1 Air balancing verification.

4.2 Fan and damper operating data.

4.3 Temperature data in the Fuel Building.

4.4 Setpoints at which alarms, interlocks, and controls occur.

4.5 Fuel Building negative pressurization data during normal and postulated emergency conditions.

4.6 Filter and carbon adsorber data.

4.7 FBVS performance data in response to radiation monitor signals.

5.0 ACCEPTANCE CRITERIA

5.1 The FBVS operates as designed (refer to Section 9.4.2):

5.1.1 FBVS alarms, interlocks, and controls (manual and automatic) function as designed.

5.1.2 FBVS valves and dampers function as design.

5.1.3 FBVS maintains the Fuel Building at the required negative pressure.

- Table 14.3-2 Item 2-9.

5.1.4 FBVS recirculation rate (e.g., through the HEPA filters, carbon adsorber) meet design requirements.

5.1.5 FBVS normal operation heating and ventilation system performs as designed.

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5.2 The FBVS responds to radiation monitor signals as designed (refer to ~~Section 9.4.2~~Table 11.5-1, Monitors R-17, R-18, and R-19).

5.3 Verify that safety-related components meet electrical independence and redundancy requirements.

14.2.12.8.10 Main Control Room Air Conditioning System (Test #082)

1.0 OBJECTIVE

1.1 To verify that operation of the main control air conditioning system (CRACS) establishes that a proper environment for personnel and equipment under postulated conditions in the following areas:

1.1.1 MCR.

1.1.2 Technical Support Center.

1.1.3 Other offices and equipment areas of the control room envelope (CRE).

1.2 To demonstrate electrical independence and redundancy of power supplies.

2.0 PREREQUISITES

2.1 Construction activities in the MCR complex have been completed and penetrations sealed.

2.2 Construction activities on the CRACS have been completed.

2.3 The CRACS system instrumentation has been calibrated and is operating satisfactorily prior to performing the following test.

2.4 Support systems required for operation of the CRACS are complete and functional.

2.5 Test instrumentation is available and calibrated.

3.0 TEST METHOD

3.1 Verify control logic.

3.2 Verify that operation, stroke speed and position indication of dampers meet design requirements.

3.3 Verify in manual operating mode that system rated air flow and air balance meet design requirements.

3.4 Demonstrate in automatic mode the transfer to emergency-operations as a result of the following:

3.4.1 Detection of radiation in one of the outside inlets places the CREF (iodine filtration) units in the filtered alignment.

3.4.2 ~~Detection of toxic gas (which includes CO, or CO2) in one of the outside air inlets will close inlet isolation dampers at that inlet.~~

3.4.3 Safety injection actuation/primary containment isolation signal.

3.5 Verify the HEPA filter efficiency, carbon adsorber efficiency, and filter bank air flow capacity.

3.6 Verify that operation of protective devices, controls, interlocks, instrumentation, and alarms using actual or simulated inputs meets design requirements.

3.7 Verify that the system maintains the CRE at the required positive pressure relative to the outside atmosphere during system operation.

3.8 ~~Verify the isolation capability of the CRE on detection of toxic gas at the intakes meets the requirements of RG 1.78.~~

3.9 Demonstrate the operation of the battery room exhaust fans.

3.10 Verify the CRE air in-leakage rate when aligned in the emergency mode.

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3.11 Verify that operation of CRACS in response to radiation monitors meets design requirements (refer to Table 11.5-1, Monitors R-29 and R-30).

3.12 Check electrical independence and redundancy of power supplies for safety-related functions by selectively removing power and determining loss of function.

4.0 DATA REQUIRED

4.1 Air balancing verification.

4.2 Fan and damper operating data.

4.3 Temperature data in the CRE.

4.4 Response to radioactivity~~, toxic gas (including CO, or CO_2),~~ and smoke.

4.5 Setpoints of alarms, interlocks, and controls.

4.6 Pressurization data for the CRE.

4.7 Filter and carbon adsorber data.

4.8 CRE in-leakage rate when aligned in the emergency mode.

4.9 The CRACS response to radiation monitors.

5.0 ACCEPTANCE CRITERIA

5.1 The CRACS operates as designed (refer to Section 9.4.1).

5.1.1 CRACS alarms, interlocks, and controls (manual and automatic) function as designed.

5.1.2 CRACS valves and dampers function as design.

5.1.3 CRACS responds as designed to a simulated ~~toxic gas (including CO, or CO_2) and~~ smoke.signal.

5.1.4 CRACS recirculation flow rate meets design requirements.

- Table 14.3-2 Item 2-7.

5.1.5 CRACS unfiltered air in-leakage rate while in recirculation mode meets design requirements.

- Table 14.3-2 Item 2-8.

5.1.6 CRACS is capable of generating a positive MCR pressure relative to adjacent areas, as designed.

- Table 14.3-2 Item 2-6.

5.1.7 CRACS responds as designed to a simulated SIS signal.

- Table 14.3-2 Item 2-5.

5.2 The CRACS radiation monitors perform as designed (refer to ~~Section 9.4.1~~Table 11.5-1, Monitors R-29 and R-30):

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5.2.1 CRACS responds as designed to a simulated high radiation signal.

- Table 11.5-1, Monitors R-29 and R-30.
- Table 14.3-2 Item 2-5.

5.3 Verify that safety-related components meet electrical independence and redundancy requirements.

14.2.12.8.11 Safeguard Building Controlled Area Ventilation System (Test #083)

1.0 OBJECTIVE

1.1 To demonstrate the operation of the safeguard building controlled area ventilation system (SBVS):

1.1.1 Hot mechanical area serviced by the SBVS.

1.1.2 SBVS air supply subsystem.

1.1.3 SBVS air exhaust subsystem.

1.1.4 Electric air heating convectors (area heaters).

1.2 To demonstrate electrical independence and redundancy of power supplies.

2.0 PREREQUISITES

2.1 Construction activities in the safeguard building mechanical area are complete with penetrations sealed.

2.2 Construction activities on the SBVS have been completed.

2.3 Safeguard building mechanical area ventilation subsystem instrumentation has been calibrated and is operating satisfactorily prior to performing the following test.

2.4 Support systems required for operation of the SBVS are functional.

2.5 Test instrumentation is available and calibrated.

3.0 TEST METHOD

3.1 Verify control logic.

3.2 Verify the operation of air handling units or fans or both.

3.3 Verify operation of the operational air exhaust mode in the mechanical area.

3.4 Verify operation of the accident air exhaust mode in the mechanical area.

3.5 Verify operation of the electric air convectors (area heaters).

3.6 Verify operation of the filter air heaters, prefilters, HEPA filters, and adsorbers.

3.7 Verify operation of the recirculation cooling units.

3.8 Verify alarms, indicating lights and status lights are functional.

3.9 Perform air flow balancing of the SBVS.

3.10 Verify that operation of dampers meet design requirements.

3.11 Check electrical independence and redundancy of power supplies for safety-related functions by selectively removing power and determining loss of function.

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3.12 Verify that operation of the SBVS radiation monitors meet design requirements (refer to Table 11.5-1, Monitors R-25 and R-26).

4.0 DATA REQUIRED

4.1 Damper operating data.

4.2 Air flow and balancing verification.

4.3 Setpoints at which alarms, center backs and control occur.

4.4 Temperature data for each of the SBVS.

5.0 ACCEPTANCE CRITERIA

5.1 The SBVS operates as designed (refer to Section 9.4.5):

5.1.1 SBVS air handlers/fans perform as designed.

5.1.2 The operation of the SBVS operational air exhaust mode in the mechanical area meets design requirements.

5.1.3 The operation of the SBVS accident air exhaust mode in the mechanical area meets design requirements.

5.1.4 The operation of the SBVS electric air convectors (area heaters) meets design requirements.

5.1.5 The operation of the SBVS filter air heaters, prefilters, HEPA filters, and adsorber meets design requirements.

5.1.6 The operation of the SBVS recirculation cooling units meets design requirements.

5.1.7 SBVS alarms, indicating lights and status lights meet design requirements.

5.2 Verify that safety-related components meet electrical independence and redundancy requirements.

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5.3 The SBVS meets design requirements to monitor radiation (refer to Table 11.5-1, Monitors R-25 and R-26).

14.2.12.8.12 Emergency Power Generating Building Ventilation System (Test #084)

1.0 OBJECTIVE

1.1 To demonstrate proper operation of the emergency power generating building ventilation system (EPGBVS).

1.2 To demonstrate proper operation of the EPGBVS.

1.3 To demonstrate electrical independence and redundancy of power supplies.

2.0 PREREQUISITES

2.1 Construction activities on the EPGBVS have been completed.

2.2 EPGBVS instrumentation has been calibrated and is operating satisfactorily prior to performing the following test.

2.3 Support systems required for operation of the EPGBVS are complete and functional.

2.4 Test instrumentation is available and calibrated.

3.0 TEST METHOD

3.1 Verify control logic.

3.2 Verify design air flow with each EPGBVS in operation.

3.3 Verify design temperature can be maintained in each Emergency Power Generating Building.

3.4 Verify alarms, indicating instruments, and status lights are functional.

be performed by pouring a liquid down the drain colored with food dye or by some other suitable means, and confirm the presence of the food dye in the receiving tank.

3.3 Confirm that ventilation hoods and other engineered radioactive containment devices are vented as designed. This could be accomplished by tracer gas or some other suitable means.

3.4 Measure the ventilation hood discharge flow rates for engineered devices.

3.5 Perform vendor supplied startup checks and calibrations for all laboratory equipment that analyze or measure radiation levels.

3.6 Perform vendor supplied startup checks and calibrations for all laboratory equipment that analyze or measure isotopic concentrations of radioactive samples.

4.0 DATA REQUIRED

4.1 Inspection report from verification of laboratory equipment drains.

4.2 Inspection report from verification of ventilation hood flow and routing.

4.3 Completed vendor specified laboratory equipment startup procedures.

5.0 ACCEPTANCE CRITERIA

5.1 The laboratory equipment drain interface with the plant systems performs as designed.

5.2 The laboratory equipment ventilation hood interface with the plant systems performs as designed.

5.3 The laboratory equipment checkout and calibration procedures meet design requirements as described in Sections 11.5 and 13.4.

14.2.12.8.19 Access Building Ventilation System (Test #224)

1.0 OBJECTIVE

1.1 To verify the access building ventilation system (ABVS) can maintain the space temperature as required.

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1.2 To verify that radiological alarms are provided in the control room for the operator to manually isolate the building exhaust to the vent stack (refer to Section 11.5.3.1.2 and Table 11.5-1, Monitor R-31.~~control features route the building exhaust to the vent stack~~.

2.0 PREREQUISITES

2.1 Construction activities on the ABVS have been completed.

2.2 ABVS instrumentation has been calibrated and is operating satisfactorily prior to performing the following test.

2.3 Support systems required for operation of the ABVS are complete and functional.

2.4 Test Instrumentation is available and calibrated.

3.0 TEST METHOD

3.1 Verify control logic and interlock.

3.2 Verify design air flow of each fan.

3.3 Verify alarms, indicating instruments and status lights are functional.

3.4 Verify design temperatures can be maintained in the structure.

4.0 DATA REQUIRED

4.1 Temperature data for the structure from each HVAC unit.

4.2 HVAC unit operating data.

4.3 Setpoints at which alarms and interlocks occur.

5.0 ACCEPTANCE CRITERIA

RAI 273, Q. 11.05-2

5.1 The ABVS operates as designed (refer to Section 9.4.14, Section 11.5.3.1.10 and Table 11.5-1, Monitor R-31):

5.1.1 ABVS alarms, interlocks, protective devices, and controls (manual and automatic) function as designed.

5.1.2 ABVS fan performance meets design requirements.

5.1.3 ABVS dampers/valve performance (i.e., thrust, opening times, closing times, and ability to control flow) meets design requirements.

5.1.4 ABVS air balance meets design requirements.

14.2.12.9 Auxiliary Systems

14.2.12.9.1 Leak-off System (Test #091)

1.0 OBJECTIVE

1.1 To demonstrate the functionality of the leak-off system (LOS) to collect and route bypass leakage from selected penetrations in the Containment Building containing primary fluid to the annulus sump.

1.2 To demonstrate the functionality of the LOS to provide a flow path for inflating/deflating the containment in support of ILRT.

1.3 To demonstrate the functionality of the LOS to provide a flow path for measuring leak tightness of the containment in support of ILRT.

2.0 PREREQUISITES

2.1 Construction activities on the LOS have been completed.

3.6 Verify that operation of the solid waste processing system radiation monitors meet design requirements (refer to Table 11.5-1, Monitor R-43).

4.0 DATA REQUIRED

4.1 Setpoints at which alarms and interlocks occur.

4.2 Solid waste transfer system operating data.

4.3 Radioactive waste processing building crane data.

4.4 System flow path data.

5.0 ACCEPTANCE CRITERIA

5.1 The solid waste management system operates as designed (refer to Section 11.4).

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5.2 The concrete shielding associated with the solid waste management system meets design requirements.

5.3 The radiation monitors meet design requirements (refer to Table 11.5-1, Monitor R-43).

14.2.12.9.4 Radioactive Concentrates Processing System - Solid Waste (Test #094)

1.0 OBJECTIVE

1.1 To verify the performance of the radioactive concentrates processing system.

2.0 PREREQUISITES

2.1 Construction activities have been completed on the radioactive concentrates processing system.

2.2 Support systems required for operation of the radioactive concentrates processing evaporator are complete and functional.

2.3 Radioactive concentrates processing system instrumentation has been calibrated and is operating satisfactorily prior to performing the following test.

3.0 TEST METHOD

3.1 Radioactive concentrates processing system components (wet solid wastes) function as designed.

3.1.1 Vacuum unit.

3.1.2 High pressure cleaning device.

3.1.3 Condensate collection pump.

3.1.4 Resin proportioning tank.

3.1.5 Concentrate buffer tank.

3.1.6 Condensate collection tank.

3.1.7 Scrubber tank.

3.1.8 Resin traps.

3.1.9 Condenser drying unit.

3.1.10 Condensate counter.

3.1.11 Condensate buffer sluice.

3.1.12 Transfer station.

3.1.13 Measuring glass.

3.1.14 Drum drying stations.

3.1.15 Drum transfer device.

3.1.16 High integrity container.

3.1.17 Sampling box.

3.1.18 Drum capping device.

3.1.19 Sampling device for dried waste.

3.1.20 Drum handling device.

3.1.21 Drum measuring device.

3.2 Initiate a high radiation signal to the appropriate radiation monitors (refer to Table 11.5-1, Monitor R-43) to verify that system response (control and alarm actuations) meet design requirements. The source of initiation of the signal, listed in order of preference, should be one of the following:

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3.2.1 Internal check source (verify that check source strength is capable of generating desired actuations).

3.2.2 Radiation calibration check source (verify that check source does not generate a personnel hazard during the test).

3.2.3 Simulated high radiation signal at the radiation detector (refer to Table 11.5-1, Monitor R-43).

3.3 Line up the radioactive concentrates processing system to interfacing systems and, using appropriate operating modes and indications, establish flow paths to these systems.

3.4 Verify that expended resin beds from the LWPS can be sluiced to the radioactive concentrates processing system.

4.0 DATA REQUIRED

4.1 Valve position indication.

4.2 Radioactive concentrates processing system response to simulated interlocks.

4.3 Setpoints at which alarms interlock and automatic actuations occur.

4.4 Flow indications.

5.0 ACCEPTANCE CRITERIA

5.1 The radioactive concentrates processing system performs as described in Sections 11.2.2, 11.4, and 11.5.

5.2 Radiation monitoring instrumentation used to monitor the radioactive concentrations activity prior to entry of the solid radwaste system that is described in Table 11.5-1 will meet the design requirements for the radiation Mmonitor R-43. This includes, but is not limited to, the following (that could adversely impact the ability to measure the parameters described in Table 11.5-1, Monitor R-43):

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5.2.1 Range.

5.2.2 Response time.

5.2.3 Sensitivity.

5.2.4 Maximum anticipated drift between calibrations.

14.2.12.9.5 Liquid Waste Processing System (Test #095)

1.0 OBJECTIVE

1.1 To demonstrate the functionality of the liquid waste processing system (LWPS) for collection, processing and recycling of liquid wastes and for preparation of liquid waste for release to the environment.

2.0 PREREQUISITES

2.1 Construction activities on the liquid waste processing system have been completed.

2.2 Verify that the liquid waste processing system's demineralizers and ultra filtration are loaded with the proper types and amounts of ion exchange resins and filtration media.

2.3 Liquid waste processing system instrumentation has been calibrated and is operating satisfactorily prior to performing the following test.

2.4 Support systems required for operation of the liquid waste processing system are completed and functional.

2.5 Test instrumentation is available and calibrated.

3.0 TEST METHOD

3.1 Evaporator system components function as designed.

3.1.1 Evaporator feed pumps.

3.1.2 Pre-heater.

3.1.3 Forced recirculation pump.

3.1.4 Evaporator.

3.1.5 Evaporator column.

3.1.6 Vapor compressor.

3.1.7 Distillate tank.

3.1.8 Distillate pump.

3.1.9 Distillate cooler.

3.1.10 Compressor injection cooler.

3.1.11 Electric heater.

3.1.12 Vent gas cooler.

3.1.13 Control valves.

3.1.14 Sealing liquids.

3.2 Centrifuge system components function as designed.

3.2.1 Centrifuge feed pump.

3.2.2 Decanter.

3.2.3 Separator.

3.2.4 Sludge tank.

3.2.5 Decanter feed pump.

3.2.6 Control valves.

3.3 Demineralizer system components function as designed.

3.3.1 Prefilter.

3.3.2 Demineralizer.

3.3.3 Ultrafilter.

3.3.4 Spent resin dryer.

3.3.5 Resin trap.

3.3.6 Solids collection.

3.3.7 Demineralizer booster pump.

3.4 Operate radioactive concentrates processing evaporator control valves from appropriate control positions:

a. Observing each valve operation and position indication.

b. Measuring valve performance data (e.g. thrust, opening and closing times).

3.5 Verify power-operated valves fail upon loss of motive power as designed (refer to Section 11.2).

3.6 Verify that operation of the tank level alarms and interlocks meets design requirements.

3.7 Verify that operation of system pumps meet design requirements.

3.8 Verify that operation of high differential pressure alarms for the process vessel meet design requirements.

3.9 Verify that operation of the tank mixers meet design requirements.

3.10 Initiate a high radiation signal to the liquid waste processing discharge radiation monitors (refer to Table 11.5-1, Monitor R-32) to verify that

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system response (control and alarm actuations) meet design requirements. The source of initiation of the signal, listed in order of preference, should be one of the following:

3.10.1 Internal check source (verify that check source strength is capable of generating desired control actuations).

3.10.2 Radiation calibration check source (verify that check source does not generate a personnel hazard during the test).

3.10.3 Simulated high radiation signal at the radiation detector.

3.11 Verify alarms, indicating instruments, and status lights are functional.

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3.12 Initiate a high radiation signal to the liquid waste processing discharge radiation monitors (refer to Table 11.5-1, Monitor R-32) to verify that system response (isolation actuations) meet design requirements. The source of initiation of the signal, listed in order of preference, should be one of the following:

3.12.1 Internal check source (verify that check source strength is capable of generating desired control actuations).

3.12.2 Radiation calibration check source (verify that check source does not generate a personnel hazard during the test).

3.12.3 Simulated high radiation signal at the radiation detector.

4.0 DATA REQUIRED

4.1 Waste pump operating data.

4.2 Valve performance data, where required.

4.3 Valve position indication.

4.4 Position response of valves to loss of motive power.

4.5 Setpoints at which alarms and interlocks occur.

5.0 ACCEPTANCE CRITERIA

5.1 The LWPS operates as designed (refer to Section 11.2).

5.2 The LWPS discharge radiation monitor operates as designed (refer to Sections 7.3.1, 11.2, and 11.5).

5.3 Radiation monitoring instrumentation used to monitor the liquid waste processing activity at the liquid radwaste release line that is described in Table 11.5-1 will meet the design requirements for the radiation monitor. This includes, but is not limited to, the following (that could adversely impact the ability to measure the parameters described in Table 11.5-1, Monitor R-32):

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5.3.1 Range.

5.3.2 Response time.

5.3.3 Sensitivity.

5.3.4 Maximum anticipated drift between calibrations.

14.2.12.9.9 Gaseous Waste Processing System (Test #099)

1.0 OBJECTIVE

1.1 To demonstrate the ability of the gaseous waste processing system (GWPS) to collect and process radioactive and potentially flammable gases vented from plant equipment.

2.0 PREREQUISITES

2.1 Construction activities on the GWPS have been completed.

2.2 The GWPS instrumentation has been calibrated and is operating satisfactorily prior to performing the following test.

2.3 Support systems required for operation of the GWPS are completed and functional.

2.4 Test instrumentation is available and calibrated.

2.5 Verify that the GWPS charcoal beds and gel driers are loaded with the proper types and amounts of charcoal and desiccant.

3.0 TEST METHOD

3.1 Verify flow paths.

3.2 Demonstrate that discharge isolation features and other system controls function as designed.

3.3 Verify alarms, indicating instruments and status lights are functional.

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3.4 Initiate a high radiation signal to the GWPS discharge radiation monitors (refer to Table 11.5-1, Monitors R-1 and R-2) to verify system response (control and alarm actuations) meet design requirements. The source of initiation of the signal, listed in order of preference, should be one of the following:

3.4.1 Internal check source (verify that check source strength is capable of generating desired actuations).

3.4.2 Radiation calibration check source (verify that check source does not generate a personnel hazard during the test).

3.4.3 Simulated high radiation signal at the radiation detector.

3.5 Verify that the GWPS discharge radiation monitor and alarm actuation function as designed.

3.6 Demonstrate the operation of the gas drying equipment.

3.7 Demonstrate that hold up time of gas through the charcoal adsorber meet design requirements.

3.8 Demonstrate the operation of the gel dryer regeneration equipment (protects delay beds from moisture while operating in surge mode).

3.9 Demonstrate the operation of the gas analyzers to detect concentrations of O_2 and H_2 specified in the plant Technical

Specifications, Section 5.5.11, "Gaseous Waste Processing System Radioactivity Monitoring Program."

3.10 Demonstrate the operation of the recombiner.

3.11 Operate control valves remotely while:

a. Observing each valve operation and position indication.

b. Measuring valve performance data (e.g., thrust, opening and closing times).

3.12 Verify power-operated valves fail upon loss of motive power as designed (refer to Section 11.3).

4.0 DATA REQUIRED

4.1 Setpoints of alarms, interlocks, and controls.

4.2 Gas dryer operating data.

4.3 Dryer regenerating equipment operating data.

4.4 Gas analyzer operating data.

4.5 Recombiner operating data.

4.6 Gas transport times.

4.7 Position response of valves to loss of motive power.

4.8 Valve performance data.

5.0 ACCEPTANCE CRITERIA

5.1 The GWPS operates as designed (refer to Sections 11.3 and 11.5).

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5.2 The GWPS discharge radiation monitor operates as designed (refer to Table 11.5-1~~Section 7.3.1~~).

5.3 Radiation monitoring instrumentation used to monitor the gaseous waste processing activity upstream of the decay beds that is described in Table 11.5-1 will meet the design requirements for the radiation monitor. This includes, but is not limited to, the following (that could adversely impact the ability to measure the parameters described in Table 11.5-1):

5.3.1 Range.

5.3.2 Response time.

5.3.3 Sensitivity.

5.3.4 Maximum anticipated drift between calibrations.

14.2.12.9.10 Nuclear Sampling System (Test #100)

1.0 OBJECTIVE

1.1 To verify the ability of nuclear sampling system (NSS) to collect and deliver representative samples of liquids and gases in various process

3.8 Verify that operation of continuous monitors and verify adequate flow meet design requirements.

3.9 Check electrical independence and redundancy of power supplies for NSS and SASS safety-related functions by selectively removing power and determining loss of function.

4.0 DATA REQUIRED

4.1 Setpoints at which alarms and interlocks occur.

4.2 Sampling flow rate from each sample point.

4.3 Analytical instrument data.

4.4 Valve performance data, where required.

4.5 Valve position indication.

4.6 Position response of valves to loss of motive power.

4.7 Calculated holdup time for RCS and pressurizer samples.

5.0 ACCEPTANCE CRITERIA

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5.1 The NSS meets design requirements (refer to Section 9.3.2.2.1.1, Section 11.5.4.6, and Table 11.5-1, Monitor R-41).

5.2 The SASS performs as described in Section 9.3.2.2.1.3.

5.3 Verify that NSS and SASS safety-related components meet electrical independence and redundancy requirements.

14.2.12.9.11 Station Blackout Diesel Generator Mechanical (Test #101)

1.0 OBJECTIVE

1.1 To demonstrate the station blackout diesel generator (SBODG) set system operates reliably.

2.0 PREREQUISITES

2.1 Construction activities on the SBODG system have been completed. This includes, but is not limited to the following:

2.1.1 SBODG fuel oil system (refer to Section 8.4.1).

2.1.2 SBODG engine lube oil system (refer to Section 8.4.1).

2.1.3 SBODG cooling system (refer to Section 8.4.1).

2.1.4 SBODG starting air system (refer to Section 8.4.1).

2.1.5 SBODG air intake and exhaust systems (refer to Section 8.4.1).

2.1.6 Crankcase ventilation system (refer to Section 9.5.8).

2.2 SBODG system instrumentation has been calibrated and is functional for performance of the following test.

4.0 DATA REQUIRED

4.1 Values of input and output signals for correlation purposes, as required.

4.2 Values of output signals triggering audio and visual alarms.

5.0 ACCEPTANCE CRITERIA

5.1 The excore instrumentation is arranged as illustrated on the plant layout drawings. Reference Figure 7.1-15 for additional information.

5.2 The intermediate range and power range detectors generate neutron flux measurement signals as inputs to the protection system using simulated signals.

5.3 The excore instrumentation receives power from its respective Class 1E division.

5.4 Verify that safety-related components meet electrical independence and redundancy requirements.

5.5 The excore instrumentation system functions as described in Section 7.1.1.5.3.

14.2.12.11.19 Radiation Monitoring System (Test #143)

1.0 OBJECTIVE

1.1 To verify the functional performance of the airborne radiation monitoring system.

1.2 To verify the functional performance of the area radiation monitoring system.

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1.3 To verify high radiation activity generates appropriate PS isolation signals.

2.0 PREREQUISITES

2.1 Construction activities on the radiation monitoring system have been completed with all radiation monitors positioned per Table 12.3-3, Table 12.3-4, and Table 11.5-1.

2.2 Radiation monitoring system instrumentation has been calibrated and is operating satisfactorily prior to performing the following test.

2.3 Support systems required for operation of the radiation monitoring system are completed and functional.

2.4 Test instrumentation is available and calibrated.

2.5 Calibration check source is available, as required.

2.6 Verify that factory acceptance testing has been completed.

2.7 Verify proper operation of alarm setpoints, operation, control, and indication functions.

3.0 TEST METHOD

3.1 Verify the operation of the radiation monitor (refer to Table 11.5-1) using a check source and external test equipment, as applicable.

3.2 Check the self-testing feature of the radiation monitor, as applicable.

3.3 Compare local and remote indications.

3.4 Verify as-designed local and remote alarm actuations, as applicable.

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3.5 Initiate a high radiation signal to the MCR air intake (refer to Table 11.5-1, Monitors R-29 and R-30, main steam line (refer to Table 11.5-1, Monitors R-55 through R-58), and containment high range radiation monitors (refer to Table 12.3-3) to verify that control actuations meet design requirements. The source of initiation of the signal, listed in order of preference, should be one of the following:

3.5.1 Internal check source (verify that check source strength is capable of generating desired control actuations).

3.5.2 Radiation calibration check source (verify that check source does not generate a personnel hazard during the test).

3.5.3 Simulated high radiation signal at the radiation detector.

3.6 Verify that the radiation monitoring system operates over the design range using actual or simulated signals.

3.7 Verify that the radiation monitoring system responds as designed to actual or simulated limiting malfunctions or failures.

3.8 Verify that the radiation monitoring system response meets the accident analysis assumptions, such as time response, accuracy, and control stability.

3.9 Verify redundancy and electrical independence of the radiation monitoring system design.

4.0 DATA REQUIRED

4.1 Radiation monitor response to a check source, as applicable.

4.2 Technical data associated with the source.

4.3 Local and remote responses to test signals, as applicable.

4.4 Signals levels necessary to cause alarm actuation.

5.0 ACCEPTANCE CRITERIA

5.1 The radiation monitoring system (MCR air intake duct activity) generates a Main Control Room air intake activity measurement signal as input to the protection system (refer to Table 12.3-3).

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5.2 The radiation monitoring system (containment high range activity) generates a containment isolation signal as an input to the ~~protection system~~PS, as designed (refer to Section 11.5.4.1 and Table 11.5-1, monitors R-55 through 5-58).

5.3 The radiation monitoring system (main steam line activity) generates a steam generator isolation signal as an input to the protection system, as designed (refer to Table 11.5-1, Monitors R-55 through R-58).

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5.4 The airborne and area radiation monitors function as described in Sections 7.1.1.5.5, 7.3.1, 7.5.1, and 12.3.4. The airborne and area radiation monitors are listed in Table 11.5-1, Table 12.3-4, and Table 12.3-3, respectively.

5.5 The radiation monitoring system (containment building ventilation system - internal filtration subsystem) meets design requirements for RCS leak detection required to demonstrate compliance with Technical Specification Chapter 16, LCO 3.4.14 (refer to Section 11.5.4.8, 11.5.3.1.4, and Table 11.5-1, Footnote 16 for Monitor R-10).

14.2.12.11.20 Process and Effluent Radiological Monitoring System (Test #144)

1.0 OBJECTIVE

1.1 To verify that the process and effluent radiological monitoring system can detect and record specific radiation levels, and to verify alarms and interlocks.

2.0 PREREQUISITES

2.1 Construction activities on the process and effluent radiological monitoring system have been completed.

2.2 Process and effluent radiological monitoring system instrumentation has been calibrated and is operating satisfactorily prior to performing the following test.

2.3 Support systems required for operation of the process and effluent radiological monitoring system is completed and functional.

2.4 Test instrumentation is available and calibrated.

2.5 Calibration check sources are available in appropriate forms (gaseous, solutions, or plated sources) for the analyses referenced in Table 11.5-1.

2.6 Verify that factory acceptance testing has been completed.

2.7 Verify proper operation of alarm, control, and indication functions.

3.0 TEST METHOD

3.1 Verify calibration and operation of the monitor using a check source and external test equipment, as necessary.

3.2 Check the self-testing feature of the monitor.

3.3 Record the response time of the monitor.

3.0 TEST METHOD

3.1 Observe process monitor indications, outputs to interface equipment and alarm operation, utilizing the built-in test features.

3.2 Verify calibration of the process monitor, utilizing the check source.

4.0 DATA REQUIRED

4.1 Check source data.

4.2 Process monitor operating data.

4.3 Process monitor response to the check source.

4.4 Value of parameters required to actuate alarms.

5.0 ACCEPTANCE CRITERIA

5.1 The process radiation monitor of the process sampling functions as follows:

5.1.1 Radiation monitors are installed on process paths as shown on plant layout drawings.

5.1.2 The radiation monitors have been source checked to verify response.

5.1.3 Preliminary alarms setpoints have been established and calibrated in the equipment.

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5.2 Process radiation monitors function as described in Section 11.5.4 and Table 11.5-1.

14.2.12.11.27 Personnel Radiation Monitors (Test #160)

A COL applicant that references the U.S. EPR design certification will provide site-specific test abstract information for personnel radiation monitors. The following is a typical COL test; if a site-specific test will be used, the COL applicant will provide the test.

1.0 OBJECTIVE

1.1 To demonstrate proper operation of personnel radiation monitors.

2.0 PREREQUISITES

2.1 Construction activities on personnel radiation monitor support systems are complete.

2.2 Construction activities related to the installation of vendor supplied personnel radiation monitors are complete. The personnel radiation monitors have been installed per manufacture's recommendations.

2.3 A suitable test source is available for testing.

3.8 Check electrical independence and redundancy of power supplies for safety-related functions by selectively removing power and determining loss of function.

4.0 DATA REQUIRED

4.1 Simulated normal and abnormal accident data.

5.0 ACCEPTANCE CRITERIA

5.1 The accident monitoring system provides the outputs described in the accident response procedures (Abnormal Operating Procedures, Emergency Operating Procedures, Severe Accident Mitigation Procedures, etc.).

5.2 Verify that safety-related components meet electrical independence and redundancy requirements.

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5.3 The accident monitoring system functions as described in Section 7.5.1.2.

5.4 The accident monitoring system meets design requirements to monitor radiation (refer to Table 11.5-1).

14.2.12.12.2 Main Steam Relief Trains (Test #148)

1.0 OBJECTIVE

1.1 To demonstrate the proper operation of the MSRT.

1.2 To demonstrate electrical independence and redundancy of safety-related power supplies.

2.0 PREREQUISITES

2.1 Construction activities on the MSRT (MSRCV and MSRIV) and interfacing equipment have been completed.

2.2 The MSRIV pilot valves have been calibrated and are operating satisfactorily prior to performing the following test.

2.3 External test equipment has been calibrated and is functional.

2.4 Support systems required for operation of the MSRT are functional.

2.5 Verify that factory acceptance testing has been completed.

2.6 Verify proper operation of alarm, control, and indication functions.

2.7 Verify that the MSRT tested flow capacity meets the design requirements.

2.7.1 Table 14.3-1 Item 1-50.

3.0 TEST METHOD

3.1 Verify that the impacted systems that contain liquids are leak tight when pressurized at normal system operating pressure and temperature, as applicable.

3.2 Identified leaks should be repaired and retested, as applicable.

3.3 Verify that impacted systems that contain gases are pressurized to normal operating pressure with a mixture of compressed air and a suitable tracer gas, such as helium/SF_6.

3.4 Leaking portions of impacted systems can be corrected by maintenance activities and retested by either of the methods described above.

4.0 DATA REQUIRED

4.1 Walkdown inspection reports completed by qualified personnel.

4.2 Helium/SF_6 equipment calibration references.

5.0 ACCEPTANCE CRITERIA

5.1 The leakage from impacted systems meets the requirements of NUREG-0578, Recommendation 2.1.6.a.

5.2 The leakage from impacted systems meets the requirements of NUREG-0660, Item III.D.1.1.

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5.3 The leakage from impacted systems meets the requirements of NUREG-0664, Part 2.

5.4 The system integrity test meets the design requirements to monitor radiation (refer to Table 11.5-1).

14.2.12.12.8 Remote Safe Shutdown (Test #154)

1.0 OBJECTIVE

1.1 To demonstrate the proper operation of the remote safe shutdown function.

2.0 PREREQUISITES

2.1 The instrumentation used during safe shutdown has been calibrated and is operating satisfactorily prior to performing the following test.

2.2 External test instrumentation is available and calibrated.

2.3 Support systems required for testing safe shutdown are functional.

3.0 TEST METHOD

3.1 Verify that safe shutdown control signals override lower priority signals.

3.0 TEST METHOD

3.1 Verify power sources to post accident related equipment.

3.2 Validate that external inputs are received and processed correctly by the appropriate system devices.

3.3 Verify that alarms and indication displays respond correctly to actual or simulated inputs.

3.4 Verify the functionality of required software application programs.

3.5 Verify the correct operation of data output devices and displays at applicable work stations and terminals.

4.0 DATA REQUIRED

4.1 Computer generated summaries of external input data, data processing, analysis functions, displayed information, and permanent data records.

5.0 ACCEPTANCE CRITERIA

5.1 The instruments that are designated as the post-accident monitoring instruments have been verified to include all of the instruments listed in the emergency operating procedures (Abnormal Operating Procedures, Emergency Operating Procedures, Severe Accident Mitigation Guidelines, etc.)

5.2 The PAM functions as described in Section 7.5.

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5.3 ~~Radiation~~Continuous monitoring instrumentation used to perform post-accident monitoring that is described in ~~Section 9.3.2~~Tables 9.3.2-1, 9.3.2-2, and 11.5-1 will meet the design requirements for the radiation monitor. This includes, but is not limited to, the following (that could adversely impact the ability to measure the parameters described in ~~Section 9.3.2~~Tables 9.3.2-1, 9.3.2-2, and 11.5-1):

5.3.1 Range.

5.3.2 Response time.

5.3.3 Sensitivity.

5.3.4 Maximum anticipated drift between calibrations.

14.2.12.12.10 Pressurizer Pressure and Level Control (Test #156)

1.0 OBJECTIVE

1.1 To verify the proper operation of the pressurizer pressure control (PPC) and pressurizer level control (PLC).

2.0 PREREQUISITES

2.1 Construction activities on the PPC and PLC have been completed.

2.3 Pressurizer surge line insulation is installed in the configuration that is anticipated for plant operation.

3.0 TEST METHOD

3.1 Secure pressurizer heaters except the proportional band heaters. Continuously monitor the proportional heater output throughout the remainder of the test.

3.2 Secure the spray valves, so that the only flow through the spray header and the pressurizer surge line is passing through the continuous spray valves.

3.3 Adjust the continuous spray valves to just clear the minimum pressurizer surge line temperature. Verify that both continuous spray valves are open approximately the same amount.

3.4 Determine the remaining proportional band heater capacity and adjust the continuous spray valves until only 50 percent of the previously determined margin remains. Verify that both continuous spray valves are open approximately the same amount.

3.5 Using various combinations of pressurizer spray valves, measure and record the rate at which the pressurizer pressure can be reduced.

4.0 DATA REQUIRED

4.1 RCS temperature and pressure.

4.2 Pressurizer surge line temperature.

4.3 Pressurizer proportional heater output with the pressurizer spray valves closed.

4.4 Continuous spray valve settings.

4.5 Pressurizer pressure and pressurization spray valve positions during depressurization.

5.0 ACCEPTANCE CRITERIA

5.1 The pressurizer meets design requirements (refer to Section 5.4.10).

14.2.12.14.9 Post-Core Reactor Coolant System Leak Rate Measurement (Test #187)

1.0 OBJECTIVE

1.1 To measure the RCS leakage at HZP (pressure and temperature) conditions. In general, it is better to measure leakage over a one hour period unless VCT makeup precludes test duration of one hour.

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1.2 To distinguish between identified and unidentified leakage, refer to Chapter 16, SR 3.4.12.1.

2.0 PREREQUISITES

2.1 The RCS is at HZP (pressure and temperature) conditions.

2.2 The RCS and the CVCS are operating normally with no makeup or letdown diversion.

2.3 The VCT level is high enough to prevent makeup during the test.

2.4 Permanently mounted instrumentation is calibrated and is operating satisfactorily prior to performing the following test.

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2.5 Verify that one containment sump level monitor is available. This monitor meets the requirements of Chapter 16 LCO 3.4.14 a. Note that this monitor can only detect RCS leakage that occurs in the containment.

2.6 Verify preoperational Test #143 has been satisfactorily completed for radiation monitoring instrumentation.

2.7 Verify that radiation monitor R-10 is available for subsequent RCS leakage tests that will be performed at power (refer to Table 11.5-1, Footnote 16). Radiation monitor R-10 meets the requirements of Chapter 16 LCO 3.4.14 b. Note that this monitor can only detect radioactive reactor coolant pressure boundary leakage that occurs in the containment.

2.8 Verify that one containment air cooler condensate flow rate monitor is available. This monitor meets the requirements of Chapter 16 LCO 3.4.14 c. Note that this monitor can only detect RCS leakage that occurs in the containment.

3.0 TEST METHOD

3.1 Convert mass changes to gallons of water at normal atmospheric conditions (pressure and temperature).

3.2 Measure changes in water inventory of the RCS as follows:

3.2.1 Record mass changes in the pressurizer due to temperature and level changes.

3.2.2 Record mass changes due to RCS pressure and temperature changes.

3.3 Measure changes in water inventory of the CVCS and connected systems as follows:

3.3.1 Record mass changes in the VCT due to temperature and level changes.

3.3.2 Record mass changes in the RCDT due to level changes.

3.3.3 Record mass changes in the passive SI accumulators due to temperature and level changes. If SI accumulator mass has not increased it shall be ignored in the RCS leakrate calculation.

3.4 Determine total leakage, identified leakage (i.e., leakage into identifiable sources) and unidentified leakage (e.g., leakage into the containment atmosphere and other, valve packing and other paths that include leakage from non–RCS sources).

3.5 Demonstrate proper response of radiation monitors that are used to determine RCS leakage by comparing calculated values to the radiation monitor estimated leakage values.

4.0 DATA REQUIRED

4.1 Pressurizer pressure, level, and temperature.

4.2 VCT level, temperature, and pressure.

4.3 RCDT level, temperature, and pressure.

4.4 RCS temperature and pressure.

4.5 SI Accumulator level and pressure.

4.6 Time interval.

5.0 ACCEPTANCE CRITERIA

5.1 Identified and unidentified leakage shall be within the limits described in ~~in~~ Technical Specification 3.4.12 and design leakage rates described in Section 5.2.5.

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5.2 ~~The values determined by the radiation monitoring instrumentation described in Technical Specification 3.4.14 are within reasonable agreement with the instrumentation used to calculate Technical Specification 3.4.12 leakage.~~ The instrumentation that is used detects and monitors unidentified leakage inside the Containment Building is available (refer to Section 5.2.5 and Chapter 16, LCO 3.4.14):

5.2.1 Containment sump level monitor.

5.2.2 Containment atmosphere radiation monitor (refer to Section 11.5.4.8 and Table 11.5-1, Footnote 16 for monitor R-10).

5.2.3 Containment air cooler condensate monitor.

14.2.12.14.10 Post-Core Incore Instrumentation (Test #188)

1.0 OBJECTIVE

1.1 To measure the leakage resistance of the fixed incore detectors.

1.2 To demonstrate that the incore thermocouples are functional (refer to Section 7.1.1.5.2 for a description of fixed thermocouples).

2.0 PREREQUISITES

2.1 Permanently installed instrumentation is calibrated and is operating satisfactorily prior to performing the following test.

1.1.4 Radio isotopic concentration data of the radioactive elements (e.g., cesium, iodine, iron, cobalt).

1.2 To demonstrate performance of permanent plant sampling and analysis procedures, while confirming that primary and secondary chemistry requirements are being met.

1.3 To verify that the primary and secondary systems are operating within design limits. This procedure shall be performed at the following plateau:

1.3.1 25 percent reactor power in accordance with RG 1.68.

1.3.2 50 percent reactor power in accordance with RG 1.68.

1.3.3 75 percent reactor power in accordance with RG 1.68.

1.3.4 ≥98 percent reactor power in accordance with RG 1.68.

2.0 PREREQUISITES

2.1 The reactor is stable at the desired power level.

2.2 Required sampling systems are functional and analysis instrumentation are calibrated using calibration gases and solutions as referenced in the radioactive and non-radioactive analyses of ~~Table 9.3.2-2~~Tables 9.3.2-1 and 9.3.2-2.

3.0 TEST METHOD

3.1 Samples shall be collected from the RCS and secondary system at various power levels and analyzed in the laboratory using applicable sampling and analysis procedures.

3.2 Collect samples at various process radiation monitors, perform analysis in the laboratory, and compare the samples with the process radiation monitor output (refer to Table 11.5-1). RAI 273, Q. 11.05-2

3.3 Verify that primary and secondary sample results meet design limits.

4.0 DATA REQUIRED

4.1 Reactor power.

4.2 RCS and secondary temperature.

4.3 Boron concentration and boron-10 isotopic abundance.

4.4 Core average burnup.

4.5 Isotopic activities.

5.0 ACCEPTANCE CRITERIA

5.1 Measured activity levels are within their limits.

5.2 Laboratory analyses and process radiation monitors agree with the within measurement uncertainties as designed (refer to Section 9.3.2), or investigation of the discrepancies has been initiated.

5.3 Samples of RCS and secondary fluids can be obtained from design locations as designed (refer to Sections 9.3.2 and 11.5).

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5.4 Continuous ~~and chemistry lab~~ instrumentation used to analyze primary and secondary sampling parameters described in Table 9.3.2-1 and Table 9.3.2-2 will meet the design requirements for the measurements. This includes, but is not limited to, the following (that could adversely impact the ability to measure the parameters described in Table 9.3.2-1 and Table 9.3.2-2):

5.4.1 Range.

5.4.2 Response time.

5.4.3 Sensitivity.

5.4.4 Maximum anticipated drift between calibrations.

5.5 Radiation monitoring instrumentation used to perform radiation monitoring that is described in Table 11.5-1 will meet the design requirements for the radiation monitor. This includes, but is not limited to, the following (that could adversely impact the ability to measure the parameters described in Table 11.5-1, Monitor R-10):

5.5.1 Range.

5.5.2 Response time.

5.5.3 Sensitivity.

5.5.4 Maximum anticipated drift between calibrations.

14.2.12.18.6 Failed Fuel Detection (Test #205)

1.0 OBJECTIVE

1.1 To collect chemistry samples of the RCS and secondary at the specified power level to record the following:

1.1.1 Boron concentration and boron-10 isotopic abundance.

1.1.2 Concentration of non-radioactive elements and soluble particulates.

1.1.3 Measured pH of the fluids.

1.1.4 Radioisotopic concentration data of the radioactive elements (e.g., cesium, iodine, strontium, barium, cerium, and noble gases).

1.2 To demonstrate performance of permanent plant sampling and analysis procedures. There is typically some RCS activity from tramp, fuel dust that is on the outer surface of the cladding.

1.3 To perform a cross-check of the failed fuel monitor instrumentation.

1.4 This test shall be performed at the following power plateaus:

1.4.1 25 percent reactor power.

1.4.2 ≥98 percent reactor power.

2.0 PREREQUISITES

2.1 The reactor is stable at the desired power level.

2.2 Required sampling systems are functional.

2.3 Calibrating gases and solutions are available for radioactive and non-radioactive analyses referenced in ~~Table 9.3.2-2~~Tables 9.3.2-1 and 9.3.2-2.

3.0 TEST METHOD

3.1 Samples shall be collected from the RCS and secondary system at various power levels and analyzed in the laboratory using applicable sampling and analysis procedures.

3.2 Collect samples at various process radiation monitors, perform analysis in the laboratory, and compare the samples with the process radiation monitor output (refer to Table 11.5-1).

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4.0 DATA REQUIRED

4.1 Reactor power.

4.2 RCS and secondary temperature.

4.3 Boron concentration and boron-10 isotopic abundance.

4.4 Core average burnup.

4.5 Isotopic activities.

5.0 ACCEPTANCE CRITERIA

5.1 Measured activity levels are within their limits.

5.2 Laboratory analyses and process radiation monitors agree with the within measurement uncertainties as designed (refer to Section 9.3.2 or investigation of the discrepancies has been initiated.

5.3 Samples of RCS and secondary fluids can be obtained from design locations as designed (refer to Sections 9.3.2 and 11.5).

5.4 ~~Chemistry lab~~Continuous instrumentation used to analyze primary sampling parameters described in ~~Table 9.3.2-2~~Tables 9.3.2-1 and 9.3.2-2 will meet the design requirements for the measurements. This includes, but is not limited to, the following (that could adversely impact the ability to measure the parameters described in ~~Table 9.3.2-2~~Tables 9.3.2-1 and 9.3.2-2):

5.4.1 Range.

5.4.2 Response time.

5.4.3 Sensitivity.

5.4.4 Maximum anticipated drift between calibrations.

14.2.12.20.1 Liquid Waste Storage and Processing Systems (Test #215)

1.0 OBJECTIVE

1.1 To demonstrate that the operation of the liquid waste storage and processing systems (LWSPS) for collection, processing, recycling, and preparation of liquid waste for release to the environment is satisfactory.

1.2 To determine the ability of plant systems to process radioactive effluents. This procedure shall be performed at the following plateaus:

1.2.1 75 percent reactor power in accordance with RG 1.68.

1.2.2 ≥98 percent reactor power in accordance with RG 1.68.

2.0 PREREQUISITES

2.1 The liquid waste processing equipment is functional.

3.0 TEST METHOD

3.1 Monitor the performance of the LWSPS.

3.2 Verify isotopic concentrations in the liquid stream.

3.3 Verify that the LWSPS is capable of collecting and processing liquid waste per design.

4.0 DATA REQUIRED

4.1 Conditions of Measurement.

4.1.1 Reactor power history and RCS radioactivity level.

4.1.2 Liquid waste processing system tank levels.

4.1.3 Liquid waste processing system demineralizer data.

4.1.4 Liquid waste processing system evaporator data.

4.1.5 Liquid waste processing system centrifuge data.

5.0 ACCEPTANCE CRITERIA

5.1 The LWSPS processes radioactive effluents as designed (refer to Sections 11.2, 11.5, ~~and~~ 13.4, and Table 11.5-1).

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14.2.12.20.2 Gaseous Waste Processing System (Test #216)

1.0 OBJECTIVE

1.1 To demonstrate that the operation of the gaseous waste processing system (GWPS) for collection and processing of radioactive and potentially flammable gases vented from plant equipment is performing satisfactorily.

1.2 To determine the ability of plant systems to process radioactive effluents. This procedure shall be performed at the following plateaus:

1.2.1 75 percent reactor power in accordance with RG 1.68.

1.2.2 ≥98 percent reactor power in accordance with RG 1.68.

2.0 PREREQUISITES

2.1 The gaseous waste processing equipment is functional.

3.0 TEST METHOD

3.1 Verify that the gaseous waste processing simultaneously collects and processes gaseous waste per design.

4.0 DATA REQUIRED

4.1 Conditions of Measurement:

4.1.1 Reactor power history and RCS radioactivity level.

4.1.2 Containment temperature and humidity.

4.1.3 Condenser operating data.

4.1.4 Effluent control monitor operating data.

4.1.5 Gas analyzer operating data.

4.1.6 Gas transport times.

4.1.7 Recombiner operating data

5.0 ACCEPTANCE CRITERIA

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5.1 The GWPS processes radioactive and potentially flammable gases effluent as designed (refer to Sections 11.3, 11.5, ~~and~~ 13.4, and Table 11.5-1, Monitors R-1 and R-2).

14.2.12.20.3 Loss of Feedwater Pump (Test #217)

1.0 OBJECTIVE

1.1 To evaluate system response to a loss of one of three operating feedwater pumps.

1.2 To demonstrate that rapid load changes can be accomplished in a manner that maintains plant safety.

1.3 This procedure shall be performed at the following plateau:

1.3.1 75 percent reactor power.

2.0 PREREQUISITES

2.1 The reactor is operating at the desired power level.