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RA-18-0219

10CFR 50.4
10CFR 50.71(e)

December 4, 2018

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Subject: Duke Energy Carolinas, LLC
Catawba Nuclear Station, Units 1 and 2
Docket Nos. 50-413 and 50-414
Technical Specification Bases Changes and UFSAR/Selected Licensee Commitment Changes

Pursuant to 10CFR 50.4, please find attached changes to the Catawba Nuclear Station Technical Specification Bases. These Bases changes were made according to the provisions of Technical Specification 5.5.14, "Technical Specifications (TS) Bases Control Program."

Also, Pursuant to 10 CFR 50.71(e), please find attached changes to the Catawba Nuclear Station Selected Licensee Commitments Manual. This document constitutes Chapter 16 of the Updated Final Safety Analysis Report (UFSAR).

Any questions regarding this information should be directed to Tolani Owusu, Regulatory Affairs, at (803) 701-5385.

I certify that I am a duly authorized officer of Duke Energy Carolinas, LLC, and that the information contained herein accurately represents changes made to the Technical Specification Bases since the previous submittal.

Sincerely,

Tom Simril

Tom Simril
Vice President, Catawba Nuclear Station

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NRR

Enclosures:
1) TS Bases Insertion/Removal Instructions
2) TS LOEP and Bases Replacement Pages
3) SLC Manual Insertion/Removal Instructions
4) SLC Manual Replacement Pages

xc: w/enclosures

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Joseph Austin, Senior Resident Inspector
Catawba Nuclear Station

Enclosure 1

TS Bases Insertion/Removal Instructions

Removal and insertion instructions for Catawba Nuclear Station Tech Spec Bases Changes for February 1, 2018 thru October 31, 2018.

REMOVE THESE PAGES	INSERT THESE PAGES

LIST OF EFFECTIVE PAGES

Pages 1-18 Revision 16 (1/23/18) ***List of Effective Pages Revisions 17-19 have been superseded by Revision 20 (10/23/18) and therefore do not need to be replaced in your manuals.***	Pages 1-18 Revision 20 (10/23/18)

TECHNICAL SPECIFICATIONS

BASES

TAB 3.0

B 3.0-1 – B 3.0-21 Revision 4	B 3.0-1 – B 3.0-21 Revision 6

TS Bases B 3.0-1 thru 21, Revision 5 was superseded by Revision 6 and although included in this package, does not need to be placed in your manual.

BASES

TAB 3.1

B 3.1.8-1 – B 3.1.8-6 Revision 3	B 3.1.8-1 – B 3.1.8-6 Revision 4

BASES

TAB 3.4

B 3.4.10-1 – B 3.4.10-4 Revision 3	B 3.4.10-1 – B 3.4.10-4 Revision 4
B 3.4.12-1 – B 3.4.12-13 Revision 5	B 3.4.12-1 – B 3.4.12-13 Revision 6

BASES

TAB 3.5

B 3.5.2-1 – B 3.5.2-11 Revision 4	B 3.5.2-1 – B 3.5.2-11 Revision 5

BASES

TAB 3.6

B 3.6.3-1 – B 3.6.3-14 Revision 6	B 3.6.3-1 – B 3.6.3-14 Revision 7
B 3.6.6-1 – B 3.6.6-8 Revision 7	B 3.6.6-1 – B 3.6.6-8 Revision 8

BASES

TAB 3.7

B 3.7.1-1 – B 3.7.1-5 Revision 2	B 3.7.1-1 – B 3.7.1-5 Revision 3
B 3.7.2-1 – B 3.7.2-5 Revision 3	B 3.7.2-1 – B 3.7.2-5 Revision 4
B 3.7.3-1 – B 3.7.3-6 Revision 2	B 3.7.3-1 – B 3.7.3-6 Revision 3
B 3.7.5-1 – B 3.7.5-9 Revision 4	B 3.7.5-1 – B 3.7.5-9 Revision 5
B 3.7.6-1 – B 3.7.6-3 Revision 5	B 3.7.6-1 – B 3.7.6-3 Revision 6
B 3.7.12-1 – B 3.7.12-7 Revision 7	B 3.7.12-1 – B 3.7.12-7 Revision 8

If you have any questions concerning the contents of this Technical Specification Bases update, please contact Toni Lowery at (803) 701-5046.

Enclosure 2

TS LOEP and Bases Replacement Pages

Catawba Nuclear Station Technical Specifications
List of Effective Pages

Page Number	Amendments	Revision Date
i	177/169	4/08/99
ii	219/214	3/01/05
iii	215/209	6/21/04
iv	173/165	9/30/98
1.1-1	173/165	9/30/98
1.1-2	268/264	6/25/12
1.1-3	299/295	10/23/18
1.1-4	268/264	6/25/12
1.1-5	281/277	4/29/16
1.1-6	268/264	6/25/12
1.1.7	179/171	8/13/99
1.2-1	173/165	9/30/98
1.2-2	173/165	9/30/98
1.2-3	173/165	9/30/98
1.3-1	298/294	2/1/18
1.3-2	298/294	2/1/18
1.3-3	298/294	2/1/18
1.3-4	298/294	2/1/18
1.3-5	298/294	2/1/18
1.3-6	298/294	2/1/18
1.3-7	298/294	2/1/18
1.3-8	298/294	2/1/18
1.3-9	298/294	2/1/18
1.3-10	298/294	2/1/18
1.3-11	298/294	2/1/18
1.3-12	298/294	2/1/18
1.3-13	298/294	2/1/18
1.3-14	298/294	2/1/18
1.4-1	173/165	9/30/98
1.4-2	173/165	9/30/98

1.4-3	173/165	9/30/98
1.4-4	173/165	9/30/98
2.0-1	210/204	12/19/03
3.0-1	288/284	4/26/17
3.0-2	298/294	2/1/18
3.0-3	235/231	3/19/07
3.0-4	288/284	4/26/17
3.0-5	298/294	2/1/18
3.0-6	235/231	3/19/07
3.1.1-1	263/259	3/29/11
3.1.2-1	296/292	10/23/17
3.1.2-2	263/259	3/29/11
3.1.3-1	173/165	9/30/98
3.1.3-2	275/271	04/14/15
3.1.3-3	173/165	9/30/98
3.1.4-1	173/165	9/30/98
3.1.4-2	173/165	9/30/98
3.1.4-3	263/259	3/29/11
3.1.4-4	263/259	3/29/11
3.1.5-1	173/165	9/30/98
3.1.5-2	263/259	3/29/11
3.1.6-1	173/165	9/30/98
3.1.6-2	173/165	9/30/98
3.1.6-3	263/259	3/29/11
3.1.7-1	173/165	9/30/98
3.1.7-2	173/165	9/30/98
3.1.8-1	291/287	7/26/17
3.1.8-2	263/259	3/29/11
3.2.1-1	173/165	9/30/98
3.2.1-2	173/165	9/30/98
3.2.1-3	263/259	3/29/11
3.2.1-4	263/259	3/29/11
3.2.1-5	263/259	3/29/11
3.2.2-1	173/165	9/30/98

3.2.2-2	173/165	9/30/98
3.2.2-3	263/259	3/29/11
3.2.2-4	263/259	3/29/11
3.2.3-1	263/259	3/29/11
3.2.4-1	173/165	9/30/98
3.2.4-2	173/165	9/30/98
3.2.4-3	173/165	9/30/98
3.2.4-4	263/259	3/29/11
3.3.1-1	173/165	9/30/98
3.3.1-2	247/240	12/30/08
3.3.1-3	247/240	12/30/08
3.3.1-4	207/201	7/29/03
3.3.1-5	247/240	12/30/08
3.3.1-6	247/240	12/30/08
3.3.1-7	247/240	12/30/08
3.3.1-8	173/165	9/30/98
3.3.1-9	263/259	3/29/11
3.3.1-10	263/259	3/29/11
3.3.1-11	263/259	3/29/11
3.3.1-12	278/274	4/08/16
3.3.1-13	263/259	3/29/11
3.3.1-14	263/259	3/29/11
3.3.1-15	263/259	3/29/11
3.3.1-16	278/274	4/08/16
3.3.1-17	263/259	3/29/11
3.3.1-18	263/259	3/29/11
3.3.1-19	278/274	4/08/16
3.3.1-20	263/259	3/29/11
3.3.1-21	263/259	3/29/11
3.3.1-22	263/259	3/29/11
3.3.2-1	173/165	9/30/98
3.3.2-2	247/240	12/30/08
3.3.2-3	247/240	12/30/08
3.3.2-4	247/240	12/30/08

3.3.2-5	264/260	6/13/11
3.3.2-6	264/260	6/13/11
3.3.2-7	249/243	4/2/09
3.3.2-8	249/243	4/2/09
3.3.2-9	249/243	4/2/09
3.3.2-10	263/259	3/29/11
3.3.2-11	263/259	3/29/11
3.3.2-12	263/259	3/29/11
3.3.2-13	277/273	12/18/15
3.3.2-14	277/273	12/18/15
3.3.2-15	277/273	12/18/15
3.3.2-16	277/273	12/18/15
3.3.2-17	277/273	12/18/15
3.3.2-18 (new)	277/273	12/18/15
3.3.3-1	219/214	3/1/05
3.3.3-2	219/214	3/1/05
3.3.3-3	263/259	3/29/11
3.3.3-4	219/214	3/1/05
3.3.4-1	213/207	4/29/04
3.3.4-2	263/259	3/29/11
3.3.4-3	272/268	2/27/14
3.3.5-1	173/165	9/30/98
3.3.5-2	277/273	12/18/15
3.3.6-1	196/189	3/20/02
3.3.6-2	263/259	3/29/11
3.3.6-3	196/189	3/20/02
3.3.9-1	207/201	7/29/03
3.3.9-2	207/201	7/29/03
3.3.9-3	263/259	3/29/11
3.3.9-4	263/259	3/29/11
3.4.1-1	210/204	12/19/03
3.4.1-2	210/204	12/19/03
3.4.1-3	263/259	3/29/11
3.4.1-4	283/279	6/02/16

3.4.1-5 (deleted)	184/176	3/01/00
3.4.1-6 (deleted)	184/176	3/01/00
3.4.2-1	173/165	9/30/98
3.4.3-1	173/165	9/30/98
3.4.3-2	263/259	3/29/11
3.4.3-3	281/277	4/29/16
3.4.3-4	212/206	3/4/04
3.4.3-5	281/277	4/29/16
3.4.3-6	212/206	3/4/04
3.4.4-1	263/259	3/29/11
3.4.5-1	207/201	7/29/03
3.4.5-2	207/201	7/29/03
3.4.5-3	263/259	3/29/11
3.4.6-1	212/206	3/4/04
3.4.6-2	263/259	3/29/11
3.4.6-3	282/278	4/26/17
3.4.7-1	212/206	3/4/04
3.4.7-2	263/259	3/29/11
3.4.7-3	282/278	4/26/17
3.4.8-1	207/201	7/29/03
3.4.8-2	282/278	4/26/17
3.4.9-1	173/165	9/30/98
3.4.9-2	263/259	3/29/11
3.4.10-1	294/290	10/23/17
3.4.10-2	299/295	10/23/18
3.4-11-1	213/207	4/29/04
3.4.11-2	173/165	9/30/98
3.4.11-3	263/259	3/29/11
3.4.11-4	263/259	3/29/11
3.4.12-1	212/206	3/4/04
3.4.12-2	213/207	4/29/04
3.4.12-3	212/206	3/4/04
3.4.12-4	212/206	3/4/04
3.4.12-5	263/259	3/29/11

3.4.12-6	263/259	3/29/11
3.4.12-7	263/259	3/29/11
3.4.12-8	263/259	3/29/11
3.4.13-1	267/263	3/12/12
3.4.13-2	267/263	3/12/12
3.4.14-1	173/165	9/30/98
3.4.14-2	173/165	9/30/98
3.4.14-3	299/295	10/23/18
3.4.14-4	263/259	3/29/11
3.4.15-1	234/230	9/30/06
3.4.15-2	234/230	9/30/06
3.4.15-3	234/230	9/30/06
3.4.15-4	263/259	3/29/11
3.4.16-1	268/264	6/25/12
3.4.16-2	268/264	6/25/12
3.4.16-3(deleted)	268/264	6/25/12
3.4.16-4(deleted)	268/264	6/25/12
3.4.17-1	263/259	3/29/11
3.4.18-1	280/276	4/26/16
3.4.18-2	280/276	4/26/16
3.5.1-1	211/205	12/23/03
3.5.1-2	263/259	3/29/11
3.5.1-3	263/259	3/29/11
3.5.2-1	253/248	10/30/09
3.5.2-2	299/295	10/23/18
3.5.2-3	263/259	3/29/11
3.5.3-1	213/207	4/29/04
3.5.3-2	173/165	9/30/98
3.5.4-1	173/165	9/30/98
3.5.4-2	269/265	7/25/12
3.5.5-1	173/165	9/30/98
3.5.5-2	263/259	3/29/11
3.6.1-1	173/165	9/30/98
3.6.1-2	192/184	7/31/01

3.6.2-1	173/165	9/30/98
3.6.2-2	173/165	9/30/98
3.6.2-3	173/165	9/30/98
3.6.2-4	173/165	9/30/98
3.6.2-5	263/259	3/29/11
3.6.3-1	173/165	9/30/98
3.6.3-2	290/286	7/21/17
3.6.3-3	290/286	7/21/17
3.6.3-4	290/286	7/21/17
3.6.3-5	263/259	3/29/11
3.6.3-6	299/295	10/23/18
3.6.3-7	192/184	7/31/01
3.6.4-1	263/259	3/29/11
3.6.5-1	173/165	9/30/98
3.6.5-2	263/259	3/29/11
3.6.6-1	282/278	4/26/17
3.6.6-2	299/295	10/23/18
3.6.8-1	213/207	4/29/04
3.6.8-2	263/259	3/29/11
3.6.9-1	253/248	10/30/09
3.6.9-2	263/259	3/29/11
3.6.10-1	173/165	9/30/98
3.6.10-2	289/285	5/08/17
3.6.11-1	263/259	3/29/11
3.6.11-2	263/259	3/29/11
3.6.12-1	263/259	3/29/11
3.6.12-2	263/259	3/29/11
3.6.12-3	263/259	3/29/11
3.6.13-1	256/251	6/28/10
3.6.13-2	263/259	3/29/11
3.6.13-3	263/259	3/29/11
3.6.14-1	173/165	9/30/98
3.6.14-2	263/259	3/29/11
3.6.14-3	270/266	8/6/13

3.6.15-1	173/165	9/30/98
3.6.15-2	263/259	3/29/11
3.6.16-1	263/259	3/29/11
3.6.16-2	263/259	3/29/11
3.6.17-1	253/248	10/30/09
3.7.1-1	173/165	9/30/98
3.7.1-2	299/295	10/23/18
3.7.1-3	281/277	4/29/16
3.7.2-1	173/165	9/30/98
3.7.2-2	299/295	10/23/18
3.7.3-1	173/165	9/30/98
3.7.3-2	299/295	10/23/18
3.7.4-1	294/290	10/23/17
3.7.4-2	263/259	3/29/11
3.7.5-1	295/291	10/23/17
3.7.5-2	173/165	9/30/98
3.7.5-3	299/295	10/23/18
3.7.5-4	263/259	3/29/11
3.7.6-1	294/290	10/23/17
3.7.6-2	263/259	3/29/11
3.7.7-1	253/248	10/30/09
3.7.7-2	263/259	3/29/11
3.7.8-1	271/267	08/09/13
3.7.8-2	271/267	08/09/13
3.7.8-3	271/267	08/09/13
3.7.8-4	271/267	08/09/13
3.7.9-1	263/259	3/29/11
3.7.9-2	263/259	3/29/11
3.7.10-1	250/245	7/30/09
3.7.10-2	260/255	8/9/10
3.7.10-3	289/285	5/08/17
3.7.11-1	198/191	4/23/02
3.7.11-2	263/259	3/29/11
3.7.12-1	253/248	10/30/09

3.7.12-2	289/285	5/08/17
3.7.13-1	198/191	4/23/02
3.7.13-2	289/285	5/08/17
3.7.14-1	263/259	3/29/11
3.7.15-1	263/259	3/29/11
3.7.16-1	233/229	9/27/06
3.7.16-2	233/229	9/27/06
3.7.16-3	233/229	9/27/06
3.7.17-1	263/259	3/29/11
3.8.1-1	253/248	10/30/09
3.8.1-2	173/165	9/30/98
3.8.1-3	253/248	10/30/09
3.8.1-4	173/165	9/30/98
3.8.1-5	263/259	3/29/11
3.8.1-6	263/259	3/29/11
3.8.1-7	263/259	3/29/11
3.8.1-8	263/259	3/29/11
3.8.1-9	292/288	9/08/17
3.8.1-10	263/259	3/29/11
3.8.1-11	263/259	3/29/11
3.8.1-12	292/288	9/08/17
3.8.1-13	292/288	9/08/17
3.8.1-14	292/288	9/08/17
3.8.1-15	263/259	3/29/11
3.8.2-1	173/165	9/30/98
3.8.2-2	207/201	7/29/03
3.8.2-3	173/165	9/30/98
3.8.3-1	175/167	1/15/99
3.8.3-2	263/259	3/29/11
3.8.3-3	263/259	3/29/11
3.8.4-1	173/165	9/30/98
3.8.4-2	263/259	3/29/11
3.8.4-3	292/288	9/08/17
3.8.4-4	292/288	9/08/17

3.8.4-5	262/258	12/20/10
3.8.5-1	173/165	9/30/98
3.8.5-2	207/201	7/29/03
3.8.6-1	253/248	10/30/09
3.8.6-2	253/248	10/30/09
3.8.6-3	253/248	10/30/09
3.8.6-4	263/259	3/29/11
3.8.6-5	223/218	4/27/05
3.8.7-1	173/165	9/30/98
3.8.7-2	263/259	3/29/11
3.8.8-1	173/165	9/30/98
3.8.8-2	263/259	3/29/11
3.8.9-1	173/165	9/30/98
3.8.9-2	173/165	9/30/98
3.8.9-3	263/259	3/29/11
3.8.10-1	207/201	7/29/03
3.8.10-2	263/259	3/29/11
3.9.1-1	263/259	3/29/11
3.9.2-1	215/209	6/21/04
3.9.2-2	263/259	3/29/11
3.9.3-1	227/222	9/30/05
3.9.3-2	263/259	3/29/11
3.9.4-1	207/201	7/29/03
3.9.4-2	297/293	1/4/18
3.9.5-1	293/289	9/29/17
3.9.5-2	297/293	1/4/18
3.9.6-1	263/259	3/29/11
3.9.7-1	263/259	3/29/11
4.0-1	284/280	6/21/16
4.0-2	233/229	9/27/06
5.1-1	273/269	2/12/15
5.2-1	273/269	2/12/15
5.2-2	273/269	2/12/15
5.2-3	Deleted	9/21/09

5.3-1	273/269	2/12/15
5.4-1	173/165	9/30/98
5.5-1	286/282	9/12/16
5.5-2	286/282	9/12/16
5.5-3	173/165	9/30/98
5.5-4	173/165	9/30/98
5.5-5	216/210	8/5/04
5.5-6	299/295	10/23/18
5.5-7	280/276	4/26/16
5.5-8	280/276	4/26/16
5.5-9	280/276	4/26/16
5.5-10	280/276	4/26/16
5.5-11	280/276	4/26/16
5.5-12	280/276	4/26/16
5.5-13	280/276	4/26/16
5.5-14	280/276	4/26/16
5.5-15	280/276	4/26/16
5.5-16	280/276	4/26/16
5.5-17	280/276	4/26/16
5.5-18	280/276	4/26/16
5.5-19	280/276	4/26/16
5.6-1	222/217	3/31/05
5.6-2	253/248	10/30/09
5.6-3	222/217	3/31/05
5.6-4	284/280	6/21/16
5.6-5	275/271	4/14/15
5.6-6	280/276	4/26/16
5.7-1	273/269	2/12/15
5.7-2	173/165	9/30/98

BASES

i	Revision 1	4/08/99
ii	Revision 2	3/01/05
iii	Revision 1	6/21/04
B 2.1.1-1	Revision 0	9/30/98
B 2.1.1-2	Revision 1	12/19/03
B 2.1.1-3	Revision 1	12/19/03
B 2.1.2-1	Revision 0	9/30/98
B 2.1.2-2	Revision 0	9/30/98
B 2.1.2-3	Revision 0	9/30/98
B 3.0-1 thru B 3.0-21	Revision 6	10/23/18
B 3.1.1-1 thru B 3.1.1-6	Revision 3	5/05/11
B 3.1.2-1 thru B 3.1.2-5	Revision 3	11/14/17
B 3.1.3-1 thru B 3.1.3-6	Revision 2	4/14/15
B 3.1.4-1 thru B 3.1.4-9	Revision 1	5/05/11
B 3.1.5-1 thru B 3.1.5-4	Revision 2	5/05/11
B 3.1.6-1 thru B 3.1.6-6	Revision 1	5/05/11
B 3.1.7-1	Revision 0	9/30/98
B 3.1.7-2	Revision 2	1/08/04
B 3.1.7-3	Revision 2	1/08/04
B 3.1.7-4	Revision 2	1/08/04
B 3.1.7-5	Revision 2	1/08/04
B 3.1.7-6	Revision 2	1/08/04
B 3.1.8-1 thru B 3.1.8-6	Revision 4	3/28/18
B 3.2.1-1 thru B 3.2.1.-11	Revision 4	5/05/11

B 3.2.2-1 thru B 3.2.2-10	Revision 3	5/05/11
B 3.2.3-1 thru B 3.2.3-4	Revision 2	5/05/11
B 3.2.4-1 thru B 3.2.4-7	Revision 2	5/05/11
B 3.3.1-1 thru B.3.3.1-55	Revision 8	4/08/16
B 3.3.2-1 thru B 3.3.2-50	Revision 12	12/18/15
B 3.3.3-1 thru B.3.3.3-16	Revision 6	4/11/14
B 3.3.4-1 thru B 3.3.4-5	Revision 2	5/05/11
B 3.3.5-1 thru B 3.3.5-6	Revision 3	12/18/15
B 3.3.6-1 thru B 3.3.6-5	Revision 6	08/02/12
B 3.3.9-1 thru B 3.3.9-5	Revision 3	06/02/14
B 3.4.1-1 thru B 3.4.1-5	Revision 3	5/05/11
B 3.4.2-1	Revision 0	9/30/98
B 3.4.2-2	Revision 0	9/30/98
B 3.4.2-3	Revision 0	9/30/98
B 3.4.3-1 thru B 3.4.3-6	Revision 2	5/05/11
B 3.4.4-1 thru B 3.4.4-3	Revision 2	5/05/11
B 3.4.5-1 thru B 3.4.5-6	Revision 3	5/05/11
B 3.4.6-1 thru B 3.4.6-6	Revision 5	4/26/17

B 3.4.7-1 thru B 3.4.7-7	Revision 7	4/26/17
B 3.4.8-1 thru B 3.4.8-4	Revision 4	4/26/17
B 3.4.9-1 thru B 3.4.9-5	Revision 3	08/02/12
B 3.4.10-1 thru B 3.4.10-4	Revision 4	10/23/18
B 3.4.11-1 thru B 3.4.11-7	Revision 4	5/05/11
B 3.4.12-1 thru B 3.4.12-13	Revision 6	10/23/18
B 3.4.13-1 thru B 3.4.13-7	Revision 7	3/15/12
B 3.4.14-1 thru B 3.4.14-6	Revision 3	5/05/11
B 3.4.15-1 thru B 3.4.15-10	Revision 6	5/05/11
B 3.4.16-1 thru B 3.4.16-5	Revision 4	10/23/12
B 3.4.17-1 thru B 3.4.17-3	Revision 2	5/05/11
B 3.4.18-1 thru B 3.4.18-8	Revision 2	4/26/16
B 3.5.1-1 thru B 3.5.1-8	Revision 4	4/26/17
B 3.5.2-1 thru B 3.5.2-11	Revision 5	10/23/18
B 3.5.3-1 thru B 3.5.3-3	Revision 2	4/26/17
B 3.5.4-1 thru B.3.5.4-5	Revision 5	4/11/14
B 3.5.5-1 thru B 3.5.5-4	Revision 1	5/05/11

B 3.6.1-1	Revision 1	7/31/01
B 3.6.1-2	Revision 1	7/31/01
B 3.6.1-3	Revision 1	7/31/01
B 3.6.1-4	Revision 1	7/31/01
B 3.6.1-5	Revision 1	7/31/01
B 3.6.2-1 thru B 3.6.2-8	Revision 2	5/05/11
B 3.6.3-1 thru B 3.6.3-14	Revision 7	10/23/18
B 3.6.4-1 thru B 3.6.4-4	Revision 2	5/05/11
B 3.6.5-1 thru B 3.6.5-4	Revision 3	07/27/13
B 3.6.6-1 thru B 3.6.6-8	Revision 8	10/23/18
B 3.6.8-1 thru B 3.6.8-5	Revision 3	5/05/11
B 3.6.9-1 thru B 3.6.9-5	Revision 6	5/05/11
B 3.6.10-1 thru B 3.6.10-6	Revision 3	9/05/17
B 3.6.11-1 thru B 3.6.11-6	Revision 5	5/05/11
B 3.6.12-1 thru B 3.6.12-11	Revision 5	5/05/11
B 3.6.13-1 thru B 3.6.13-9	Revision 4	5/05/11
B 3.6.14-1 thru B 3.6.14-5	Revision 2	4/11/14
B 3.6.15-1 thru B 3.6.15-4	Revision 1	5/05/11
B 3.6.16-1 thru B 3.6.16-4	Revision 3	5/05/11
B 3.6.17-1	Revision 1	3/13/08

B 3.6.17-2	Revision 0	9/30/98
B 3.6.17-3	Revision 0	9/30/98
B 3.6.17-4	Revision 0	9/30/98
B 3.6.17-5	Revision 1	3/13/08
B 3.7.1-1 thru 3.7.1-5	Revision 3	10/23/18
B 3.7.2-1 thru B 3.7.2-5	Revision 4	10/23/18
B 3.7.3-1 B 3.7.3-6	Revision 3	10/23/18
B 3.7.4-1 thru B 3.7.4-4	Revision 3	11/14/17
B 3.7.5-1 thru B 3.7.5-9	Revision 5	10/23/18
B 3.7.6-1 thru B 3.7.6-3	Revision 6	9/10/18
B 3.7.7-1 thru B 3.7.7-5	Revision 2	5/05/11
B 3.7.8-1 thru B 3.7.8-8	Revision 5	8/09/13
B 3.7.9-1 thru B 3.7.9-4	Revision 3	5/05/11
B 3.7.10-1 thru B 3.7.10-9	Revision 11	9/05/17
B 3.7.11-1 thru B 3.7.11-4	Revision 3	10/24/11
B 3.7.12-1 thru B 3.7.12-7	Revision 8	3/28/18
B 3.7.13-1 thru B 3.7.13-5	Revision 5	9/05/17
B 3.7.14-1 thru B 3.7.14-3	Revision 2	5/05/11
B 3.7.15-1 thru B 3.7.15-4	Revision 2	5/05/11

B 3.7.16-1	Revision 2	9/27/06
B 3.7.16-2	Revision 2	9/27/06
B 3.7.16-3	Revision 2	9/27/06
B 3.7.16-4	Revision 0	9/27/06
B 3.7.17-1 thru B 3.7.17-3	Revision 2	5/05/11
B 3.8.1-1 thru B.3.8.1-30	Revision 6	10/30/17
B 3.8.2-1	Revision 0	9/30/98
B 3.8.2-2	Revision 0	9/30/98
B 3.8.2-3	Revision 0	9/30/98
B 3.8.2-4	Revision 1	5/10/05
B 3.8.2-5	Revision 2	5/10/05
B 3.8.2-6	Revision 1	5/10/05
B 3.8.3-1 thru B 3.8.3-8	Revision 4	5/05/11
B 3.8.4-1 thru B3.8.4.11	Revision 11	10/30/17
B 3.8.5-1	Revision 0	9/30/98
B 3.8.5-2	Revision 2	7/29/03
B 3.8.5-3	Revision 1	7/29/03
B 3.8.6-1 thru B 3.8.6-7	Revision 4	5/05/11
B 3.8.7-1 thru B 3.8.7-4	Revision 3	5/05/11
B 3.8.8-1 thru B 3.8.8-4	Revision 3	5/05/11
B 3.8.9-1 thru B 3.8.9-10	Revision 2	5/05/11
B 3.8.10-1 thru B 3.8.10-4	Revision 3	5/05/11
B 3.9.1-1 thru B 3.9.1-4	Revision 3	5/05/11

B 3.9.2-1 thru B 3.9.2-3	Revision 6	3/21/17
B 3.9.3-1 thru B 3.9.3-5	Revision 4	5/05/11
B 3.9.4-1 thru B 3.9.4-6	Revision 6	1/23/18
B 3.9.5-1 thru B 3.9.5-5	Revision 6	1/23/18
B 3.9.6-1 thru B 3.9.6-3	Revision 2	5/05/11
B 3.9.7-1 thru B 3.9.7-3	Revision 1	5/05/11

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs

LCO 3.0.1 through LCO 3.0.10 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

LCO 3.0.1

LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered, unless otherwise specified. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

LCO 3.0.2 (continued)

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, modifications, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Alternatives that would not result in redundant equipment being inoperable should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time other conditions exist which result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable, and the ACTIONS Condition(s) are entered.

BASES

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:

a. An associated Required Action and Completion Time is not met and no other Condition applies; or

b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The LCO phrase, "Action shall be initiated within 1 hour..." does not mean that an actual change in load must be commenced by the end of the 1-hour period (Reference 1). The action initiated at the end of the 1-hour period may be administrative in nature, such as preparing shutdown procedures. If at the end of 1 hour, corrective measures which would allow exiting LCO 3.0.3 are not complete, but there is reasonable assurance that they will be completed with enough time remaining to still allow for an orderly unit shutdown, if required, commencing a load decrease may be delayed until that time. The time limits specified to enter lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

BASES

LCO 3.0.3 (continued)

A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met,

b. The LCO is no longer applicable,

c. A Condition exists for which the Required Actions have now been performed, or

d. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

The time limits of Specification 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for entering the next lower MODE applies. If a lower MODE is entered in less time than allowed, however, the total allowable time to enter MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is entered in 2 hours, then the time allowed for entering MODE 4 is the next 11 hours, because the total time for entering MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to enter a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

BASES

LCO 3.0.3 (continued)

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.14, "Spent Fuel Pool (SFP) Water Level." LCO 3.7.14 has an Applicability of "During movement of irradiated fuel assemblies in the spent fuel pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.14 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.14 of "Suspend movement of irradiated fuel assemblies in the spent fuel pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It allows placing the unit in a MODE or other specified condition stated in that Applicability (e.g., the Applicability desired to be entered) when unit conditions are such that the requirements of the LCO would not be met, in accordance with either LCO 3.0.4.a, LCO 3.0.4.b, or LCO 3.0.4.c.

LCO 3.0.4.a allows entry into a MODE or other specified condition in the Applicability with the LCO not met when the associated ACTIONS to be entered following entry into the MODE or other specified condition in the Applicability will permit continued operation within the MODE or other specified condition for an unlimited period of time. Compliance with ACTIONS that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made and the Required Actions followed after entry into the Applicability.

For example, LCO 3.0.4.a may be used when the Required Action to be entered states that an inoperable instrument channel must be placed in the trip condition within the Completion Time. Transition into a MODE or other specified condition in the Applicability may be made in accordance with LCO 3.0.4 and the channel is subsequently placed in the tripped condition within the Completion Time, which begins when the Applicability is entered. If the instrument channel cannot be placed in the tripped condition and the subsequent default ACTION ("Required Action and associated Completion Time not met") allows the OPERABLE train to be placed in operation, use of LCO 3.0.4.a is acceptable because the subsequent ACTIONS to be entered following entry into the MODE

BASES

LCO 3.0.4 (continued)

include ACTIONS (place the OPERABLE train in operation) that permit safe plant operation for an unlimited period of time in the MODE or other specified condition to be entered.

LCO 3.0.4.b allows entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate

The risk assessment may use quantitative, qualitative, or blended approaches, and the risk assessment will be conducted using the plant program, procedures, and criteria in place to implement 10 CFR 50.65(a)(4), which requires that risk impacts of maintenance activities be assessed and managed. The risk assessment, for the purposes of LCO 3.0.4.b, must take into account all inoperable Technical Specification equipment regardless of whether the equipment is included in the normal 10 CFR 50.65(a)(4) risk assessment scope. The risk assessments will be conducted using the procedures and guidance endorsed by Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." These documents address general guidance for conduct of the risk assessment, quantitative and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to plan and conduct other activities in a manner that controls overall risk, increased risk awareness by shift and management personnel, actions to reduce the duration of the condition, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed MODE change is acceptable. Consideration should also be given to the probability of completing restoration such that the requirements of the LCO would be met prior to the expiration of ACTIONS Completion Times that would require exiting the Applicability.

LCO 3.0.4.b may be used with single, or multiple systems and components unavailable. NUMARC 93-01 provides guidance relative to consideration of simultaneous unavailability of multiple systems and components.

The results of the risk assessment shall be considered in determining the acceptability of entering the MODE or other specified condition in the Applicability, and any corresponding risk management actions. The LCO3.0.4.b risk assessments do not have to be documented.

BASES

LCO 3.0.4 (continued)

The Technical Specifications allow continued operation with equipment unavailable in MODE 1 for the duration of the Completion Time. Since this is allowable, and since in general the risk impact in that particular MODE bounds the risk of transitioning into and through the applicable MODES or other specified conditions in the Applicability of the LCO, the use of the LCO 3.0.4.b allowance should be generally acceptable, as long as the risk is assessed and managed as stated above. However, there is a small subset of systems and components that have been determined to be more important to risk and use of the LCO 3.0.4.b allowance is prohibited. The LCOs governing these systems and components contain Notes prohibiting the use of LCO 3.0.4.b by stating that LCO 3.0.4.b is not applicable.

LCO 3.0.4.c allows entry into a MODE or other specified condition in the Applicability with the LCO not met based on a Note in the Specification which states LCO 3.0.4.c is applicable. These specific allowances permit entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time and a risk assessment has not been performed. This allowance may apply to all the ACTIONS or to a specific Required Action of a Specification. The risk assessments performed to justify the use of LCO 3.0.4.b usually only consider systems and components. For this reason, LCO 3.0.4.c is typically applied to Specifications which describe values and parameters (e.g., RCS Specific Activity).

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

LCO 3.0.4 (continued)

Upon entry into a MODE or other specified condition in the Applicability with the LCO not met, LCO 3.0.1 and LCO 3.0.2 require entry into the applicable Conditions and Required Actions until the condition is resolved, until the LCO is met, or until the unit is not within the Applicability of the Technical Specification.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, utilizing LCO 3.0.4 is not a violation of SR 3.0.1 or SR 3.0.4 for any Surveillances that have not been performed on inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of SRs to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed SRs. This Specification does not provide time to perform any other preventive or corrective maintenance. .LCO 3.0.5 should not be used in lieu of other practicable alternatives that comply with Required Actions and that do not require changing the MODE or other specified conditions in the Applicability in order to demonstrate equipment is OPERABLE. LCO 3.0.5 is not intended to be used repeatedly.

An example of demonstrating equipment is OPERABLE with the Required Actions not met is opening a manual valve that was closed to comply with Required Actions to isolate a flowpath with excessive Reactor Coolant System {RCS) Pressure Isolation Valve {PIV) leakage in order to perform testing to demonstrate that RCS PIV leakage is now within limit.

Examples of demonstrating equipment OPERABILITY include instances in which it is necessary to take an inoperable channel or trip system out of a tripped condition that was directed by a Required Action, if there is

LCO 3.0.5 (continued)

no Required Action Note for this purpose. An example of verifying OPERABILITY of equipment removed from service is taking a tripped channel out of the tripped condition to permit the logic to function and indicate the appropriate response during performance of required testing on the inoperable channel. Examples of demonstrating the OPERABILITY of other equipment are taking an inoperable channel or trip system out of the tripped condition 1) to prevent the trip function from occurring during the performance of required testing on another channel in the other trip system, or 2) to permit the logic to function and indicate the appropriate response during the performance of required testing on another channel in the same trip system.

The administrative controls in LCO 3.0.5 apply in all cases to systems or components in Chapter 3 of the Technical Specifications, as long as the testing could not be conducted while complying with the Required Actions. This includes the realignment or repositioning of redundant or alternate equipment or trains previously manipulated to comply with ACTIONS, as well as equipment removed from service or declared inoperable to comply with ACTIONS.

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

BASES

LCO 3.0.6 (continued)

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.15, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCOs 3.1.8 and 3.4.17 allow specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

BASES

LCO 3.0.7 (continued)

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

LCO 3.0.8

LCO 3.0.8 establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s). This LCO states that the supported system is not considered to be inoperable solely due to one or more required snubbers not capable of performing their associated support function(s). This is appropriate because a limited length of time is allowed for maintenance, testing, or repair of one or more required snubbers not capable of performing their associated support function(s) and appropriate compensatory measures are specified in the snubber requirements, which are located outside of the Technical Specifications (TS) under licensee control. The snubber requirements do not meet the criteria in 10 CFR 50.36(c)(2)(ii), and, as such, are appropriate for control by the licensee.

If the allowed time expires and the required snubber(s) are unable to perform their associated support function(s), the affected supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.

LCO 3.0.8.a applies when one or more required snubbers are not capable of providing their associated support function(s) to a single train or subsystem of a multiple train or subsystem supported system or to a single train or subsystem supported system. LCO 3.0.8.a allows 72 hours to restore the required snubber(s) before declaring the supported system inoperable. The 72 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the required snubber(s) are not capable of performing their associated support function and due to the availability of the redundant train of the supported system.

LCO 3.0.8.b applies when one or more required snubbers are not capable of providing their associated support function(s) to more than one train or subsystem of a multiple train or subsystem supported system. LCO 3.0.8.b allows 12 hours to restore the required snubber(s)

BASES

LCO 3.0.8 (continued)

before declaring the supported system inoperable. The 12 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the required snubber(s) are not capable of performing their associated support function.

LCO 3.0.8 requires that risk be assessed and managed. Industry and NRC guidance on the implementation of 10 CFR 50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of LCO 3.0.8 should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. The risk assessment need not be quantified, but may be a qualitative awareness of the vulnerability of systems and components when one or more required snubbers are not able to perform their associated support function.

LCO 3.0.9

LCO 3.0.9 delineates the applicability of each specification to Unit 1 and Unit 2 operations.

LCO 3.0.10

LCO 3.0.10 establishes conditions under which systems described in the Technical Specifications are considered to remain OPERABLE when required barriers are not capable of providing their related support function(s).

As stated in NEI 04-08, "Allowance for Non Technical Specification Barrier Degradation on Supported System OPERABILITY (TSTF-427) Industry Implementation Guidance," March 2006, if the inability of a barrier to perform its support function does not render a supported system governed by the Technical Specifications inoperable (see NRC Regulatory Issues Summary 2001-09, Control of Hazard Barriers, dated April 2, 2001), the provisions of LCO 3.0.10 are not necessary, as the supported system is Operable.

Barriers are doors, walls, floor plugs, curbs, hatches, installed structures or components, or other devices, not explicitly described in Technical Specifications, that support the performance of the safety function of systems described in the Technical Specifications. This LCO states that the supported system is not considered to be inoperable solely due to required barriers not capable of performing their related support function(s) under the described conditions. LCO 3.0.10 allows 30 days before declaring the supported system(s) inoperable and the LCO(s)

LCO 3.0.10 (continued)

associated with the supported system(s) not met. A maximum time is placed on each use of this allowance to ensure that as required barriers are found or are otherwise made unavailable, they are restored. However, the allowable duration may be less than the specified maximum time based on the risk assessment.

If the allowed time expires and the barriers are unable to perform their related support function(s), the supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.

This provision does not apply to barriers which support ventilation systems or to fire barriers. The Technical Specifications for ventilation systems provide specific Conditions for inoperable barriers. Fire barriers are addressed by other regulatory requirements and associated plant programs. This provision does not apply to barriers which are not required to support system OPERABILITY (see NRC Regulatory Issue Summary 2001-09, "Control of Hazard Barriers," dated April 2, 2001).

The provisions of LCO 3.0.10 are justified because of the low risk associated with required barriers not being capable of performing their related support function. This provision is based on consideration of the following initiating event categories:

- Loss of coolant accidents;
- High energy line breaks;
- Feedwater line breaks;
- Internal flooding;
- External flooding;
- Turbine missile ejection; and
- Tornado or high wind.

The risk impact of the barriers which cannot perform their related support function(s) must be addressed pursuant to the risk assessment and management provision of the Maintenance Rule, 10 CFR 50.65 (a)(4), and the associated implementation guidance, Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Regulatory Guide 1.160 endorses the guidance in Section 11 of NUMARC 93-01, Revision 4A, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." This guidance provides for the consideration of dynamic plant configuration issues, emergent conditions, and other aspects pertinent to plant operation with the barriers unable to perform their related support function(s). These considerations may result in risk management and other compensatory actions being required during the period that barriers are unable to perform their related support function(s).

BASES

LCO 3.0.10 (continued)

LCO 3.0.10 may be applied to one or more trains or subsystems of a system supported by barriers that cannot provide their related support function(s), provided that risk is assessed and managed (including consideration of the effects on Large Early Release and from external events). If applied concurrently to more than one train or subsystem of a multiple train or subsystem supported system, the barriers supporting each of these trains or subsystems must provide their related support function(s) for different categories of initiating events. For example, LCO 3.0.10 may be applied for up to 30 days for more than one train of a multiple train supported system if the affected barrier for one train protects against internal flooding and the affected barrier for the other train protects against tornado missiles. In this example, the affected barrier may be the same physical barrier but serve different protection functions for each train. If during the time that LCO 3.0.10 is being used, the required OPERABLE train or subsystem becomes inoperable, it must be restored to OPERABLE status within 24 hours. Otherwise, the train(s) or subsystem(s) supported by barriers that cannot perform their related support function(s) must be declared inoperable and the associated LCOs declared not met. This 24 hour period provides time to respond to emergent conditions that would otherwise likely lead to entry into LCO 3.0.3 and a rapid plant shutdown, which is not justified given the low probability of an initiating event which would require the barrier(s) not capable of performing their related support function(s). During this 24 hour period, the plant risk associated with the existing conditions is assessed and managed in accordance with 10 CFR 50.65(a)(4).

BASES

REFERENCES

1. Letter from Christopher I. Grimes, Chief, Technical Specifications Branch, Division of Operating Reactor Support, to Frederick J. Hebdon, Director, Project Directorate II-4, Division of Reactor Projects I/II, "Use of Shutdown Times for Corrective Maintenance (TIA 92-08)," December 11, 1992.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 3.0.1 through SR 3.0.5 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known not to be met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a test exception are only applicable when the test exception is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

BASES

SR 3.0.1 (continued)

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. An example of where SR 3.0.2 does not apply is in the Containment Leakage Rate Testing Program. This program establishes testing requirements and Frequencies in accordance with the requirements of regulations. The TS cannot in and of themselves extend a test interval specified in the regulations.

BASES

SR 3.0.2 (continued)

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per ..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or the action accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been performed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to perform Surveillances that have been missed. This delay period permits the performance of a Surveillance before complying with Required Actions or other remedial measures that might preclude performance of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

BASES

SR 3.0.3 (continued)

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

SR 3.0.3 is only applicable if there is a reasonable expectation the associated equipment is OPERABLE or that variables are within limits, and it is expected that the Surveillance will be met when performed. Many factors should be considered, such as the period of time since the Surveillance was last performed, or whether the Surveillance, or a portion thereof, has ever been performed, and any other indications, tests, or activities that might support the expectation that the Surveillance will be met when performed. An example of the use of SR 3.0.3 would be a relay contact that was not tested as required in accordance with a particular SR, but previous successful performances of the SR included the relay contact; the adjacent, physically connected relay contacts were tested during the SR performance; the subject relay contact has been tested by another SR; or historical operation of the subject relay contact has been successful. It is not sufficient to infer the behavior of the associated equipment from the performance of similar equipment. The rigor of determining whether there is a reasonable expectation a Surveillance will be met when performed should increase based on the length of time since the last performance of the Surveillance. If the Surveillance has been performed recently, a review of the Surveillance history and equipment performance may be sufficient to support a reasonable expectation that the Surveillance will be met when performed. For Surveillances that have not been performed for a long period or that have never been performed, a rigorous evaluation based on objective evidence should provide a high degree of confidence that the equipment is OPERABLE. The evaluation should be documented in sufficient detail to allow a knowledgeable individual to understand the basis for the determination.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used repeatedly to extend Surveillance intervals. While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable

BASES

SR 3.0.3 (continued)

opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to

BASES

SR 3.0.4 (continued)

OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

A provision is included to allow entry into a MODE or other specified condition in the Applicability when an LCO is not met due to a Surveillance not being met in accordance with LCO 3.0.4.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes. SR 3.0.4 does not restrict changing MODES or other specified conditions of the Applicability when a Surveillance has not been performed within the specified Frequency, provided the requirement to declare the LCO not met has been delayed in accordance with SR 3.0.3.

The provisions of SR 3.0.4 shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the

BASES

SR 3.0.4 (continued)

Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO's Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note, as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

SR 3.0.5

SR 3.0.5 delineates the applicability of the surveillance activities to Unit 1 and Unit 2 operations.

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs

LCO 3.0.1 through LCO 3.0.10 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

LCO 3.0.1

LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered, unless otherwise specified. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

BASES

LCO 3.0.2 (continued)

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, modifications, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Alternatives that would not result in redundant equipment being inoperable should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time other conditions exist which result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable, and the ACTIONS Condition(s) are entered.

BASES

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:

a. An associated Required Action and Completion Time is not met and no other Condition applies; or

b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The LCO phrase, "Action shall be initiated within 1 hour..." does not mean that an actual change in load must be commenced by the end of the 1-hour period (Reference 1). The action initiated at the end of the 1-hour period may be administrative in nature, such as preparing shutdown procedures. If at the end of 1 hour, corrective measures which would allow exiting LCO 3.0.3 are not complete, but there is reasonable assurance that they will be completed with enough time remaining to still allow for an orderly unit shutdown, if required, commencing a load decrease may be delayed until that time. The time limits specified to enter lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

BASES

LCO 3.0.3 (continued)

A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met,

b. The LCO is no longer applicable,

c. A Condition exists for which the Required Actions have now been performed, or

d. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

The time limits of Specification 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for entering the next lower MODE applies. If a lower MODE is entered in less time than allowed, however, the total allowable time to enter MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is entered in 2 hours, then the time allowed for entering MODE 4 is the next 11 hours, because the total time for entering MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to enter a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

BASES

LCO 3.0.3 (continued)

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.14, "Spent Fuel Pool (SFP) Water Level." LCO 3.7.14 has an Applicability of "During movement of irradiated fuel assemblies in the spent fuel pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.14 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.14 of "Suspend movement of irradiated fuel assemblies in the spent fuel pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It allows placing the unit in a MODE or other specified condition stated in that Applicability (e.g., the Applicability desired to be entered) when unit conditions are such that the requirements of the LCO would not be met, in accordance with either LCO 3.0.4.a, LCO 3.0.4.b, or LCO 3.0.4.c.

LCO 3.0.4.a allows entry into a MODE or other specified condition in the Applicability with the LCO not met when the associated ACTIONS to be entered following entry into the MODE or other specified condition in the Applicability will permit continued operation within the MODE or other specified condition for an unlimited period of time. Compliance with ACTIONS that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made and the Required Actions followed after entry into the Applicability.

For example, LCO 3.0.4.a may be used when the Required Action to be entered states that an inoperable instrument channel must be placed in the trip condition within the Completion Time. Transition into a MODE or other specified condition in the Applicability may be made in accordance with LCO 3.0.4 and the channel is subsequently placed in the tripped condition within the Completion Time, which begins when the Applicability is entered. If the instrument channel cannot be placed in the tripped condition and the subsequent default ACTION ("Required Action and associated Completion Time not met") allows the OPERABLE train to be placed in operation, use of LCO 3.0.4.a is acceptable because the subsequent ACTIONS to be entered following entry into the MODE

BASES

LCO 3.0.4 (continued)

include ACTIONS (place the OPERABLE train in operation) that permit safe plant operation for an unlimited period of time in the MODE or other specified condition to be entered.

LCO 3.0.4.b allows entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate

The risk assessment may use quantitative, qualitative, or blended approaches, and the risk assessment will be conducted using the plant program, procedures, and criteria in place to implement 10 CFR 50.65(a)(4), which requires that risk impacts of maintenance activities be assessed and managed. The risk assessment, for the purposes of LCO 3.0.4.b, must take into account all inoperable Technical Specification equipment regardless of whether the equipment is included in the normal 10 CFR 50.65(a)(4) risk assessment scope. The risk assessments will be conducted using the procedures and guidance endorsed by Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." These documents address general guidance for conduct of the risk assessment, quantitative and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to plan and conduct other activities in a manner that controls overall risk, increased risk awareness by shift and management personnel, actions to reduce the duration of the condition, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed MODE change is acceptable. Consideration should also be given to the probability of completing restoration such that the requirements of the LCO would be met prior to the expiration of ACTIONS Completion Times that would require exiting the Applicability.

LCO 3.0.4.b may be used with single, or multiple systems and components unavailable. NUMARC 93-01 provides guidance relative to consideration of simultaneous unavailability of multiple systems and components.

The results of the risk assessment shall be considered in determining the acceptability of entering the MODE or other specified condition in the Applicability, and any corresponding risk management actions. The LCO3.0.4.b risk assessments do not have to be documented.

BASES

LCO 3.0.4 (continued)

The Technical Specifications allow continued operation with equipment unavailable in MODE 1 for the duration of the Completion Time. Since this is allowable, and since in general the risk impact in that particular MODE bounds the risk of transitioning into and through the applicable MODES or other specified conditions in the Applicability of the LCO, the use of the LCO 3.0.4.b allowance should be generally acceptable, as long as the risk is assessed and managed as stated above. However, there is a small subset of systems and components that have been determined to be more important to risk and use of the LCO 3.0.4.b allowance is prohibited. The LCOs governing these systems and components contain Notes prohibiting the use of LCO 3.0.4.b by stating that LCO 3.0.4.b is not applicable.

LCO 3.0.4.c allows entry into a MODE or other specified condition in the Applicability with the LCO not met based on a Note in the Specification which states LCO 3.0.4.c is applicable. These specific allowances permit entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time and a risk assessment has not been performed. This allowance may apply to all the ACTIONS or to a specific Required Action of a Specification. The risk assessments performed to justify the use of LCO 3.0.4.b usually only consider systems and components. For this reason, LCO 3.0.4.c is typically applied to Specifications which describe values and parameters (e.g., RCS Specific Activity).

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

LCO 3.0.4 (continued)

Upon entry into a MODE or other specified condition in the Applicability with the LCO not met, LCO 3.0.1 and LCO 3.0.2 require entry into the applicable Conditions and Required Actions until the condition is resolved, until the LCO is met, or until the unit is not within the Applicability of the Technical Specification.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, utilizing LCO 3.0.4 is not a violation of SR 3.0.1 or SR 3.0.4 for any Surveillances that have not been performed on inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of SRs to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed SRs. This Specification does not provide time to perform any other preventive or corrective maintenance. .LCO 3.0.5 should not be used in lieu of other practicable alternatives that comply with Required Actions and that do not require changing the MODE or other specified conditions in the Applicability in order to demonstrate equipment is OPERABLE. LCO 3.0.5 is not intended to be used repeatedly.

An example of demonstrating equipment is OPERABLE with the Required Actions not met is opening a manual valve that was closed to comply with Required Actions to isolate a flowpath with excessive Reactor Coolant System {RCS) Pressure Isolation Valve {PIV) leakage in order to perform testing to demonstrate that RCS PIV leakage is now within limit.

Examples of demonstrating equipment OPERABILITY include instances in which it is necessary to take an inoperable channel or trip system out of a tripped condition that was directed by a Required Action, if there is

BASES

LCO 3.0.5 (continued)

no Required Action Note for this purpose. An example of verifying OPERABILITY of equipment removed from service is taking a tripped channel out of the tripped condition to permit the logic to function and indicate the appropriate response during performance of required testing on the inoperable channel. Examples of demonstrating the OPERABILITY of other equipment are taking an inoperable channel or trip system out of the tripped condition 1) to prevent the trip function from occurring during the performance of required testing on another channel in the other trip system, or 2) to permit the logic to function and indicate the appropriate response during the performance of required testing on another channel in the same trip system.

The administrative controls in LCO 3.0.5 apply in all cases to systems or components in Chapter 3 of the Technical Specifications, as long as the testing could not be conducted while complying with the Required Actions. This includes the realignment or repositioning of redundant or alternate equipment or trains previously manipulated to comply with ACTIONS, as well as equipment removed from service or declared inoperable to comply with ACTIONS.

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

LCO 3.0.6 (continued)

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.15, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCOs 3.1.8 and 3.4.17 allow specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

BASES

LCO 3.0.7 (continued)

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

LCO 3.0.8

LCO 3.0.8 establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s). This LCO states that the supported system is not considered to be inoperable solely due to one or more required snubbers not capable of performing their associated support function(s). This is appropriate because a limited length of time is allowed for maintenance, testing, or repair of one or more required snubbers not capable of performing their associated support function(s) and appropriate compensatory measures are specified in the snubber requirements, which are located outside of the Technical Specifications (TS) under licensee control. The snubber requirements do not meet the criteria in 10 CFR 50.36(c)(2)(ii), and, as such, are appropriate for control by the licensee.

If the allowed time expires and the required snubber(s) are unable to perform their associated support function(s), the affected supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.

LCO 3.0.8.a applies when one or more required snubbers are not capable of providing their associated support function(s) to a single train or subsystem of a multiple train or subsystem supported system or to a single train or subsystem supported system. LCO 3.0.8.a allows 72 hours to restore the required snubber(s) before declaring the supported system inoperable. The 72 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the required snubber(s) are not capable of performing their associated support function and due to the availability of the redundant train of the supported system.

LCO 3.0.8.b applies when one or more required snubbers are not capable of providing their associated support function(s) to more than one train or subsystem of a multiple train or subsystem supported system. LCO 3.0.8.b allows 12 hours to restore the required snubber(s)

BASES

LCO 3.0.8 (continued)

before declaring the supported system inoperable. The 12 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the required snubber(s) are not capable of performing their associated support function.

LCO 3.0.8 requires that risk be assessed and managed. Industry and NRC guidance on the implementation of 10 CFR 50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of LCO 3.0.8 should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. The risk assessment need not be quantified, but may be a qualitative awareness of the vulnerability of systems and components when one or more required snubbers are not able to perform their associated support function.

LCO 3.0.9

LCO 3.0.9 delineates the applicability of each specification to Unit 1 and Unit 2 operations.

LCO 3.0.10

LCO 3.0.10 establishes conditions under which systems described in the Technical Specifications are considered to remain OPERABLE when required barriers are not capable of providing their related support function(s).

As stated in NEI 04-08, "Allowance for Non Technical Specification Barrier Degradation on Supported System OPERABILITY (TSTF-427) Industry Implementation Guidance," March 2006, if the inability of a barrier to perform its support function does not render a supported system governed by the Technical Specifications inoperable (see NRC Regulatory Issues Summary 2001-09, Control of Hazard Barriers, dated April 2, 2001), the provisions of LCO 3.0.10 are not necessary, as the supported system is Operable..

Barriers are doors, walls, floor plugs, curbs, hatches, installed structures or components, or other devices, not explicitly described in Technical Specifications, that support the performance of the safety function of systems described in the Technical Specifications. This LCO states that the supported system is not considered to be inoperable solely due to required barriers not capable of performing their related support function(s) under the described conditions. LCO 3.0.10 allows 30 days before declaring the supported system(s) inoperable and the LCO(s)

BASES

LCO 3.0.10 (continued)

associated with the supported system(s) not met. A maximum time is placed on each use of this allowance to ensure that as required barriers are found or are otherwise made unavailable, they are restored. However, the allowable duration may be less than the specified maximum time based on the risk assessment.

If the allowed time expires and the barriers are unable to perform their related support function(s), the supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.

This provision does not apply to barriers which support ventilation systems or to fire barriers. The Technical Specifications for ventilation systems provide specific Conditions for inoperable barriers. Fire barriers are addressed by other regulatory requirements and associated plant programs. This provision does not apply to barriers which are not required to support system OPERABILITY (see NRC Regulatory Issue Summary 2001-09, "Control of Hazard Barriers," dated April 2, 2001).

The provisions of LCO 3.0.10 are justified because of the low risk associated with required barriers not being capable of performing their related support function. This provision is based on consideration of the following initiating event categories:

- Loss of coolant accidents;
- High energy line breaks;
- Feedwater line breaks;
- Internal flooding;
- External flooding;
- Turbine missile ejection; and
- Tornado or high wind.

The risk impact of the barriers which cannot perform their related support function(s) must be addressed pursuant to the risk assessment and management provision of the Maintenance Rule, 10 CFR 50.65 (a)(4), and the associated implementation guidance, Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Regulatory Guide 1.160 endorses the guidance in Section 11 of NUMARC 93-01, Revision 4A, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." This guidance provides for the consideration of dynamic plant configuration issues, emergent conditions, and other aspects pertinent to plant operation with the barriers unable to perform their related support function(s). These considerations may result in risk management and other compensatory actions being required during the period that barriers are unable to perform their related support function(s).

BASES

LCO 3.0.10 (continued)

LCO 3.0.10 may be applied to one or more trains or subsystems of a system supported by barriers that cannot provide their related support function(s), provided that risk is assessed and managed (including consideration of the effects on Large Early Release and from external

events). If applied concurrently to more than one train or subsystem of a multiple train or subsystem supported system, the barriers supporting each of these trains or subsystems must provide their related support function(s) for different categories of initiating events. For example, LCO 3.0.10 may be applied for up to 30 days for more than one train of a multiple train supported system if the affected barrier for one train protects against internal flooding and the affected barrier for the other train protects against tornado missiles. In this example, the affected barrier may be the same physical barrier but serve different protection functions for each train. If during the time that LCO 3.0.10 is being used, the required OPERABLE train or subsystem becomes inoperable, it must be restored to OPERABLE status within 24 hours. Otherwise, the train(s) or subsystem(s) supported by barriers that cannot perform their related support function(s) must be declared inoperable and the associated LCOs declared not met. This 24 hour period provides time to respond to emergent conditions that would otherwise likely lead to entry into LCO 3.0.3 and a rapid plant shutdown, which is not justified given the low probability of an initiating event which would require the barrier(s) not capable of performing their related support function(s). During this 24 hour period, the plant risk associated with the existing conditions is assessed and managed in accordance with 10 CFR 50.65(a)(4).

REFERENCES

1. Letter from Christopher I. Grimes, Chief, Technical Specifications Branch, Division of Operating Reactor Support, to Frederick J. Hebdon, Director, Project Directorate II-4, Division of Reactor Projects I/II, "Use of Shutdown Times for Corrective Maintenance (TIA 92-08)," December 11, 1992.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 3.0.1 through SR 3.0.5 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated. SR 3.0.2 and SR 3.0.3 apply in Chapter 5 only when invoked by a Chapter 5 Specification.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known not to be met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a test exception are only applicable when the test exception is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.

SR 3.0.1 (continued)

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

When a Section 5.5, "Programs and Manuals," specification states that the provisions of 3.0.2 are applicable, a 25% extension of the testing interval, whether stated in the specification or incorporated by reference, is permitted.

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs.

BASES

The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulation take precedence over the TS. Example of where SR 3.0.2 does not apply are the Containment Leakage Rate Testing Program required by 10 CFR 50, Appendix J, and the inservice testing of pumps and valves in accordance with applicable American Society of Mechanical Engineers Operation and Maintenance Code, as required by 10 CFR 50.55a. These programs establish testing requirements and Frequencies in accordance with the requirements of regulation. The TS cannot, in and of themselves, extend a test interval specified in the regulations directly or by reference.

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per ..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or the action accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been performed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

When a Section 5.5, "Programs and Manuals," specification states that the provisions of SR 3.0.3 are applicable, it permits the flexibility to defer declaring the testing requirement not met in accordance with SR 3.0.3 when the testing has not been completed within the testing interval;

(including the allowance of SR 3.0.2 if invoked by the Section 5.5 specification).

This delay period provides adequate time to perform Surveillances that have been missed. This delay period permits the performance of a Surveillance before complying with Required Actions or other remedial measures that might preclude performance of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

SR 3.0.3 is only applicable if there is a reasonable expectation the associated equipment is OPERABLE or that variables are within limits, and it is expected that the Surveillance will be met when performed. Many factors should be considered, such as the period of time since the Surveillance was last performed, or whether the Surveillance, or a portion thereof, has ever been performed, and any other indications, tests, or activities that might support the expectation that the Surveillance will be met when performed. An example of the use of SR 3.0.3 would be a relay contact that was not tested as required in accordance with a particular SR, but previous successful performances of the SR included the relay contact; the adjacent, physically connected relay contacts were tested during the SR performance; the subject relay contact has been tested by another SR; or historical operation of the subject relay contact has been successful. It is not sufficient to infer the behavior of the associated equipment from the performance of similar equipment. The rigor of determining whether there is a reasonable expectation a Surveillance will be met when performed should increase based on the length of time since the last performance of the Surveillance. If the Surveillance has been performed recently, a review of the Surveillance

BASES

SR 3.0.3 (continued)

history and equipment performance may be sufficient to support a reasonable expectation that the Surveillance will be met when performed. For Surveillances that have not been performed for a long period or that have never been performed, a rigorous evaluation based on objective evidence should provide a high degree of confidence that the equipment is OPERABLE. The evaluation should be documented in sufficient detail to allow a knowledgeable individual to understand the basis for the determination.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used repeatedly to extend Surveillance intervals. While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The

degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the

BASES

SR 3.0.3 (continued)

applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to

OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

A provision is included to allow entry into a MODE or other specified condition in the Applicability when an LCO is not met due to a Surveillance not being met in accordance with LCO 3.0.4.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes. SR 3.0.4 does not restrict changing MODES or other specified conditions of the Applicability when a Surveillance has not been performed within the specified Frequency, provided the requirement to declare the LCO not met has been delayed in accordance with SR 3.0.3.

The provisions of SR 3.0.4 shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent

BASES

SR 3.0.4 (continued)

changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the

Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO's Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note, as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

SR 3.0.5

SR 3.0.5 delineates the applicability of the surveillance activities to Unit 1 and Unit 2 operations.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 PHYSICS TESTS Exceptions

BASES

BACKGROUND

The primary purpose of the PHYSICS TESTS exceptions is to permit relaxations of existing LCOs to allow PHYSICS TESTS to be performed.

Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

The key objectives of a test program are to (Ref. 3):

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in the design and analysis;

c. Verify the assumptions used to predict unit response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and

e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality, during startup, during low power operations, during power ascension, at high power, and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed (Ref. 4).

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include all information necessary to permit a detailed execution of the testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long term power operation.

BACKGROUND (continued)

The PHYSICS TESTS required for reload fuel cycles (Ref. 4) are listed below:

a. Critical Boron Concentration—Control Rods Withdrawn;

b. Control Rod Worth;

c. Isothermal Temperature Coefficient (ITC); and

These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance.

APPLICABLE SAFETY ANALYSES

The fuel is protected by LCOs that preserve the initial conditions of the core assumed during the safety analyses. The methods for development of the LCOs that are excepted by this LCO are described in the Westinghouse Reload Safety Evaluation Methodology Report (Ref. 5). The above mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational problems, may require the operating control or process variables to deviate from their LCO limitations.

The UFSAR defines requirements for initial testing of the facility, including PHYSICS TESTS. UFSAR Section 14.3 summarizes the zero, low power, and power tests. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits for all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in LCO 3.1.3, "Moderator Temperature Coefficient (MTC)," LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to ≤ 5% RTP, the reactor coolant temperature is kept ≥ 541°F, and SDM is within limit specified in the COLR.

The PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. Among the process variables involved are AFD and QPTR, which represent initial conditions of the unit safety analyses. Also involved are the movable control components (control and shutdown rods), which are required to shut down the reactor. The limits for these variables are

APPLICABLE SAFETY ANALYSES (continued)

specified for each fuel cycle in the COLR. PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the components and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of 10 CFR 50.36 (Ref.6).

Reference 7 allows special test exceptions (STEs) to be included as part of the LCO that they affect. It was decided, however, to retain this STE as a separate LCO because it was less cumbersome and provided additional clarity.

LCO

This LCO allows the reactor parameters of MTC and minimum temperature for criticality to be outside their specified limits. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. One Power Range Neutron Flux Channel may be bypassed, reducing the number of required channels from "4" to "3." Operation beyond specified limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

The requirements of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 may be suspended, and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3, 6, and 16.e, may be reduced to "3" required channels, during the performance of PHYSICS TESTS provided:

a. RCS lowest loop average temperature is ≥ 541°F; and

b. SDM is within limit specified in the COLR.

APPLICABILITY

This LCO is applicable in MODE 2 when performing low power PHYSICS TESTS. The applicable PHYSICS TESTS are performed in MODE 2 at HZP.

ACTIONS

A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

ACTIONS (continued)

B.1

When THERMAL POWER is > 5% RTP, the only acceptable action is to open the reactor trip breakers (RTBs) to prevent operation of the reactor beyond its design limits. Immediately opening the RTBs will shut down the reactor and prevent operation of the reactor outside of its design limits.

C.1

When the RCS lowest T_{avg} is < 541°F, the appropriate action is to restore T_{avg} to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring T_{avg} to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with temperature below 541°F could violate the assumptions for accidents analyzed in the safety analyses.

D.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, for reaching MODE 3 in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

The power range and intermediate range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." A CHANNEL OPERATIONAL TEST is performed on each power range and intermediate range channel prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.8.2

Verification that the RCS lowest loop T_{avg} is ≥ 541°F will ensure that the unit is not operating in a condition that could invalidate the safety analyses. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.1.8.3

Verification that THERMAL POWER is ≤ 5% RTP will ensure that the plant is not operating in the condition that could invalidate the safety analyses. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.1.8.4

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.

2. 10 CFR 50.59.

3. Regulatory Guide 1.68, Revision 2, August, 1978.

4. ANSI/ANS-19.6.1-1985, December 13, 1985.

5. WCAP-9273-NP-A, Westinghouse Reload Safety Evaluation Methodology Report," July 1985.

6. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

7. WCAP-11618, including Addendum 1, April 1989.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. The pressurizer safety valves are totally enclosed pop type, spring loaded, self actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2735 psig, which is 110% of the design pressure.

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The relief capacity for each valve, 420,000 lb/hr, is based on postulated overpressure transient conditions resulting from a locked rotor. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves or increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, 4, and 5; however, in MODE 4, with one or more RCS cold leg temperatures ≤ 210°F, and MODE 5 and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

The upper pressure limit of +3% is consistent with the ASME requirement (Ref. 1) for lifting pressures above 1000 psig. The lower pressure limit of -2% is selected such that the minimum LCO lift pressure remains above the uncertainty adjusted high pressure reactor trip setpoint. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure. The consequences of exceeding the American Society of

BASES

BACKGROUND (continued)

Mechanical Engineers (ASME) pressure limit (Ref. 1) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

APPLICABLE SAFETY ANALYSES

All accident and safety analyses in the UFSAR (Ref. 2) that require safety valve actuation assume operation of three pressurizer safety valves to limit increases in RCS pressure. The overpressure protection analysis (Ref. 3) is also based on operation of three safety valves. Accidents that could result in overpressurization if not properly terminated include:

a. Uncontrolled rod withdrawal;

b. Loss of reactor coolant flow;

c. Loss of external electrical load;

d. Loss of normal feedwater;

e. Loss of all AC power to station auxiliaries;

f. Locked rotor; and

g. Turbine trip.

Detailed analyses of the above transients are contained in Reference 2. Compliance with this LCO is consistent with the design bases and accident analyses assumptions.

Pressurizer safety valves satisfy Criterion 3 of 10 CFR 50.36 (Ref. 5).

LCO

The three pressurizer safety valves are set to open at the RCS design pressure 2485 psig, and within the ASME specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper pressure tolerance limit of +3% is consistent with the ASME requirements (Ref. 1) for lifting pressures above 1000 psig. The lower pressure limit of -2% is selected such that the minimum LCO lift pressure remains above the uncertainty adjusted high pressure reactor trip setpoint.

BASES

LCO (continued)

The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

In MODES 1, 2, and 3, and portions of MODE 4 above the LTOP arming temperature, OPERABILITY of three valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 and portions of MODE 4 are conservatively included, although the listed accidents may not require the safety valves for protection.

The LCO is not applicable in MODE 4 when all RCS cold leg temperatures are ≤ 210°F or in MODE 5 because LTOP is provided. Overpressure protection is not required in MODE 6 with the reactor vessel head removed.

The Note allows entry into MODES 3 and 4 with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 54 hour exception is based on 18 hour outage time for each of the three valves. The 18 hour period is derived from operating experience that hot testing can be performed in this timeframe.

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS Overpressure Protection System. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

BASES

ACTIONS (continued)

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with any RCS cold leg temperatures ≤ 210°F within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. With any RCS cold leg temperatures at or below 210°F, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the INSERVICE TESTING PROGRAM. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is +3% and -2% of the nominal setpoint of 2485 psig for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
2. UFSAR, Chapter 15.
3. UFSAR, Section 5.2.
4. ASME Code for Operation and Maintenance of Nuclear Power Plants.
5. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

The LTOP System controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. This specification provides the maximum allowable actuation logic setpoints for the power operated relief valves (PORVs) and LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," provides the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the LTOP MODES.

The reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only while shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3 requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the specified limits.

This LCO provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires all but two pumps incapable of injection into the RCS, isolating the accumulators, and limiting reactor coolant pump operation at low temperatures. The pressure relief capacity requires two redundant RCS relief valves. One RCS relief valve is the overpressure protection device that acts to terminate an increasing pressure event.

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control

BASES

BACKGROUND (continued)

system deactivated or the safety injection (SI) actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve. If conditions require the use of more than one charging pump for makeup in the event of loss of inventory, then additional pumps can be made available through manual actions.

The LTOP System for pressure relief consists of two PORVs with reduced lift settings or two residual heat removal (RHR) suction relief valves or one PORV and one RHR suction relief valve. Two RCS relief valves are required for redundancy. One RCS relief valve has adequate relieving capability to keep from overpressurization for the required coolant input capability.

PORV Requirements

As designed for the LTOP System, each PORV is signaled to open if the RCS pressure reaches 400 psig (as left calibrated), allowable value ≤ 425 psig (as found), when the PORVS are in the "lo-press" mode of operation. If the PORVs are being used to meet the requirements of this Specification, then indicated RCS cold leg temperature is limited to ≥ 70°F in accordance with the LTOP analysis. When all Reactor Coolant Pumps are secured, this temperature is measured at the outlet of the residual heat removal heat exchanger. This location will provide the most conservative (lower) temperature measurement of water capable of being delivered into the Reactor Coolant System. The LTOP actuation logic monitors both RCS temperature and RCS pressure. The signals used to generate the pressure setpoints originate from the wide range pressure transmitters. The signals used to generate the temperature permissives originate from the wide range RTDs. Each signal is input to the appropriate NSSS protection system cabinet where it is converted to an internal signal and then input to a comparator to generate an actuation signal. If the indicated pressure meets or exceeds the calculated value, a PORV is signaled to open.

This Specification presents the PORV setpoints for LTOP. Having the setpoints of both valves within the limits ensures that the Reference 1 limits will not be exceeded in any analyzed event.

When a PORV is opened in an increasing pressure transient, the release of coolant will cause the pressure increase to slow and reverse. As the PORV releases coolant, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

BASES

BACKGROUND (continued)

RHR Suction Relief Valve Requirements

During LTOP MODES, the RHR system is operated for decay heat removal and low-pressure letdown control. Therefore, the RHR suction isolation valves (there are two suction isolation valves per line) are open in the piping from the RCS hot legs to the inlets of the RHR pumps. While these valves are open, the RHR suction relief valves are exposed to the RCS and are able to relieve pressure transients in the RCS.

The RHR suction isolation valves must be open with operator power removed to make the RHR suction relief valves OPERABLE for RCS overpressure mitigation. The RHR suction relief valves are spring loaded, bellows type water relief valve with pressure tolerances and accumulation limits established by Section III of the American Society of Mechanical Engineers (ASME) Code (Ref. 8) for Class 2 relief valves.

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 3) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, and 3, and in MODE 4 with RCS cold leg temperature exceeding 210°F, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. At about 210°F and below, overpressure prevention falls to two OPERABLE RCS relief valves. Each of these means has a limited overpressure relief capability.

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the P/T curves are revised, the LTOP System must be re-evaluated to ensure its functional requirements can still be met using the RCS relief valve method.

Any change to the RCS must be evaluated against the Reference 3 analyses to determine the impact of the change on the LTOP acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

Mass Input Type Transients

a. Inadvertent safety injection of one safety injection pump and one charging pump; or

b. Charging/letdown flow mismatch.

APPLICABLE SAFETY ANALYSES (continued)

Heat Input Type Transients

a. Inadvertent actuation of pressurizer heaters;

b. Loss of RHR cooling; or

c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

The following are required during the LTOP MODES to ensure that mass and heat input transients do not occur, which either of the LTOP overpressure protection means cannot handle:

a. Rendering all but two pumps incapable of injection;

b. Deactivating the accumulator discharge isolation valves in their closed positions;

c. Limiting RCP operation based on the existing temperature in the RCS cold legs; and

d. Disallowing start of an RCP if secondary temperature is more than 50°F above primary temperature in any one loop. LCO 3.4.6, "RCS Loops—MODE 4," and LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled," provide this protection.

The Reference 3 analyses demonstrate that one RCS relief valve can maintain RCS pressure below limits when any two pumps (charging and/or safety injection) are actuated. Thus, the LCO allows two pumps OPERABLE during the LTOP MODES. The LCO also requires the accumulators be isolated when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in LCO 3.4.3.

The isolated accumulators must have their discharge valves closed and power removed.

The restrictions on the number of RCPs in operation at a given temperature ensures that during a LTOP mass injection event that the pressure/temperature (P/T) limits of 10 CFR 50, Appendix G to protect the

BASES

APPLICABLE SAFETY ANALYSES (continued)

reactor vessel are not exceeded. During startup and shutdown, when the RCPs are operated, their induced flows create a pressure drop across the vessel. This pressure drop along with the difference in elevation between the beltline region and the instrumentation locations are additive to the peak pressure from the mass injection event.

The amount of the pressure at the reactor vessel beltline region from the RCPs is dependent on the number of RCPs operated. Adequate margin to prevent exceeding the P/T limits is assured by restricting the number of RCPs operated. Since LTOP events are basically acknowledged as being steady-state events, these RCP operating restrictions are designed to work with the LTOP setpoint to provide protection from exceeding the steady-state Appendix G P/T limits.

Fracture mechanics analyses established the temperature of LTOP Applicability at 210°F.

The consequences of a small break loss of coolant accident (LOCA) in LTOP MODE 4 conform to 10 CFR 50.46 and 10 CFR 50, Appendix K (Refs. 4 and 5), requirements by having a maximum of two pumps (charging and/or safety injection) OPERABLE and SI actuation enabled.

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the specified limit. The setpoints are derived by analyses that model the performance of the LTOP System, assuming the limiting LTOP transient of one charging pump and one safety injection pump injecting into the RCS. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensures the Reference 1 P/T limits will be met.

The PORV setpoints will be updated when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Revised limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," discuss these examinations.

APPLICABLE SAFETY ANALYSES (continued)

The PORVs are considered active components. Thus, the failure of one PORV is assumed to represent the worst case, single active failure.

RHR Suction Relief Valve Performance

The RHR suction relief valves do not have variable pressure and temperature lift setpoints like the PORVs. Analyses show that one RHR suction relief valve with a setpoint at or between 417 psig and 509 psig will pass flow greater than that required for the limiting LTOP transient while maintaining RCS pressure less than the P/T limit curve. Assuming all relief flow requirements during the limiting LTOP event, an RHR suction relief valve will maintain RCS pressure to within the valve rated lift setpoint, plus an accumulation ≤ 10% of the rated lift setpoint.

Although each RHR suction relief valve may itself meet single failure criteria, its inclusion and location within the RHR system does not allow it to meet single failure criteria when spurious RHR suction isolation valve closure is postulated. Also, as the RCS P/T limits are decreased to reflect the loss of embrittlement, the RHR suction relief valves must be analyzed to still accommodate the design basis transients for LTOP.

The RHR suction relief valves are considered to be active components. Thus, the failure of one valve is assumed to represent the worst case single active failure.

The LTOP System satisfies Criterion 2 of 10 CFR 50.36 (Ref. 6).

LCO

This LCO requires that the LTOP System is OPERABLE. The LTOP System is OPERABLE when the minimum coolant input and pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

To limit the coolant input capability, the LCO permits a maximum of two pumps (charging and/or safety injection) capable of injecting into the RCS and requires all accumulator discharge isolation valves closed and immobilized when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in LCO 3.4.3. The LCO also limits RCP operation based on existing RCS cold leg temperature as required by the LTOP analysis.

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

BASES

LCO (continued)

a. Two OPERABLE PORVs (NC-32B and NC-34A); or

A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set to the specified limit and testing proves its automatic ability to open at this setpoint, and motive power is available to the valve and its control circuit. The following restrictions are placed on PORV OPERABILITY for LTOP due to commonalities between the PORV power supplies and letdown isolation:

- NC-32B is not OPERABLE for LTOP if excess letdown is in service.
- NC-32B is not OPERABLE for LTOP if normal letdown is in service and centrifugal charging pump B is in operation.
- NC-34A is not OPERABLE for LTOP if normal letdown is in service.

b. Two OPERABLE RHR suction relief valves (ND-3 and ND-38); or

An RHR suction relief valve is OPERABLE for LTOP when both of its RHR suction isolation valves are open, its setpoint is at or between 417 psig and 509 psig, and testing has proven its ability to open in this pressure range.

c. One OPERABLE PORV and one OPERABLE RHR suction relief valve.

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is ≤ 210°F, in MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above 210°F. When the reactor vessel head is off, overpressurization cannot occur.

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 above 210°F.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows

BASES

APPLICABILITY (continued)

operator action to mitigate the event.

The Applicability is modified by a Note stating that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable LTOP system. There is an increased risk associated with entering MODE 4 from MODE 5 with LTOP inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1

With more than two pumps (charging and/or safety injection) capable of injecting into the RCS, RCS overpressurization is possible.

To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.

B.1

With RCP operation not limited in accordance with Table 3.4.12-1, RCS overpressurization is possible.

To immediately initiate action to limit pump operation reflects the urgency of removing the RCS from this condition.

C.1, D.1, and D.2

An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

If isolation is needed and cannot be accomplished in 1 hour, Required Action D.1 and Required Action D.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS

BASES

ACTIONS (continued)

temperature to > 210°F, an accumulator pressure of 678 psig cannot exceed the LTOP limits if the accumulators are fully injected. Depressurizing the accumulators below the LTOP limit also gives this protection.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed times.

E.1

In MODE 4 when any RCS cold leg temperature is ≤ 210°F, with one RCS relief valve inoperable, the RCS relief valve must be restored to OPERABLE status within a Completion Time of 7 days. Two RCS relief valves (in any combination of the PORVs and RHR suction relief valves) are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

The Completion Time considers the facts that only one of the RCS relief valves is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low.

F.1

The consequences of operational events that will overpressurize the RCS are more severe at lower temperature (Ref. 7). Thus, with one of the two RCS relief valves inoperable in MODE 5 or in MODE 6 with the head on Completion Time to restore two valves to OPERABLE status is 24 hours.

The Completion Time represents a reasonable time to investigate and repair several types of relief valve failures without exposure to a lengthy period with only one OPERABLE RCS relief valve to protect against overpressure events.

BASES

ACTIONS (continued)

G.1 and G.2

Steps must be taken immediately to limit potential mass input into the RCS, and the RCS must be depressurized and a vent must be established within 12 hours when:

a. Both required RCS relief valves are inoperable; or

b. A Required Action and associated Completion Time of Condition A, D, E, or F is not met; or

c. The LTOP System is inoperable for any reason other than Condition A, C, D, E, or F.

The Reference 3 analyses demonstrate that with the mass input into the RCS reduced to that of one injection pump (charging or safety injection) an RCS vent of ≥ 4.5 square inches can maintain RCS pressure below limits. Therefore the Condition requires action to be taken immediately to reduce the input to that of one injection pump (charging or safety injection) prior to commencing RCS pressure reduction and establishing the required RCS vent. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle fracture of the reactor vessel.

The capacity of a vent this size is greater than the flow of the limiting transient for the LTOP configuration, one charging pump or one safety injection pump OPERABLE, maintaining RCS pressure less than the maximum pressure on the P/T limit curve. The required vent capacity may be provided by one or more vent paths. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

The RCS vent size will be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure.

The Completion Time considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

BASES

ACTIONS (continued)

G.3

The RCS vent of $\geq$ 4.5 square inches is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a valve that is not locked, (valves that are sealed or secured in the open position are considered "locked" in this context); or

b. Once every 31 days for other vent path(s) (e.g., a vent valve that is locked, sealed or secured in position or a removed pressurizer safety valve or open manway also fits this category).

The passive vent valve arrangement must only be open to be OPERABLE. This Required Action is required to be performed if the vent is being used to satisfy the pressure relief requirements of Required Action G.2.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1 and SR 3.4.12.2

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, a maximum of two pumps (charging and/or safety injection) are verified capable of injecting into the RCS and the accumulator discharge isolation valves are verified closed and power removed.

The pumps are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control. An alternate method of LTOP control may be employed using at least two independent means to prevent a pump start such that a single failure or single action will not result in an injection into the RCS. This may be accomplished through two valves in the discharge flow path being closed.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.4.12.3

Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction isolation valves are open and by testing it in accordance with the INSERVICE TESTING PROGRAM. This

Surveillance is only required to be performed if the RHR suction relief valve is being used to meet this LCO.

The RHR suction isolation valves are verified to be opened. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

The ASME Code (Ref. 9), test per INSERVICE TESTING PROGRAM verifies OPERABILITY by proving relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

SR 3.4.12.4

The PORV block valve must be verified open to provide the flow path for each required PORV to perform its function when actuated. The valve must be remotely verified open in the main control room. This Surveillance is performed if the PORV satisfies the LCO.

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required removed, and the manual operator is not required locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure situation.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.4.12.5

Performance of a COT is required within 12 hours after decreasing RCS temperature to ≤ 210°F and periodically on each required PORV to verify and, as necessary, adjust its lift setpoint. The COT will verify the setpoint is within the allowed maximum limits. PORV actuation could depressurize the RCS and is not required. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

The 12 hour Frequency considers the unlikelihood of a low temperature overpressure event during this time.

A Note has been added indicating that this SR is required to be met 12 hours after decreasing RCS cold leg temperature to ≤ 210°F. The COT cannot be performed until in the LTOP MODES when the PORV lift setpoint can be reduced to the LTOP setting. The test must be performed within 12 hours after entering the LTOP MODES.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.6

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.4.12.7

Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction isolation valves are open and by testing it in accordance with the INSERVICE TESTING PROGRAM. (Refer to SR 3.4.12.3 for the RHR suction isolation valves Surveillance and for a description of the INSERVICE TESTING PROGRAM.) This Surveillance is only required to be performed if the RHR suction relief valve is being used to meet this LCO.

The RHR suction isolation valves are verified open, with power to the valve operator removed and locked in the removed position, to ensure that accidental closure will not occur. The "locked open in the removed position" power supply must be locally verified in its open position with the power supply to the valve locked in its inactive position. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. UFSAR, Section 5.2
4. 10 CFR 50, Section 50.46.
5. 10 CFR 50, Appendix K.
6. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
7. Generic Letter 90-06.
8. ASME, Boiler and Pressure Vessel Code, Section III.
9. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS—Operating

BASES

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system;

b. Rod ejection accident;

c. Loss of secondary coolant accident, including uncontrolled steam or feedwater release; and

d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

There are three phases of ECCS operation: injection, cold leg recirculation, and hot leg recirculation. In the injection phase, water is taken from the refueling water storage tank (RWST) and injected into the Reactor Coolant System (RCS) through the cold legs. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment sumps have enough water to supply the required net positive suction head to the ECCS pumps, suction is switched to the containment sump for cold leg recirculation. When the core decay heat has decreased to a level low enough to be successfully removed without direct RHR pump injection flow, the RHR cold leg injection path is realigned to discharge to the auxiliary containment spray header. After approximately 7 hours, part of the ECCS flow is shifted to the hot leg recirculation phase to provide a backflush which, for a cold leg break, would reduce the boiling in the top of the core and prevent excessive boron concentration.

The ECCS consists of three separate subsystems: centrifugal charging (high head), safety injection (SI) (intermediate head), and residual heat removal (RHR) (low head). Each subsystem consists of two redundant, 100% capacity trains. The ECCS accumulators and the RWST are also part of the ECCS, but are not considered part of an ECCS flow path as described by this LCO.

BASES

BACKGROUND (continued)

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the RWST can be injected into the RCS following the accidents described in this LCO. The major components of each subsystem are the centrifugal charging pumps, the RHR pumps, heat exchangers, and the SI pumps. Each of the three subsystems consists of two 100% capacity trains that are interconnected and redundant such that either train is capable of supplying 100% of the flow required to mitigate the accident consequences. This interconnecting and redundant subsystem design provides the operators with the ability to utilize components from opposite trains to achieve the required 100% flow to the core.

During the injection phase of LOCA recovery, a suction header supplies water from the RWST to the ECCS pumps. Mostly separate piping supplies each subsystem and each train within the subsystem. The discharge from the centrifugal charging pumps combines, then divides again into four supply lines, each of which feeds the injection line to one RCS cold leg. The discharge from the SI and RHR pumps divides and feeds an injection line to each of the RCS cold legs. Throttle valves in the SI lines are set to balance the flow to the RCS. This balance ensures sufficient flow to the core to meet the analysis assumptions following a LOCA in one of the RCS cold legs. The flow split from the RHR lines cannot be adjusted. Although much of the two ECCS trains are composed of completely separate piping, certain areas are shared between trains. The most important of these are 1) where both trains flow through a single physical pipe, and 2) at the injection connections to the RCS cold legs. Since each train must supply sufficient flow to the RCS to be considered 100% capacity, credit is taken in the safety analyses for flow to three intact cold legs. Any configuration which, when combined with a single active failure, prevents the flow from either ECCS pump in a given train from reaching all four cold legs injection points on that train is unanalyzed and might render both trains of that ECCS subsystem inoperable.

For LOCAs that are too small to depressurize the RCS below the shutoff head of the SI pumps, the centrifugal charging pumps supply water until the RCS pressure decreases below the SI pump shutoff head. During this period, the steam generators are used to provide part of the core cooling function.

During the recirculation phase of LOCA recovery, RHR pump suction is transferred to the containment sump. The RHR pumps then supply the other ECCS pumps. Initially, recirculation is through the same paths as the injection phase. Subsequently, for large LOCAs, the recirculation phase includes injection into both the hot and cold legs.

BASES

BACKGROUND (continued)

The high and intermediate head subsystems of the ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as a main steam line break (MSLB). The limiting design conditions occur when the moderator temperature coefficient is highly negative, such as at the end of each cycle.

During low temperature conditions in the RCS, limitations are placed on the maximum number of ECCS pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for the basis of these requirements.

The ECCS subsystems are actuated upon receipt of an SI signal. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start immediately in the programmed sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a safety injection actuation.

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a small break LOCA and there is a high level of probability that the criteria are met following a large break LOCA:

a. Maximum fuel element cladding temperature is $\leq$ 2200°F;

b. Maximum cladding oxidation is $\leq$ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is $\leq$ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

BASES

APPLICABLE SAFETY ANALYSES (continued)

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

The LCO also limits the potential for a post trip return to power following an MSLB event and ensures that containment pressure and temperature limits are met.

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Refs. 3 and 4). This event has the greatest potential to challenge the limits on runout flow set by the manufacturer of the ECCS pumps. It also sets the maximum response time for their actuation. Direct flow from the centrifugal charging pumps and SI pumps is credited in a small break LOCA event. The RHR pumps are also credited, for larger small break LOCAs, as the means of supplying suction to these higher head ECCS pumps after the switch to sump recirculation. This event establishes the flow and discharge head at the design point for the centrifugal charging pumps. The MSLB analysis also credits the SI and centrifugal charging pumps. Although some ECCS flow is necessary to mitigate a SGTR event, a single failure disabling one ECCS train is not the limiting single failure for this transient. The SGTR analysis primary to secondary break flow is increased by the availability of both centrifugal charging and SI trains. Therefore, the SGTR analysis is penalized by assuming both ECCS trains are operable as required by the LCO. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

a. A large break LOCA event, with loss of offsite power and a single failure disabling one ECCS train; and

b. A small break LOCA event, with a loss of offsite power and a single failure disabling one ECCS train.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

The effects on containment mass and energy releases are accounted for in appropriate analyses (Ref. 3). The LCO ensures that an ECCS train will deliver sufficient water to match boiloff rates soon enough to minimize the consequences of the core being uncovered following a large LOCA.

BASES

APPLICABLE SAFETY ANALYSES (continued)

It also ensures that the centrifugal charging and SI pumps will deliver sufficient water and boron during a small LOCA to maintain core subcriticality. For smaller LOCAs, the centrifugal charging pump delivers sufficient fluid to maintain RCS inventory. For a small break LOCA, the steam generators continue to serve as the heat sink, providing part of the required core cooling.

The ECCS trains satisfy Criterion 3 of 10 CFR 50.36 (Ref. 5).

LCO

In MODES 1, 2, and 3, two independent (and redundant) ECCS trains are required to ensure that sufficient ECCS flow is available, assuming a single failure affecting either train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

In MODES 1, 2, and 3, an ECCS train consists of a centrifugal charging subsystem, an SI subsystem, and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an SI signal and automatically transferring suction to the containment sump.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the four cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to supply its flow to the RCS hot and cold legs. The flow path for each train must maintain its designed independence to ensure that no single failure can disable both ECCS trains. Management of gas voids is important to ECCS OPERABILITY.

APPLICABILITY

In MODES 1, 2, and 3, the ECCS OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The centrifugal charging pump performance is based on a small break LOCA, which establishes the pump performance curve and has less dependence on power. The SI pump performance requirements are based on a small break LOCA. For both of these types of pumps, the large break LOCA analysis depends only on the flow value at containment pressure, not on the shape of the flow versus pressure curve at higher pressures. MODE 2

BASES

APPLICABILITY (continued)

and MODE 3 requirements are bounded by the MODE 1 analysis.

This LCO is only applicable in MODE 3 and above. Below MODE 3, the SI signal setpoint is manually bypassed by operator control, and system functional requirements are relaxed as described in LCO 3.5.3, "ECCS—Shutdown."

As indicated in the Note, the flow path may be isolated for 2 hours in MODE 3, under controlled conditions, to perform pressure isolation valve testing per SR 3.4.14.1. The flow path is readily restorable from the control room.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level," and LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level."

ACTIONS

A.1

With one or more trains inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 6) and is a reasonable time for repair of many ECCS components.

An ECCS train is inoperable if it is not capable of delivering design flow to the RCS. Individual components are inoperable if they are not capable of performing their design function or supporting systems are not available.

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. The intent of this Condition is to maintain a combination of equipment such that 100% of the ECCS flow equivalent to a single OPERABLE ECCS train remains available. This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable.

BASES

ACTIONS (continued)

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 6) has shown that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

Reference 7 describes situations in which one component, such as an RHR crossover valve, can disable both ECCS trains. With one or more component(s) inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered.

B.1 and B.2

If the inoperable trains cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Misalignment of these valves could render both ECCS trains inoperable. Securing these valves using the power disconnect switches in the correct position ensures that they cannot change position as a result of an active failure or be inadvertently misaligned. These valves are of the type, described in Reference 7, that can disable the function of both ECCS trains and invalidate the accident analyses. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing,

SURVEILLANCE REQUIREMENTS (continued)

or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

The Surveillance is modified by a Note which exempts system vent flow paths opened under administrative control. The administrative control should be proceduralized and include stationing a dedicated individual at the system vent flow path who is in continuous communication with the operators in the control room. This individual will have a method to rapidly close the system vent flow path if directed.

SR 3.5.2.3

ECCS piping and components have the potential to develop voids and pockets of entrained gases. Preventing and managing gas intrusion and accumulation is necessary for proper operation of the ECCS and may also prevent water hammer, pump cavitation, and pumping of noncondensible gas into the reactor vessel.

Selection of ECCS locations susceptible to gas accumulation is based on a review of system design information, including piping and instrumentation drawings, isometric drawings, plan and elevation drawings, and calculations. The design review is supplemented by system walk downs to validate the system high points and to confirm the location and orientation of important components that can become sources of gas or could otherwise cause gas to be trapped or difficult to remove during system maintenance or restoration. Susceptible locations depend on plant and system configuration, such as stand-by versus operating conditions.

The ECCS is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds an acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If it is determined by subsequent evaluation that the ECCS is not rendered inoperable by the accumulated

BASES

SURVEILLANCE REQUIREMENTS (continued)

gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits.

ECCS locations susceptible to gas accumulation are monitored and, if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative sub-set of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, the plant configuration, or personnel safety. For these locations alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. The Surveillance Frequency may vary by location susceptible to gas accumulation.

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the INSERVICE TESTING PROGRAM, the ASME Code. The ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.5 and SR 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI and Containment Sump Recirculation signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.5.2.7

The position of throttle valves in the flow path on an SI signal is necessary for proper ECCS performance. These valves have mechanical locks to ensure proper positioning for restricted flow to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.5.2.8

Periodic inspections of the containment sump suction inlet ensure that it is unrestricted and stays in proper operating condition. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

Upon completion of the ECCS sump strainer assembly modifications during outage 2EOC15 for Unit 2 and 1EOC17 for Unit 1, the following SR Bases will apply:

Periodic inspections of the ECCS containment sump strainer assembly (consisting of modular tophats, grating, plenums, and waterboxes) ensure it is unrestricted and remains in proper operating condition.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Inspections will consist of a visual examination of the exterior surfaces of the strainer assembly for any evidence of debris, structural distress or abnormal corrosion. The intent of this surveillance is to ensure the absence of any condition which could adversely affect strainer functionality. Surveillance performance does not require removal of any tophat modules or grating, but the strainer exteriors shall be visually inspected. This surveillance is not a commitment to inspect 100% of the surface area of all tophats, but a sufficiently detailed inspection of exterior strainer surfaces is required to establish a high confidence that no adverse conditions are present. The scope of inspection necessary to provide high confidence includes 100% of the strainer areas that can be accessed and inspected using normal means and tools (i.e., flashlight, extendable mirror, hand held digital camera) without disassembly, and that difficult to access areas will be inspected to the extent possible using these same means.

Any damage detected in the strainer assembly inspection will result in an expansion of the scope of the inspection to include other areas of potential damage. Inspection scope should be expanded, as needed, for degradation of strainer components identified during this inspection that were not considered readily accessible during the inspector's initial evaluation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.
2. 10 CFR 50.46.
3. UFSAR, Section 6.2.1.
4. UFSAR, Chapter 15.
5. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
6. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
7. IE Information Notice No. 87-01.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves

BASES

BACKGROUND

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on a containment isolation signal. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system. These barriers (typically containment isolation valves) make up the Containment Isolation System.

Automatic isolation signals are produced during accident conditions. Containment Phase "A" isolation occurs upon receipt of a safety injection signal. The Phase "A" isolation signal isolates nonessential process lines in order to minimize leakage of fission product radioactivity. Containment Phase "B" isolation occurs upon receipt of a containment pressure High-High signal and isolates the remaining process lines, except systems required for accident mitigation. The Containment Purge Ventilation and Containment Air Release and Addition valves also receive an isolation signal on a containment high radiation condition. As a result, the containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated from the environment in the event of a release of fission product radioactivity to the containment atmosphere as a result of a Design Basis Accident (DBA).

The OPERABILITY requirements for containment isolation valves help ensure that containment is isolated within the Time limits assumed in the safety analyses. Therefore, the OPERABILITY requirements provide assurance that the containment function assumed in the safety analyses will be maintained.

BASES

BACKGROUND (continued)

Containment Purge Ventilation System

The Containment Purge Ventilation System consists of the Containment Purge Supply and Exhaust Systems and the Incore Instrumentation Room Purge Supply and Exhaust Systems. These systems are used during refueling and post LOCA conditions and are not utilized during MODES 1 - 4. The penetration valves are sealed closed in MODES 1 – 4.

The Containment Purge Supply System includes one supply duct penetration through the Reactor Building wall into the annulus area. There are four purge air supply penetrations through the containment vessel, two to the upper compartment and two to the lower compartment. Two normally closed isolation valves at each penetration through the containment vessel provide containment isolation.

The Containment Purge Exhaust System includes one purge exhaust duct penetration through the Reactor Building wall from the annulus area. There are three purge exhaust penetrations through the containment vessel, two from the upper compartment and one from the lower compartment. Two normally closed isolation valves at each penetration through the containment vessel provide containment isolation.

The Incore Instrumentation Room Purge Supply System consists of one purge supply penetration through the Reactor Building wall and one through the containment vessel. Two normally closed isolation valves at the containment penetration provide containment isolation.

The Incore Instrumentation Room Purge Exhaust System consists of one purge exhaust penetration through the Reactor Building wall and one through the containment vessel. Two normally closed isolation valves at the penetration through the containment vessel provide containment isolation.

Containment Hydrogen Purge System

The Containment Hydrogen Purge System consists of a containment hydrogen purge inlet blower, which blows air from the Auxiliary Building through a 4 inch pipe into the upper compartment of the containment. Another 4 inch pipe originating in the upper compartment of the containment purges air from the containment to the annulus. The penetration valves are sealed closed during MODES 1 - 4.

BASES

BACKGROUND (continued)

Containment Air Release and Addition System

The Containment Air Release and Addition System is only used for controlling Containment pressure during normal unit operation. Isolation valves are located both inside and outside of the Containment on each containment penetration.

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analyses of any event requiring isolation of containment is applicable to this LCO.

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA) and a rod ejection accident (Ref. 1). In the analyses for each of these accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including containment purge valves) are minimized. The safety analyses assume that the containment purge supply and/or exhaust isolation valves for the lower compartment and the upper compartment, instrument room, and the Hydrogen Purge System are closed at event initiation. Since the Containment Purge Ventilation System and the Hydrogen Purge System isolation valves are sealed closed in MODES 1 – 4, they are not analyzed mechanistically in the dose calculations.

The DBA analysis assumes that, within ≤ 76 seconds after the accident, isolation of the containment is complete and leakage terminated except for the design leakage rate, L_a. The containment isolation total response time of ≤ 76 seconds includes signal delay, diesel generator startup (for loss of offsite power), and containment isolation valve stroke times.

The single failure criterion required to be imposed in the conduct of plant safety analyses was considered in the original design of the containment purge valves. Two valves in series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred.

APPLICABLE SAFETY ANALYSES (continued)

The containment purge and hydrogen purge valves may be unable to close in the environment following a LOCA. Therefore, each of the containment purge and hydrogen purge valves is required to remain sealed closed during MODES 1, 2, 3, and 4. The containment air release and addition valves may be opened during normal operation. In this case, the single failure criterion remains applicable to the containment air release and addition valves due to failure in the control circuit associated with each valve. The system valve design precludes a single failure from compromising the containment boundary as long as the system is operated in accordance with the subject LCO.

The containment isolation valves satisfy Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valves' safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during a DBA.

The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The containment purge supply and exhaust isolation valves for the lower compartment, upper compartment, instrument room, and the Hydrogen Purge System must be maintained sealed closed. The valves covered by this LCO are listed along with their associated stroke times in the UFSAR (Ref. 3).

The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact. These passive isolation valves/devices are those listed in Reference 3.

Valves with resilient seals and reactor building bypass valves must meet additional leakage rate requirements. The other containment isolation valve leakage rates are addressed by LCO 3.6.1, "Containment," as Type C testing.

This LCO provides assurance that the containment isolation valves and purge valves will perform their designed safety functions to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow paths, except for containment purge supply and exhaust isolation valves for the lower and upper compartment, instrument room, and hydrogen purge penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. For valve controls located in the control room, an operator may monitor containment isolation signal status rather than be stationed at the valve controls. Due to the size of the containment purge line penetration and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, the penetration flow path containing these valves may not be opened under administrative controls. A single purge valve in a penetration flow path may be opened to effect repairs to an inoperable valve, as allowed by SR 3.6.3.1.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are further modified by a third Note, which ensures appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve.

In the event the containment isolation valve leakage results in exceeding the overall containment leakage rate, Note 4 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1.

BASES

ACTIONS (continued)

A.1 and A.2

In the event one containment isolation valve in one or more penetration flow paths is inoperable except for purge valve or reactor building bypass leakage not within limit, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, a blind flange, and a check valve inside containment with flow through the valve secured. For a penetration flow path isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within 4 hours. The 4 hour Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification, through a system walkdown or computer status indication, that those isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

Condition A has been modified by a Note indicating that this Condition is only applicable to those penetration flow paths with two containment

BASES

ACTIONS (continued)

isolation valves. For penetration flow paths with only one containment isolation valve and a closed system, Condition C provides the appropriate actions.

Required Action A.2 is modified by two Notes. Note 1 applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these devices once they have been verified to be in the proper position, is small.

<u>B.1</u>

With two containment isolation valves in one or more penetration flow paths inoperable, the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative control and the probability of their misalignment is low.

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two containment isolation valves. Condition A of this LCO addresses the condition of one containment isolation valve inoperable in this type of penetration flow path.

ACTIONS (continued)

C.1 and C.2

With one or more penetration flow paths with one containment isolation valve inoperable, the inoperable valve flow path must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration flow path. Required Action C.1 must be completed within the 72 hour Completion Time. The specified time period is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of maintaining containment integrity during MODES 1, 2, 3, and 4. In the event the affected penetration flow path is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate because the valves are operated under administrative controls and the probability of their misalignment is low.

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. The closed system must meet the requirements of Reference 4. This Note is necessary since this Condition is written to specifically address those penetration flow paths in a closed system.

Required Action C.2 is modified by two Notes. Note 1 applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

BASES

ACTIONS (continued)

D.1

With the reactor building bypass leakage rate not within limit, the assumptions of the safety analyses are not met. Therefore, the leakage must be restored to within limit within 4 hours. Restoration can be accomplished by isolating the penetration(s) that caused the limit to be exceeded by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. When a penetration is isolated the leakage rate for the isolated penetration is assumed to be the actual pathway leakage through the isolation device. If two isolation devices are used to isolate the penetration, the leakage rate is assumed to be the lesser actual pathway leakage of the two devices. The 4 hour Completion Time is reasonable considering the time required to restore the leakage by isolating the penetration(s) and the relative importance of secondary containment bypass leakage to the overall containment function.

E.1, E.2, and E.3

In the event one or more containment purge, hydrogen purge, or containment air release and addition valves in one or more penetration flow paths are not within the leakage limits, leakage must be restored to within limits, or the affected penetration flow path must be isolated. The method of isolation must be by the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, closed manual valve, or blind flange. A valve with resilient seals utilized to satisfy Required Action E.1 must have been demonstrated to meet the leakage requirements of SR 3.6.3.6. The specified Completion Time is reasonable, considering that one containment purge valve remains closed so that a gross breach of containment does not exist.

In accordance with Required Action E.2, this penetration flow path must be verified to be isolated on a periodic basis. The periodic verification is necessary to ensure that containment penetrations required to be isolated following an accident, which are no longer capable of being automatically isolated, will be in the isolation position should an event occur. This Required Action does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown or computer status indication, that those isolation devices outside containment capable of being mispositioned are in the correct position.

ACTIONS (continued)

For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

For the valve with resilient seal that is isolated in accordance with Required Action E.1, SR 3.6.3.6 must be performed at least once every 92 days. This assures that degradation of the resilient seal is detected and confirms that the leakage rate of the containment purge valve does not increase during the time the penetration is isolated.

Required Action E.2 is modified by two Notes. Note 1 applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned.

F.1 and F.2

If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

Each containment purge supply and exhaust isolation valve for the lower compartment and the upper compartment, instrument room, and the Hydrogen Purge System is required to be verified sealed closed. This Surveillance is designed to ensure that a gross breach of containment is not caused by an inadvertent or spurious opening of a containment purge

SURVEILLANCE REQUIREMENTS (continued)

valve. Detailed analysis of these valves to conclusively demonstrate their ability to close during a LOCA in time to limit offsite doses has not been performed. Therefore, these valves are required to be in the sealed closed position during MODES 1, 2, 3, and 4. A valve that is sealed closed must have motive power to the valve operator removed. This can be accomplished by de-energizing the source of electric power or by removing the air supply to the valve operator. In this application, the term "sealed" has no connotation of leak tightness.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. In the event valve leakage requires entry into Condition E, the Surveillance permits opening one valve in a penetration flow path to perform repairs.

SR 3.6.3.2

This SR ensures that the Containment Air Release and Addition System isolation valves are closed as required or, if open, open for an allowable reason. If a valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits. The SR is not required to be met when the valves are open for the reasons stated. The valves may be opened for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.6.3.3

This SR requires verification that each containment isolation manual valve and blind flange located outside containment or annulus and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through system walkdown or computer status indication, that those containment isolation

SURVEILLANCE REQUIREMENTS (continued)

valves outside containment and capable of being mispositioned are in the correct position. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be the correct position upon locking, sealing, or securing.

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3 and 4 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once

SR 3.6.3.4

This SR requires verification that each containment isolation manual valve and blind flange located inside containment or annulus and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. For containment isolation valves inside containment, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate since these containment isolation valves are operated under administrative controls and the probability of their misalignment is low. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time they are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be the correct position upon locking, sealing, or securing.

This Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4, for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in their proper position, is small.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.5

Verifying that the isolation time of each automatic power operated containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time is specified in the UFSAR and the Frequency of this SR is in accordance with the INSERVICE TESTING PROGRAM.

SR 3.6.3.6

For the Containment Purge System valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B is required to ensure OPERABILITY. The measured leakage rate for the Containment Purge System and Hydrogen Purge System valves must be $\leq$ 0.05 L_a when pressurized to P_a. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), these valves will not be placed on the maximum extended test interval. Therefore, these valves will be tested in accordance with NEI 94-01 with a maximum test interval of 30 months.

The Containment Air Release and Addition System valves have a demonstrated history of acceptable leakage. The measured leakage rate for containment air release and addition valves must be $\leq$ 0.01 L_a when pressurized to P_a. These valves will be tested in accordance with NEI 94-01 with a maximum test interval of 30 months.

SR 3.6.3.7

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment

SURVEILLANCE REQUIREMENTS (continued)

isolation signal. The isolation signals involved are Phase A, Phase B, and Safety Injection. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.6.3.8

This SR ensures that the combined leakage rate of all reactor building bypass leakage paths is less than or equal to the specified leakage rate. This provides assurance that the assumptions in the safety analysis are met. The Frequency is required by the Containment Leakage Rate Testing Program. This SR simply imposes additional acceptance criteria.

Bypass leakage is considered part of L_a.

REFERENCES

1. UFSAR, Section 15.
2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
3. UFSAR, Section 6.2.
4. Standard Review Plan 6.2.4.
5. Not used.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Containment Spray System

BASES

BACKGROUND

The Containment Spray System provides containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduce the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA). The Containment Spray System is designed to meet the requirements of 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal Systems," GDC 40, "Testing of Containment Heat Removal Systems," GDC 41, "Containment Atmosphere Cleanup," GDC 42, "Inspection of Containment Atmosphere Cleanup Systems," and GDC 43, "Testing of Containment Atmosphere Cleanup Systems" (Ref. 1).

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the system design basis spray coverage. Each train includes a containment spray pump, one containment spray heat exchanger, spray headers, nozzles, valves, and piping. Each train is powered from a separate Engineered Safety Feature (ESF) bus. The containment spray pumps are started manually with the pump suction aligned to the containment sump once the ECCS is aligned to the recirculation mode of operation.

The diversion of a portion of the recirculation flow from each train of the Residual Heat Removal (RHR) System to additional redundant spray headers completes the Containment Spray System heat removal capability. Each RHR train is capable of supplying spray coverage, if desired, to supplement the Containment Spray System.

The Containment Spray System provides a spray of cold or subcooled borated water into the upper containment volume to limit the containment pressure and temperature during a DBA. In the recirculation mode of operation, heat is removed from the containment sump water by the Containment Spray System and RHR heat exchangers. Each train of the Containment Spray System, provides adequate spray coverage to meet the system design requirements for containment heat removal.

BASES

BACKGROUND (continued)

For the hypothetical double-ended rupture of a Reactor Coolant System pipe, the pH of the sump solution (and, consequently, the spray solution) is raised to approximately 7.9 within one hour of the onset of the LOCA. The resultant pH of the sump solution is based on the mixing of the RCS fluids, ECCS injection fluid, and the melted ice which are combined in the sump. The alkaline pH of the containment sump water minimizes the evolution of iodine and the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid.

The Containment Spray System is actuated manually. The Containment Spray System is not actuated until an RWST level Low-Low alarm is received and the ECCS has been realigned in the recirculation mode of operation. The Low-Low alarm for the RWST signals the operator to manually align the ECCS to the recirculation mode and to manually initiate the Containment Spray System. The Containment Spray System maintains an equilibrium temperature between the containment atmosphere and the recirculated sump water. Operation of the Containment Spray System in the recirculation mode is controlled by the operator in accordance with the emergency operation procedures.

The RHR spray operation is initiated manually, when required by the emergency operating procedures, after the ECCS is operating in the recirculation mode. The RHR sprays are available to supplement the Containment Spray System, if desired, in limiting containment pressure. This additional spray capacity would typically be used after the ice bed has been depleted and in the event that containment pressure rises above a predetermined limit. However, RHR spray operation is not credited for design basis events. The Containment Spray System is designed to ensure that the heat removal capability required during the post accident period can be attained.

The operation of the Containment Spray System, together with the ice condenser, is adequate to assure pressure suppression subsequent to the initial blowdown of steam and water from a DBA. During the post blowdown period, the Air Return System (ARS) is automatically started. The ARS returns upper compartment air through the divider barrier to the lower compartment. This serves to equalize pressures in containment and to continue circulating heated air and steam through the ice condenser, where heat is removed by the remaining ice.

The Containment Spray System limits the temperature and pressure that could be expected following a DBA. Protection of containment integrity limits leakage of fission product radioactivity from containment to the environment.

BASES

APPLICABLE SAFETY ANALYSES

The limiting DBAs considered relative to containment OPERABILITY are the loss of coolant accident (LOCA) and the steam line break (SLB). The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed, in regard to containment ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train of the Containment Spray System, the RHR System, and the ARS being rendered inoperable (Ref. 2).

The DBA analyses show that the maximum peak containment pressure results from the LOCA analysis, and is calculated to be less than the containment design pressure. The maximum peak containment atmosphere temperature results from the SLB analysis and was calculated to be within the containment environmental qualification temperature during the DBA SLB. The basis of the containment environmental qualification temperature is to ensure the OPERABILITY of safety related equipment inside containment (Ref. 3).

The Containment Spray System actuation modeled in the containment analysis is based on the time associated with reaching the RWST low level setpoint prior to achieving full flow through the containment spray nozzles. A delayed response time initiation provides conservative analyses of peak calculated containment temperature and pressure responses. The Containment Spray System total response time is composed of operator action delay and system startup time.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the ECCS cooling effectiveness during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 4).

Inadvertent actuation is precluded by a design feature consisting of an additional set of containment pressure sensors which prevents operation when the containment pressure is below the containment pressure control system permissive.

The Containment Spray System satisfies Criterion 3 of 10 CFR 50.36 (Ref. 5).

BASES

LCO

During a DBA, one train of Containment Spray System is required to provide the heat removal capability assumed in the safety analyses. To ensure that this requirement is met, two containment spray trains must be OPERABLE with power from two safety related, independent power supplies. Therefore, in the event of an accident, at least one train operates.

Each Containment Spray System includes a spray pump, headers, valves, heat exchangers, nozzles, piping, instruments, and controls to ensure an OPERABLE flow path capable of being manually initiated to take suction from the containment sump and delivering it to the containment spray rings. Management of gas voids is important to Containment Spray System OPERABILITY.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the Containment Spray System.

In MODES 5 and 6, the probability and consequences of these events are reduced because of the pressure and temperature limitations of these MODES. Thus, the Containment Spray System is not required to be OPERABLE in MODE 5 or 6.

ACTIONS

A.1

With one containment spray train inoperable, the affected train must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the heat removal after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal and iodine removal capabilities afforded by the OPERABLE train and the low probability of a DBA occurring during this period.

BASES

ACTIONS (continued)

B.1 and B.2

If the affected containment spray train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time and is reasonable when considering that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment of manual and power operated valves, excluding check valves, in the Containment Spray System provides assurance that the proper flow path exists for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since they were verified in the correct position prior to being secured. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown or computer status indication, that those valves outside containment and capable of potentially being mispositioned, are in the correct position. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

The Surveillance is modified by a Note which exempts system vent flow paths opened under administrative control. The administrative control should be proceduralized and include stationing a dedicated individual at the system vent flow path who is in continuous communication with the operators in the control room. This individual will have a method to rapidly close the system vent flow path if directed.

SR 3.6.6.2

Verifying that each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump

SURVEILLANCE REQUIREMENTS (continued)

performance required by the ASME Code (Ref. 6). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on bypass flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the INSERVICE TESTING PROGRAM.

SR 3.6.6.3 and SR 3.6.6.4

Not used.

SR 3.6.6.5 and SR 3.6.6.6

These SRs require verification of proper interaction between the CPCS system and the Containment Spray System.

SR 3.6.6.5 deals solely with the containment spray pumps. It must be shown through testing that: (1) the containment spray pumps are prevented from starting in the absence of a CPCS permissive, (2) the containment spray pumps can be manually started when given a CPCS permissive, and (3) when running, the containment spray pumps stop when the CPCS permissive is removed. The "inhibit", "permit", and "terminate" parts of the CPCS interface with the containment spray pumps are verified by testing in this fashion.

SR 3.6.6.6 deals solely with containment spray header containment isolation valves NS12B, NS15B, NS29A, and NS32A. It must be shown through testing that: (1) each valve closes when the CPCS permissive is removed, OR (2) each valve is prevented from opening in the absence of a CPCS permissive. In addition to one of the above, it must also be shown that each valve can be manually opened when given a CPCS permissive.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.7

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. The spray nozzles can also be tested using a vacuum blower to induce air flow through each nozzle to verify unobstructed flow. This SR requires verification that each spray nozzle is unobstructed following activities that could cause nozzle blockage. Normal plant operation and activities are not expected to initiate this SR. However, activities such as inadvertent spray actuation that causes fluid flow through the nozzles, major configuration change, or a loss of foreign material control when working within the respective system boundary may require Surveillance performance.

SR 3.6.6.8

Containment Spray System piping and components have the potential to develop voids and pockets of entrained gases. Preventing and managing gas intrusion and accumulation is necessary for proper operation of the containment spray trains and may also prevent water hammer and pump cavitation.

Selection of Containment Spray System locations susceptible to gas accumulation is based on a review of system design information, including piping and instrumentation drawings, isometric drawings, plan and elevation drawings, and calculations. The design review is supplemented by system walk downs to validate the system high points and to confirm the location and orientation of important components that can become sources of gas or could otherwise cause gas to be trapped or difficult to remove during system maintenance or restoration. Susceptible locations depend on plant and system configuration, such as stand-by versus operating conditions.

The Containment Spray System is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds an acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If it is determined by subsequent evaluation that the Containment Spray System is not rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Accumulated gas should be eliminated or brought within the acceptance criteria limits.

Containment Spray System locations susceptible to gas accumulation are monitored and, if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative sub-set of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, the plant configuration, or personnel safety. For these locations alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. The Surveillance Frequency may vary by location susceptible to gas accumulation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, and GDC 43.
2. UFSAR, Section 6.2.
3. 10 CFR 50.49.
4. 10 CFR 50, Appendix K.
5. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
6. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Five MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the UFSAR, Section 10.3.1 (Ref. 1). The MSSV capacity criteria is 110% of rated steam flow at 110% of the steam generator design pressure. This meets the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-2 in the accompanying LCO, so that only the needed valves will actuate. Each valve is orificed to a size of 14.18 square inches. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine reactor trip.

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2 and its purpose is to limit the secondary system pressure to ≤ 110% of design pressure when passing 100% of design steam flow. This design basis is sufficient to cope with any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in the UFSAR, Section 15.2 (Ref. 3). Of these, the full power turbine trip without steam dump is the limiting AOO.

The transient response for turbine trip without a direct reactor trip presents no hazard to the integrity of the RCS or the Main Steam System.

BASES

APPLICABLE SAFETY ANALYSES (continued)

For the peak secondary pressure case, the reactor is tripped on overtemperature ΔT. Pressurizer relief valves and MSSVs are activated and prevent overpressurization in the primary and secondary systems.

The MSSVs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 4).

LCO

The accident analysis assumes five MSSVs per steam generator to provide overpressure protection for design basis transients occurring at 3479 MWt. An MSSV will be considered inoperable if it fails to open on demand. The LCO requires that five MSSVs be OPERABLE in compliance with Reference 2, even though this is not a requirement of the DBA analysis. This is because operation with less than the full number of MSSVs requires limitations on allowable THERMAL POWER (to meet ASME Code requirements). These limitations are according to Table 3.7.1-1 in the accompanying LCO, and Required Action A.1 and A.2.

The OPERABILITY of the MSSVs is defined as the ability to open within the setpoint tolerances, relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the INSERVICE TESTING PROGRAM.

The lift settings, according to Table 3.7.1-2 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB.

APPLICABILITY

In MODE 1, the number of MSSVs per steam generator required to be OPERABLE must be according to Table 3.7.1-1 in the accompanying LCO. In MODES 2 and 3, only two MSSVs per steam generator are required to be OPERABLE.

In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

BASES

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1 and A.2

With one or more MSSVs inoperable, reduce power so that the available MSSV relieving capacity meets Reference 2 requirements for the applicable THERMAL POWER.

Operation with less than all five MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is proportionally limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator. For example, if one MSSV is inoperable in one steam generator, the relief capacity of that steam generator is reduced by approximately 20%. To offset this reduction in relief capacity, energy transfer to that steam generator must be similarly reduced. This is accomplished by reducing THERMAL POWER by the necessary amount to conservatively limit the energy transfer to all steam generators, consistent with the relief capacity of the most limiting steam generator.

The maximum power level specified for the power range neutron flux high trip setpoint with inoperable MSSVs must ensure that power is limited to less than the heat removal capacity of the remaining OPERABLE MSSVs. The reduced high flux trip setpoint also ensures that the reactor trip occurs early enough in the loss of load/turbine trip event to limit primary to secondary heat transfer and preclude overpressurization of the primary and secondary systems. To calculate this power level, the governing equation is the relationship $q = m \Delta h$, where q is the heat input from the primary side, m is the steam flow rate and Δh is the heat of vaporization at the steam relief pressure (assuming no subcooled feedwater). The algorithm use is consistent with the recommendations of the Westinghouse Nuclear Safety Advisory Letter, NSAL-94-001, dated January 20, 1994 (Ref. 5). Additionally, the calculated values are reduced by 4.2% to account for instrument and channel uncertainties.

The allowed Completion Time of 4 hours provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time and provides sufficient time to reduce the trip setpoints. The adjustment of the trip setpoints is a sensitive operation that may inadvertently trip the Reactor Protection System.

BASES

ACTIONS (continued)

B.1 and B.2

If the MSSVs cannot be restored to OPERABLE status within the associated Completion Time, or if one or more steam generators have less than two MSSVs OPERABLE, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the INSERVICE TESTING PROGRAM. The ASME Code (Ref. 6), requires that safety and relief valve tests be performed. According to Reference 6, the following tests are required:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

d. Compliance with seat tightness criteria; and

e. Verification of the balancing device integrity on balanced valves.

All valves are required to be tested every 5 years, and a minimum of 20% of the valves are required to be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a $\pm$ 3% setpoint tolerance for OPERABILITY; however, the valves are reset to $\pm$ 1% during the Surveillance to allow for drift.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

BASES

REFERENCES

1. UFSAR, Section 10.3.1.

2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.

3. UFSAR, Section 15.2.

4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

5. Westinghouse Nuclear Safety Advisory Letter, NSAL-94-001, dated January 20, 1994.

6. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a high energy line break (HELB). MSIV closure terminates flow from the unaffected (intact) steam generators.

One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are downstream from the main steam safety valves (MSSVs) and auxiliary feedwater (AFW) pump turbine steam supply, to prevent MSSV and AFW isolation from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the others, and isolates the turbine, Steam Dump System, and other auxiliary steam supplies from the steam generators.

The MSIVs close on a main steam isolation signal generated by either low steam line pressure or high-high containment pressure. The MSIVs fail closed on loss of control or actuation power.

Each MSIV has an MSIV bypass valve. Although these bypass valves are normally closed, they receive the same emergency closure signal as do their associated MSIVs. The MSIVs may also be actuated manually.

A description of the MSIVs is found in the UFSAR, Section 10.3 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment and core response analyses for the large steam line break (SLB) events, discussed in the UFSAR, Sections 6.2 and 15.1.5 (Refs. 2 and 3 respectively). The design precludes the blowdown of more than one steam generator.

The limiting case for the containment analysis is the SLB from full power with continued offsite power. At full power, the steam generator inlet temperatures are at their maximum, maximizing the analyzed mass and energy release to the containment. With the most reactive rod cluster control assembly assumed stuck in the fully withdrawn position, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the boric acid injection delivered by the Emergency Core Cooling System.

APPLICABLE SAFETY ANALYSES (continued)

The accident analysis compares several different SLB events against different acceptance criteria. The large SLB outside containment upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The large SLB inside containment at hot zero power is the limiting case for a post trip return to power. The analysis includes scenarios with offsite power available, and with a loss of offsite power following turbine trip. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System cooldown. With a loss of offsite power, the response of mitigating systems is delayed. Significant single failures considered include failure of an MSIV to close.

The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:

a. An HELB inside containment. In order to maximize the containment temperature response, the analysis assumes quick closure of all MSIVs. For this accident scenario, steam is discharged into containment from all steam generators until the MSIVs close. After MSIV closure, steam is discharged into containment only from the affected steam generator. Closure of the MSIVs isolates the break from the unaffected steam generators.

b. A break outside of containment and upstream from the MSIVs is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs isolates the break and limits the blowdown to a single steam generator.

c. A break downstream of the MSIVs will be isolated by the closure of the MSIVs.

d. Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator from the intact steam generators to minimize radiological releases.

e. The MSIVs are also utilized during other events such as a feedwater line break. This event is less limiting so far as MSIV OPERABILITY is concerned.

The MSIVs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 4).

BASES

LCO

This LCO requires that four MSIVs in the steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation actuation signal.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 50.67 (Ref. 5) limits or the NRC staff approved licensing basis.

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when closed and de-activated, when there is significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing the safety function.

In MODE 4, the steam generator energy is low.

In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A.1

With one MSIV inoperable in MODE 1, action must be taken to restore OPERABLE status within 8 hours. Some repairs to the MSIV can be made with the unit hot. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the MSIVs.

The 8 hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from other containment isolation valves in that the closed system provides an additional means for containment isolation.

B.1

If the MSIV cannot be restored to OPERABLE status within 8 hours, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 2 within 6 hours and Condition C would be entered. The Completion Times are

BASES

ACTIONS (continued)

reasonable, based on operating experience, to reach MODE 2 and to close the MSIVs in an orderly manner and without challenging unit systems.

C.1 and C.2

Condition C is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODES 2 and 3, the inoperable MSIVs may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

The 8 hour Completion Time is consistent with that allowed in Condition A.

For inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, the inoperable MSIVs must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of MSIV status indications available in the control room, and other administrative controls, to ensure that these valves are in the closed position.

D.1 and D.2

If the MSIVs cannot be restored to OPERABLE status or are not closed within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that the closure time of each MSIV is ≤ 8.0 seconds on an actual or simulated actuation signal. The MSIV closure time is assumed in the accident and containment analyses. This SR also

BASES

SURVEILLANCE REQUIREMENTS (continued)

verifies the valve closure time is in accordance with the INSERVICE TESTING PROGRAM. This SR is normally performed upon returning the unit to operation following a refueling outage. The MSIVs should not be tested at power, since even a part stroke exercise increases the risk of a valve closure when the unit is generating power. As the MSIVs are not tested at power, they are exempt from the ASME Code (Ref. 6), requirements during operation in MODE 1 or 2. The Frequency is in accordance with the INSERVICE TESTING PROGRAM.

This test is conducted in MODE 3 with the unit at operating temperature and pressure, as discussed in Reference 6 exercising requirements. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

REFERENCES

1. UFSAR, Section 10.3.
2. UFSAR, Section 6.2.
3. UFSAR, Section 15.1.5.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
5. 10 CFR 50.67.
6. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Isolation Valves (MFIVs), Main Feedwater Control Valves (MFCVs), their Associated Bypass Valves, and the Tempering Valves

BASES

BACKGROUND

The MFIVs isolate main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break (HELB). The safety related function of the MFCVs is to provide the second isolation of MFW flow to the secondary side of the steam generators following an HELB. Closure of the MFIVs and associated bypass valves or MFCVs and associated bypass valves terminates flow to the steam generators, terminating the event for feedwater line breaks (FWLBs) occurring upstream of the MFIVs or MFCVs. The consequences of events occurring in the main steam lines or in the MFW lines downstream from the MFIVs will be mitigated by their closure. Closure of the MFIVs and associated bypass valves, or MFCVs and associated bypass valves, effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FWLBs inside containment, and reducing the cooldown effects for SLBs.

The MFIVs and associated bypass valves, which includes the tempering and the MFW to auxiliary feedwater (AFW) bypass nozzle isolation valves, isolate the nonsafety related portions from the safety related portions of the system. In the event of a secondary side pipe rupture inside containment, the valves limit the quantity of high energy fluid that enters containment through the break, and provide a pressure boundary for the controlled addition of AFW to the intact loops.

One MFIV and associated bypass valve, and one MFCV and its associated bypass valve, are located on each MFW line, outside but close to containment. The MFIVs and MFCVs are located on different supply lines from the AFW injection line so that AFW may be supplied to the steam generators following MFIV or MFCV closure. The piping volume from these valves to the steam generators must be accounted for in calculating mass and energy releases, and refilled prior to AFW reaching the steam generator following either an SLB or FWLB.

The MFIVs and associated bypass valves, and MFCVs and associated bypass valves, close on receipt of a safety injection signal, T_{avg}—Low coincident with reactor trip (P-4), or steam generator water level—high high signal. They may also be actuated manually. The check valve

BACKGROUND (continued)

outside containment prevents multiple steam generator blowdown and overcooling in the event of a nonsafety related pipe failure or faulted steam generator concurrent with a single failure of a MFIV on an otherwise intact steam generator.

A description of the MFIVs and MFCVs is found in the UFSAR, Section 10.4.7 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the MFIVs and MFCVs is established by the analyses for the large SLB. It is also influenced by the accident analysis for the large FWLB. Closure of the MFIVs and associated bypass valves, or MFCVs and associated bypass valves, may also be relied on to terminate an SLB for core response analysis and excess feedwater event upon the receipt of a steam generator water level—high high signal.

Failure of an MFIV, MFCV, or the associated bypass valves to close following an SLB or FWLB can result in additional mass being delivered to the steam generators, contributing to cooldown. This failure also results in additional mass and energy releases following an SLB or FWLB event.

The MFIVs and MFCVs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

This LCO ensures that the MFIVs, MFCVs, and their associated bypass valves and tempering valves will isolate MFW flow to the steam generators, following an FWLB or main steam line break.

This LCO requires that four MFIVs and associated bypass valves and four MFCVs and associated bypass valves and tempering valves be OPERABLE. The MFIVs and MFCVs and the associated bypass valves and tempering valves considered OPERABLE when isolation times are within limits and they close on an isolation actuation signal.

Failure to meet the LCO requirements can result in additional mass being released to containment following an SLB or FWLB inside containment. If a feedwater isolation signal on high steam generator level is relied on to terminate an excess feedwater flow event, failure to meet the LCO may result in the introduction of water into the main steam lines.

BASES

APPLICABILITY

The MFIVs and MFCVs and the associated bypass valves and tempering valves must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and steam generators. This ensures that, in the event of an HELB, a single failure cannot result in the blowdown of more than one steam generator. In MODES 1, 2, and 3, the MFIVs and MFCVs and the associated bypass valves and tempering valves are required to be OPERABLE to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed and de-activated or isolated by a closed manual valve, they are already performing their safety function.

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the MFIVs, MFCVs, and the associated bypass valves are normally closed since MFW is not required.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1 and A.2

With one MFIV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours by use of a closed and de-activated automatic valve, a closed manual valve, or blind flange. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

Inoperable MFIVs that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

BASES

ACTIONS (continued)

B.1 and B.2

With one MFCV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours by use of a closed and de-activated automatic valve, a closed manual valve, or blind flange. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

Inoperable MFCVs, that are closed or isolated, must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls to ensure that the valves are closed or isolated.

C.1 and C.2

With one associated bypass valve in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours by use of a closed and de-activated automatic valve, a closed manual valve, or blind flange. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

Inoperable associated bypass valves that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering

BASES

ACTIONS (continued)

judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

D.1

With the tempering valve inoperable or two inoperable valves in the same flow path, there may be no redundant system to operate automatically and perform the required safety function. The tempering valves have no other automatic isolation valves in series to provide isolation. Under these conditions, affected valves in each flow path must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. This action returns the system to the condition where at least one valve in each flow path is performing the required safety function. The 8 hour Completion Time is reasonable, based on operating experience, to complete the actions required to close the MFIV or MFCV, or otherwise isolate the affected flow path.

E.1 and E.2

If the MFIV(s), MFCV(s), and the associated bypass valve(s) or the tempering valve(s) cannot be restored to OPERABLE status, or closed, or isolated within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFIV, MFCV, and associated bypass valves, and the tempering valve is ≤ 12 seconds on an actual or simulated actuation signal. The MFIV and MFCV closure times are assumed in the accident and containment analyses. This SR also verifies the valve closure time is in accordance with the INSERVICE TESTING PROGRAM. This SR is normally performed upon returning the unit to operation following a refueling outage. These valves should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. This is consistent with the

BASES

SURVEILLANCE REQUIREMENTS (continued)

ASME Code (Ref. 3), quarterly stroke requirements during operation in MODES 1 and 2.

The Frequency for this SR is in accordance with the INSERVICE TESTING PROGRAM.

REFERENCES

1. UFSAR, Section 10.4.7.
2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
3. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply. The AFW pumps take suction through suction lines from the condensate storage system (CSS) (LCO 3.7.6) and pump to the steam generator secondary side. The normal supply of water to the AFW pumps is from the condensate system. The supply valves are open with power removed from the valve operator. The assured source of water to the AFW System is supplied by the Nuclear Service Water System. The turbine and motor driven pump discharge lines to each individual steam generator join into a single line outside containment. These individual lines penetrate the containment and enter each steam generator through the auxiliary feedwater nozzle. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1) or SG PORVs (LCO 3.7.4). If the main condenser is available, steam may be released via the steam dump valves and recirculated to the hotwell.

The AFW System consists of two motor driven AFW pumps and one steam turbine driven pump configured into three trains. Each of the motor driven pumps supply 100% of the flow requirements to two steam generators, although each pump has the capability to be realigned to feed other steam generators. The turbine driven pump provides 200% of the flow requirements and supplies water to all four steam generators. Travel stops are set on the steam generator flow control valves such that the pumps can supply the minimum flow required without exceeding the maximum flow allowed. The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system. Each motor driven AFW pump is powered from an independent Class 1E power supply. The steam turbine driven AFW pump receives steam from two main steam lines upstream of the main steam isolation valves. Each of the steam feed lines will supply 100% of the requirements of the turbine driven AFW pump.

BACKGROUND (continued)

The AFW System is capable of supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions. One turbine driven pump at full flow is sufficient to remove decay heat and cool the unit to residual heat removal (RHR) entry conditions. During unit cooldown, SG pressures and Main Steam pressures decrease simultaneously. Thus, the turbine driven AFW pump with a reduced steam supply pressure remains fully capable of providing flow to all SGs. Thus, the requirement for diversity in motive power sources for the AFW System is met.

The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the lowest setpoint of the MSSVs plus 3% accumulation. Subsequently, the AFW System supplies sufficient water to cool the unit to RHR entry conditions, with steam released through the SG PORVs or MSSVs.

The motor driven AFW pumps actuate automatically on steam generator water level low-low in 1 out of 4 steam generators by the ESFAS (LCO 3.3.2). The motor driven pumps also actuate on loss of offsite power, safety injection, and trip of all MFW pumps. The turbine driven AFW pump actuates automatically on steam generator water level low-low in 2 out of 4 steam generators and on loss of offsite power.

The AFW System is discussed in the UFSAR, Section 10.4.9 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event with loss of normal feedwater.

The design basis of the AFW System is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest steam generator safety valve set pressure plus 3%.

In addition, the AFW System must supply enough makeup water to replace steam generator secondary inventory lost as the unit cools to MODE 4 conditions. Sufficient AFW flow must also be available to account for flow losses such as pump recirculation valve leakage and line breaks.

The limiting Design Basis Accidents (DBAs) and transients for the AFW System are as follows:

BASES

APPLICABLE SAFETY ANALYSES (continued)

a. Feedwater Line Break (FWLB); and

b. Loss of MFW.

In addition, the minimum available AFW flow and system characteristics are considered in the analysis of a small break loss of coolant accident (LOCA) and events that could lead to steam generator tube bundle uncovery for dose considerations.

A range of AFW flows is considered for the analyzed accidents, with the Main Steam Line Break being the most limiting for the maximum AFW flowrate.

The AFW System design is such that it can perform its function following a FWLB between the steam generator and the downstream check valve, combined with a loss of offsite power following turbine trip, and a single active failure of the steam turbine driven AFW pump. In such a case, one motor driven AFW pump would deliver to the broken MFW header at the pump runout flow until the problem was detected, and flow terminated by the operator. Sufficient flow would be delivered to the intact steam generators by the redundant AFW pump.

The ESFAS automatically actuates the AFW turbine driven pump and associated power operated valves and controls when required to ensure an adequate feedwater supply to the steam generators during loss of offsite power.

The AFW System satisfies the requirements of Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

This LCO provides assurance that the AFW System will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary. Three independent AFW pumps in three diverse trains are required to be OPERABLE to ensure the availability of RHR capability for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering two of the pumps from independent emergency buses. The third AFW pump is powered by a different means, a steam driven turbine supplied with steam from a source that is not isolated by closure of the MSIVs.

BASES

LCO (continued)

The AFW System is configured into three trains. The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the steam generators are OPERABLE. This requires that the two motor driven AFW pumps be OPERABLE in two diverse paths, each supplying AFW to separate steam generators. The turbine driven AFW pump is required to be OPERABLE with redundant steam supplies from two main steam lines upstream of the MSIVs, and shall be capable of supplying AFW to any of the steam generators. The piping, valves, instrumentation, and controls in the required flow paths also are required to be OPERABLE. The NSWS assured source of water supply is configured into two trains. The turbine driven AFW pump receives NSWS from both trains of NSWS, therefore, the loss of one train of assured source renders only one AFW train inoperable. The remaining NSWS train provides an OPERABLE assured source to the other motor driven pump and the turbine driven pump.

The LCO is modified by a Note indicating that one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4. This is because of the reduced heat removal requirements and short period of time in MODE 4 during which the AFW is required and the insufficient steam available in MODE 4 to power the turbine driven AFW pump.

APPLICABILITY

In MODES 1, 2, and 3, the AFW System is required to be OPERABLE in the event that it is called upon to function when the MFW is lost. In addition, the AFW System is required to supply enough makeup water to replace the steam generator secondary inventory, lost as the unit cools to MODE 4 conditions.

In MODE 4 the AFW System may be used for heat removal via the steam generators.

In MODE 5 or 6, the steam generators are not normally used for heat removal, and the AFW System is not required.

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable AFW train when entering MODE 1. There is an increased risk associated with entering MODE 1 with an AFW train inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not

BASES

ACTIONS (continued)

be applied in this circumstance.

A.1

If one of the two steam supplies to the turbine driven AFW train is inoperable, or if a turbine driven pump is inoperable while in MODE 3 immediately following refueling, action must be taken to restore the inoperable equipment to an OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

a. For the inoperability of a steam supply to the turbine driven AFW pump, the 7 day Completion Time is reasonable since there is a redundant steam supply line for the turbine driven pump.

b. For the inoperability of a turbine driven AFW pump while in MODE 3 immediately subsequent to a refueling, the 7 day Completion Time is reasonable due to the minimal decay heat levels in this situation.

c. For both the inoperability of a steam supply line to the turbine driven pump and an inoperable turbine driven AFW pump while in MODE 3 immediately following a refueling, the 7 day Completion Time is reasonable due to the availability of redundant OPERABLE motor driven AFW pumps; and due to the low probability of an event requiring the use of the turbine driven AFW pump.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 7 days and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

Condition A is modified by a Note which limits the applicability of the Condition to when the unit has not entered MODE 2 following a refueling. Condition A allows the turbine driven AFW pump to be inoperable for 7 days vice the 72 hour Completion Time in Condition B. This longer Completion Time is based on the reduced decay heat following refueling and prior to the reactor being critical.

BASES

ACTIONS (continued)

B.1

With one of the required AFW trains (pump or flow path) inoperable in MODE 1, 2, or 3 for reasons other than Condition A, action must be taken to restore OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines to the turbine driven AFW pump. The 72 hour Completion Time is reasonable, based on redundant capabilities afforded by the AFW System, time needed for repairs, and the low probability of a DBA occurring during this time period.

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 72 hours and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

C.1 and C.2

When Required Action A.1 or B.1 cannot be completed within the required Completion Time, or if two AFW trains are inoperable in MODE 1, 2, or 3, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

In MODE 4 with two AFW trains inoperable, operation is allowed to continue because only one motor driven pump AFW train is required in accordance with the Note that modifies the LCO. Although not required, the unit may continue to cool down and initiate RHR.

D.1

If all three AFW trains are inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting

ACTIONS (continued)

a cooldown, and only limited means for conducting a cooldown with nonsafety related equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one AFW train to OPERABLE status.

Required Action D.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one AFW train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the unit into a less safe condition.

E.1

In MODE 4, either the reactor coolant pumps or the RHR loops can be used to provide forced circulation. This is addressed in LCO 3.4.6, "RCS Loops—MODE 4." With one required AFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status. The immediate Completion Time is consistent with LCO 3.4.6.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW System water and steam supply flow paths provides assurance that the proper flow paths will exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. The SR is also modified by a note that excludes automatic valves when THERMAL POWER is $\leq$ 10% RTP. Some automatic valves may be in a throttled position to support low power operation.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref. 3). Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing as discussed in the ASME Code (Ref. 3) and the INSERVICE TESTING PROGRAM satisfies this requirement. This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.

SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note that states the SR is not required in MODE 4. In MODE 4, the required AFW train may already be aligned and operating.

SR 3.7.5.4

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an ESFAS by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal in MODES 1, 2, and 3. In MODE 4, the required pump may already be operating and the autostart function is not required. The Surveillance Frequency is based on operating experience, equipment

BASES

SURVEILLANCE REQUIREMENTS (continued)

reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

This SR is modified by two Notes. Note 1 indicates that the SR can be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test. Note 2 states that the SR is not required in MODE 4. In MODE 4, the required pump may already be operating and the autostart function is not required. In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required AFW pump if it were not in operation.

SR 3.7.5.5

This SR verifies that the AFW is properly aligned by verifying the flow paths from the CSS to each steam generator prior to entering MODE 2 after more than 30 days in MODE 5 or 6. OPERABILITY of AFW flow paths must be verified before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgment and other administrative controls that ensure that flow paths remain OPERABLE. To further ensure AFW System alignment, flow path OPERABILITY is verified following extended outages to determine no misalignment of valves has occurred. This SR ensures that the flow path from the CSS to the steam generators is properly aligned.

REFERENCES

1. UFSAR, Section 10.4.9.
2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
3. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.7 PLANT SYSTEMS

B 3.7.6 Condensate Storage System (CSS)

BASES

BACKGROUND

The CSS provides a source of water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS). The CSS provides a passive flow of water, by gravity, to the Auxiliary Feedwater (AFW) System (LCO 3.7.5). The steam produced is released to the atmosphere by the main steam safety valves, the steam generator PORVs, or to the turbine condenser. The CSS is formed from the Upper Surge Tanks (two 42,500 gallon tanks per unit) and the Condenser Hotwell (normal operating level of 170,000 gallons). The safety grade and seismically designed source of water for the AFW system, which serves as the ultimate long-term safety related source is the Standby Nuclear Service Water Pond. This required source is covered in LCO 3.7.9, "Standby Nuclear Service Water Pond (SNSWP)" and satisfies all short and long term water supply requirements for the AFW system except for Station Blackout (SBO) requirements.

When the main steam isolation valves are open, the preferred means of heat removal is to discharge steam to the condenser by the nonsafety grade path of the steam dumps to the condenser valves. The condensed steam is returned to the CSS by the condensate pump. This has the advantage of conserving condensate while minimizing releases to the environment.

A description of the CSS is found in the UFSAR, Section 10.4 (Ref. 1).

Note: The Auxiliary Feedwater Condensate Storage Tank (one 42,500 gallon tank per unit) is currently isolated as a normal suction source to the AFW pumps, when the AFW system is aligned for standby readiness, due to air entrainment concerns. This inventory is not available to meet the CSS requirement during power operation, but is permissible in Modes 3 and 4 during periods of low decay heat (defined as after refueling) with both Auxiliary Feedwater Condensate Storage Tanks (CACSTs) aligned at full volume to the outage unit.

APPLICABLE SAFETY ANALYSES

The SNSWP provides cooling water to remove decay heat and to cool down the unit following all events in the accident analysis as discussed in the UFSAR, Chapters 6 and 15 (Refs. 2 and 3, respectively). Because of the water quality, the SNSWP is not used for the normal source of water to the AFW system. The SNSWP serves as a backup source to supply only when the CSS can not supply AFW.

LCO

In order to satisfy recommendations made for the sizing of the system, the CSS contains sufficient cooling water to remove decay heat for 2 hours following a reactor trip from 100% RTP, and then to cool down the RCS to RHR entry conditions, assuming a natural circulation cooldown. In doing this, it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during cooldown.

The CSS level required is equivalent to a capacity ≥ 225,000 gallons, which is based on holding the unit in MODE 3 for 2 hours, followed by a 5 hour cooldown to RHR entry conditions at 50°F/hour. The OPERABILITY of the CSS is determined by maintaining the tanks' levels at or above the minimum required volume.

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the CSS is required to be OPERABLE.

In MODE 5 or 6, the CSS is not required because the AFW System is not required.

ACTIONS

A.1 and A.2

If the CSS inventory is not within limits, the OPERABILITY of the assured supply should be verified by administrative means within 4 hours and once every 12 hours thereafter. The assured supply is considered the Nuclear Service Water System (NSWS) and ultimately the SNSWP. OPERABILITY of the assured feedwater supply must include verification that the flow paths from the assured water supply to the AFW pumps are OPERABLE, and that the assured supply has the required volume of water available. The CSS must be restored to OPERABLE status within 7 days, because the assured supply may be performing this function in addition to its normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the assured water supply. The 7 day Completion Time is reasonable, based on an OPERABLE assured water supply being available, and the low probability of an event occurring during this time period requiring the CSS.

B.1 and B.2

If the CSS cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least

BASES

ACTIONS (continued)

MODE 3 within 6 hours, and in MODE 4, without reliance on the steam generator for heat removal, within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the CSS contains the required inventory of cooling water. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 10.4.
2. UFSAR, Chapter 6.
3. UFSAR, Chapter 15.

B 3.7 PLANT SYSTEMS

B 3.7.12 Auxiliary Building Filtered Ventilation Exhaust System (ABFVES)

BASES

BACKGROUND

The ABFVES consists of two independent and redundant trains. Each train consists of a heater demister section and a filter unit section. The heater demister section consists of a prefilter/moisture separator (to remove entrained water droplets) and an electric heater (to reduce the relative humidity of air entering the filter unit). The filter unit section consists of a prefilter, an upstream HEPA filter, an activated carbon adsorber (for the removal of gaseous activity, principally iodines), a downstream HEPA, and a fan. The downstream HEPA filter is not credited in the accident analysis, but serves to collect carbon fines. Ductwork, valves or dampers, and instrumentation also form part of the system. Following receipt of a safety injection (SI) signal, the system isolates non safety portions of the ABFVES and exhausts air only from the Emergency Core Cooling System (ECCS) pump rooms.

The ABFVES is normally aligned to bypass the system HEPA filters and carbon adsorbers. During emergency operations, the ABFVES dampers are realigned to the filtered position, and fans are started to begin filtration. During emergency operations, the ABFVES dampers are realigned to isolate the non-safety portions of the system and only draw air from the ECCS pump rooms, as well as the Elevation 522 pipe chase, and Elevation 543 and 560 mechanical penetration rooms.

The ABFVES is discussed in the UFSAR, Sections 6.5, 9.4, 14.4, and 15.6 (Refs. 1, 2, 3, and 4, respectively) since it may be used for normal, as well as post accident, atmospheric cleanup functions. The heaters are not required for OPERABILITY, since the laboratory test of the carbon is performed at 95% relative humidity, but have been maintained in the system to provide additional margin (Ref. 9).

BASES

APPLICABLE SAFETY ANALYSES

The design basis of the ABFVES is established by the large break LOCA. The system evaluation assumes filtered and unfiltered leak rates in the Auxiliary Building throughout the accident. In such a case, the system limits radioactive release to within the 10 CFR 50.67 (Ref. 6) limits. The analysis of the effects and consequences of a large break LOCA is presented in Reference 4.

The ABFVES satisfies Criterion 3 of 10 CFR 50.36 (Ref. 7).

LCO

Two independent and redundant trains of the ABFVES are required to be OPERABLE to ensure that at least one is available, assuming that a single failure disables the other train coincident with a loss of offsite power. Total system failure could result in the atmospheric release from the ECCS pump rooms exceeding 10 CFR 50.67 limits in the event of a Design Basis Accident (DBA).

ABFVES is considered OPERABLE when the individual components necessary to maintain the ECCS pump rooms filtration are OPERABLE in both trains.

An ABFVES train is considered OPERABLE when its associated:

a. Fan is OPERABLE;

b. HEPA filters and carbon adsorbers are capable of performing their filtration functions; and

c. Ductwork, valves, and dampers are OPERABLE and air circulation can be maintained.

The ABFVES fans power supply is provided by buses which are shared between the two units. A shutdown unit supplying its associated emergency power source (1EMXG/2EMXH) cannot be credited for OPERABILITY of components supporting the operating unit. If normal or emergency power to the ABFVES becomes inoperable, then the Required Actions of this LCO must be entered independently for each unit that is in the MODE of applicability of the LCO.

BASES

LCO (continued)

The LCO is modified by a Note allowing the ECCS pump rooms pressure boundary to be opened intermittently under administrative controls. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for ECCS pump rooms pressure boundary isolation is indicated.

APPLICABILITY

In MODES 1, 2, 3, and 4, the ABFVES is required to be OPERABLE consistent with the OPERABILITY requirements of the ECCS.

In MODE 5 or 6, the ABFVES is not required to be OPERABLE since the ECCS is not required to be OPERABLE.

ACTIONS

A.1

With one ABFVES train inoperable, action must be taken to restore OPERABLE status within 7 days. During this time, the remaining OPERABLE train is adequate to perform the ABFVES function.

The 7 day Completion Time is appropriate because the risk contribution is less than that for the ECCS (72 hour Completion Time), and this system is not a direct support system for the ECCS. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

Concurrent failure of two ABFVES trains would result in the loss of functional capability; therefore, LCO 3.0.3 must be entered immediately.

B.1

If the ECCS pump rooms pressure boundary is inoperable such that the ABFVES trains cannot establish or maintain the required pressure, action must be taken to restore an OPERABLE ECCS pump rooms pressure boundary within 24 hours. During the period that the ECCS pump rooms pressure boundary is inoperable, appropriate compensatory measures (consistent with the intent, as applicable, of GDC 19, 60, 64, and 10 CFR 50.67) should be utilized to protect plant personnel from potential

BASES

ACTIONS (continued)

hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the ECCS pump rooms pressure boundary.

C.1 and C.2

If the ABFVES train or ECCS pump rooms pressure boundary cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1 and D.2

With one or more ABFVES heaters inoperable, the heater must be restored to OPERABLE status within 7 days. Alternatively, a report must be initiated per Specification 5.6.6, which details the reason for the heater's inoperability and the corrective action required to return the heater to OPERABLE status.

The heaters do not affect OPERABILITY of the ABFVES filter trains because carbon adsorber efficiency testing is performed at 30°C and 95% relative humidity. The accident analysis shows that site boundary radiation doses are within 10 CFR 50.67 limits during a DBA LOCA under these conditions.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

Systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once a month provides an adequate check on this system. Operation for ≥ 15 continuous minutes

SURVEILLANCE REQUIREMENTS (continued)

demonstrates OPERABILITY of the system. Periodic operation ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.7.12.2

This SR verifies that the required ABFVES testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The ABFVES filter tests are in accordance with Reference 5. The VFTP includes testing HEPA filter performance, carbon adsorbers efficiency, system flow rate, and the physical properties of the activated carbon (general use and following specific operations). The system flow rate determination and in-place testing of the filter unit components is performed in the normal operating alignment with both trains in operation. Flow through each filter unit in this alignment is approximately 30,000 cfm. The normal operating alignment has been chosen to minimize normal radiological protection concerns that occur when the system is operated in an abnormal alignment for an extended period of time. Operation of the system in other alignments may alter flow rates to the extent that the 30,000 cfm $\pm$10% specified in Technical Specification 5.5.11 will not be met. Flow rates outside the specified band under these operating alignments will not require the system to be considered inoperable.

Certain postulated failures and post accident recovery operational alignments may result in post accident system operation with only one train of ABFVES in a "normal" alignment. Under these conditions system flow rate is expected to increase above the normal flow band specified in Technical Specification 5.5.11. An analysis has been performed which conservatively predicts the maximum flow rate under these conditions is approximately 37,000 cfm. 37,000 cfm corresponds to a face velocity of approximately 48 ft/min that is significantly more than the normal 40 ft/min velocity specified in ASTM D3803-1989 (Ref. 10). Therefore, the laboratory test of the carbon penetration is performed in accordance with ASTM D3803-1989 and Generic Letter 99-02 at a face velocity of 48 ft/min. These test results are to be adjusted for a 2.27 inch bed using the methodology presented in ASTM D3803-1989 prior to comparing them to the Technical Specification 5.5.11 limit. Specific test Frequencies and additional information are discussed in detail in the VFTP.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.3

This SR verifies that each ABFVES train starts and operates with flow through the HEPA filters and carbon adsorbers on an actual or simulated actuation signal. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.7.12.4

This SR verifies the pressure boundary integrity of the ECCS pump rooms. The following rooms are considered to be ECCS pump rooms (with respect to the ABFVES): centrifugal charging pump rooms, safety injection pump rooms, residual heat removal pump rooms, and the containment spray pump rooms. Although the containment spray system is not normally considered an ECCS system, it is included in this ventilation boundary because of its accident mitigation function which requires the pumping of post accident containment sump fluid. The Elevation 522 pipe chase area is also maintained at a negative pressure by the ABFVES. Since the Elevation 543 and 560 mechanical penetration rooms communicate directly with the Elevation 522 pipe chase area, these penetration rooms are also maintained at a negative pressure by the ABFVES. The ability of the system to maintain the ECCS pump rooms at a negative pressure, with respect to potentially unfiltered adjacent areas, is periodically tested to verify proper functioning of the ABFVES. Upon receipt of a safety injection signal to initiate LOCA operation, the ABFVES is designed to maintain a slight negative pressure in the ECCS pump rooms, with respect to adjacent areas, to prevent unfiltered LEAKAGE. The ABFVES will continue to operate in this mode until the safety injection signal is reset. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

BASES

REFERENCES

1. UFSAR, Section 6.5.
2. UFSAR, Section 9.4.
3. UFSAR, Section 14.4.
4. UFSAR, Section 15.6.
5. Regulatory Guide 1.52 (Rev. 2).
6. 10 CFR 50.67.
7. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
8. Not used.
9. Catawba Nuclear Station License Amendments 90/84 for Units 1/2, August 23, 1991.
10. ASTM D3803-1989.

Enclosure 3

SLC Manual Insertion/Removal Instructions

Remove and Insert

Replace the following page(s) of Catawba Nuclear Station Selected Licensee Commitments (SLC) Manual with the attached revised page(s). The revised page(s) are identified by Section number and contains marginal lines indicating the areas of change.

REMOVE THESE PAGES	**INSERT THESE PAGES**
LIST OF EFFECTIVE SECTIONS	
Pages 1-5 Revision 72	Pages 1-5 Revision 79

List of Effective Sections Revisions 73-78 have been superseded by Revision 79 and therefore do not need to be replaced in your manuals.

REMOVE THESE PAGES	**INSERT THESE PAGES**
TAB 16.5	
16.5-9, Pages 1-3 Revision 1	16.5-9, Pages 1-3 Revision 2
TAB 16.7	
16.7-9, Pages 1-6 Revision 10	16.7-9, Pages 1-6 Revision 11
TAB 16.9	
16.9-3, Pages 1-7 Revision 4	16.9-3, Pages 1-7 Revision 5
16.9-5, Pages 1-22 Revision 8	16.9-5, Pages 1-22 Revision 10

SLC 16.9-5, Revision 9 was superseded by Revision 10 and although included in this package, does not need to be placed in your manual.

REMOVE THESE PAGES	**INSERT THESE PAGES**
16.9-6, Pages 1-13 Revision 11	16.9-6, Pages 1-13 Revision 12
16.9-26, Pages 1-4 Revision 0	16.9-26, Pages 1-5 Revision 1

TAB 16.13

16.13-4, Pages 1-5 Revision 1	16.13-4, Pages 1-5 Revision 2

If you have any questions concerning the contents of this Catawba Nuclear Station Selected Licensee Commitments (SLC) Manual update, please contact Toni Lowery at (803) 701-5046.

Enclosure 4

SLC Manual Replacement Pages

LIST OF EFFECTIVE SECTIONS

SECTION	REVISION NUMBER	REVISION DATE
TABLE OF CONTENTS	15	05/10/16
16.1	1	08/27/08
16.2	2	08/21/09
16.3	1	08/21/09
16.5-1	4	09/15/16
16.5-2	Deleted	
16.5-3	1	02/20/04
16.5-4	0	10/09/02
16.5-5	1	01/28/10
16.5-6	1	08/21/09
16.5-7	2	02/06/15
16.5-8	2	12/22/08
16.5-9	2	11/06/18
16.5-10	Deleted	
16.6-1	0	10/09/02
16.6-2	Deleted	
16.6-3	1	08/21/09
16.6-4	1	08/21/09
16.6-5	2	01/09/13
16.7-1	1	08/21/09
16.7-2	4	02/03/11
16.7-3	4	07/27/13
16.7-4	2	08/21/09
16.7-5	2	08/21/09

LIST OF EFFECTIVE SECTIONS

SECTION	REVISION NUMBER	REVISION DATE
16.7-6	3	06/10/16
16.7-7	1	08/21/09
16.7-8	2	08/21/09
16.7-9	11	07/18/18
16.7-10	7	03/28/16
16.7-11	1	08/21/09
16.7-12	1	08/21/09
16.7-13	3	06/10/16
16.7-14	1	08/21/09
16.7-15	1	08/21/09
16.7-16	0	06/08/09
16.7-17	0	02/10/15
16.7-18	0	05/10/16
16.8-1	6	12/10/15
16.8-2	2	02/20/12
16.8-3	1	10/24/06
16.8-4	2	11/05/07
16.8-5	3	08/21/09
16.9-1	9	12/18/17
16.9-2	6	08/03/17
16.9-3	5	07/03/18
16.9-4	5	09/11/17
16.9-5	10	07/12/18
16.9-6	12	07/03/18

LIST OF EFFECTIVE SECTIONS

SECTION	REVISION NUMBER	REVISION DATE
16.9-7	4	08/21/09
16.9-8	5	08/21/09
16.9-9	3	08/21/09
16.9-10	5	08/21/09
16.9-11	3	08/21/09
16.9-12	3	02/10/15
16.9-13	4	09/27/16
16.9-14	1	09/25/06
16.9-15	2	08/21/09
16.9-16	2	08/21/09
16.9-17	0	10/09/02
16.9-18	0	10/09/02
16.9-19	3	02/20/12
16.9-20	0	10/09/02
16.9-21	1	10/13/16
16.9-22	1	08/21/09
16.9-23	5	08/03/17
16.9-24	2	10/24/06
16.9-25	2	08/21/09
16.9-26	1	11/15/18
16.10-1	1	08/21/09
16.10-2	1	10/24/06
16.10-3	1	08/21/09
16.11-1	1	07/27/13

LIST OF EFFECTIVE SECTIONS

SECTION	REVISION NUMBER	REVISION DATE
16.11-2	4	02/10/15
16.11-3	0	10/09/02
16.11-4	1	08/21/09
16.11-5	0	10/09/02
16.11-6	3	08/03/15
16.11-7	10	11/29/17
16.11-8	0	10/09/02
16.11-9	0	10/09/02
16.11-10	1	08/21/09
16.11-11	1	03/20/03
16.11-12	0	10/09/02
16.11-13	1	07/27/13
16.11-14	0	10/09/02
16.11-15	0	10/09/02
16.11-16	1	10/24/11
16.11-17	0	10/09/02
16.11-18	1	08/21/09
16.11-19	0	10/09/02
16.11-20	2	03/28/16
16.11-21	0	10/09/02
16.12-1	0	10/09/02
16.13-1	1	08/03/17
16.13-2	Deleted	
16.13-3	Deleted	

LIST OF EFFECTIVE SECTIONS

SECTION	REVISION NUMBER	REVISION DATE
16.13-4	2	03/11/18

16.5 REACTOR COOLANT SYSTEM

16.5-9 Operation with Irradiated Fuel in the Core with Steam Generator (SG) Nozzle Dams Installed or Cold Leg Opening $\geq$ 1 in^2

COMMITMENT The Reactor Coolant System (RCS) shall be properly vented when SG nozzle dams are in use or the RCS cold leg is open $\geq$ 1 in^2.

APPLICABILITY: Whenever irradiated fuel is in the reactor vessel.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. COMMITMENT(S) not met.	A.1 Initiate action to restore compliance with COMMITMENT(S).	Immediately

TESTING REQUIREMENTS None

BASES Generic Letter 88-17 and NUREG-1410 involve concerns associated with a loss of residual heat removal. Numerous events have occurred in the industry that resulted in a loss of residual heat removal. This is of great concern due to the potential for substantial core damage occurring in a relatively short time period.

The vent path is provided to address Expeditious Action (7) of Generic Letter 88-17. The vent path is provided to assure that the nozzle dams are not dislodged due to pressurization that could result on a loss of decay heat removal event resulting in core boiling. In addition, in the absence of a sufficient vent path, pressurization in combination with cold leg openings could allow water to be ejected from the vessel and lead to core uncovery.

Vent path 3 is restricted to the nozzle dams analyzed in Reference (Ref.) 9 for Unit 1 and Ref. 10 for Unit 2.

This Selected Licensee Commitment depicts those commitments which are extremely important to nuclear safety, however, are not presently covered by Technical Specifications.

The vent path may be satisfied by:

BASES (continued)

1. On the vented loop, no hot leg nozzle dam installed

   AND

   Removal of either of the following:

   a. A hot leg diaphragm and manway

   OR

   b. A cold leg diaphragm and manway

OR

2. Reactor vessel head is removed.

OR

3. All three Pressurizer Code Safety Valves have been removed.

This option does NOT apply if there is a cold leg opening (this includes potential openings that could develop when pressurization occurs such as performing maintenance on Reactor Coolant Pump seals while on the backseat). Option 3 may only be used when all of the following conditions are satisfied:

1. The reactor has been shutdown for at least 12 days.
2. The reactor has been refueled.
3. Reactor coolant level is at least 12 inches below the reactor vessel flange (≤ 21% wide range level).

REFERENCES

1. Generic Letter 88-17, "Loss of Decay Heat Removal."
2. NUREG-1410, "Loss of Vital AC Power and Residual Heat Removal during Mid-Loop Operation at Vogtle."
3. AD-WC-CNS-0420, "Catawba Nuclear Station Shutdown Risk Management."
4. Catawba Design Basis Specification for the Reactor Coolant (NC) System.
5. Catawba Nuclear Station responses to Generic Letter 88-17, dated January 3, 1989 and February 2, 1989.

6. DPC-1552.08-00-0017, "Analysis of Potential NC System Vent Paths During Nozzle Seal Dam Installation and Use."

7. NUREG-1449, "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States."

8. NRC letter to Hal Tucker, "Comments on Expeditious Action and Notice of Audit on Generic Letter 88-17, McGuire and Catawba Nuclear Stations, Units 1 and 2," dated May 17, 1989.

9. CNC-1201.37-00-0059, "Nozzle Dam Design Report."

10. CNC-1223.03-00-0039, "Engineering Analysis for Unit 2 Type WR-2 Steam Generator Nozzle Dams."

11. DPC-1552.08-00-0323, "GOTHIC Analysis of NC system Vent Paths During Steam Generator Nozzle Dam Use."

16.7 INSTRUMENTATION

16.7-9 Standby Shutdown System (SSS)

COMMITMENT The SSS shall be FUNCTIONAL.

APPLICABILITY: MODES 1, 2, and 3.

REMEDIAL ACTIONS

---NOTE---
SLC 16.2.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SSS non-functional.	A.1 Restore SSS to FUNCTIONAL status.	7 days
B. Total accumulative LEAKAGE from unidentified LEAKAGE, identified LEAKAGE, and reactor coolant pump seal LEAKAGE > 20 gpm.	B.1 Declare the standby makeup pump non-functional and enter Condition A.	Immediately
C. A required cell in a 24-Volt battery bank is < 1.36 volts on float charge.	C.1 Enter Condition A.	Immediately
D. Required Action and associated Completion Time of Condition A not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4.	6 hours 12 hours

TESTING REQUIREMENTS

	TEST	FREQUENCY
TR 16.7-9-1	Verify that the electrolyte level of each SSS diesel starting 24-Volt battery is ≥ the low mark and ≤ the high mark.	7 days
TR 16.7-9-2	Verify that the overall SSS diesel starting 24-Volt battery voltage is ≥ 24 volts on float charge.	7 days
TR 16.7-9-3	Verify that the requirements of SLC 16.9-21 are met and the boron concentration in the storage pool is ≥ the minimum specified in the COLR.	7 days
TR 16.7-9-4	Verify the fuel level in the SSS diesel generator fuel storage tank is ≥ 67 inches.	31 days
TR 16.7-9-5	Verify the SSS diesel generator starts from ambient conditions and operates for ≥ 30 minutes at ≥ 700 kW.	31 days
TR 16.7-9-6	Verify that the electrolyte level of each SSS 250/125-Volt battery is above the plates.	31 days
TR 16.7-9-7	Verify the total SSS 250/125-Volt battery terminal voltage is ≥ 258/129 volts on float charge.	31 days
TR 16.7-9-8	Perform CHANNEL CHECK of each SSS instrumentation device.	31 days
TR 16.7-9-9	Verify the fuel oil properties of new and stored fuel oil for the SSS diesel generator are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program
TR 16.7-9-10	Verify that the individual battery cell voltage of the required cells in the SSS diesel starting 24-Volt battery is ≥ 1.36 volts on float charge.	92 days

(continued)

TESTING REQUIREMENTS (continued)

TEST		FREQUENCY
TR 16.7-9-11	Verify that the Standby Makeup Pump's developed head at the test flow point is ≥ the required developed head, in accordance with the Inservice Testing Program.	92 days
TR 16.7-9-12	Verify that the specific gravity of the SSS 250/125-Volt battery is appropriate for continued service of the battery.	92 days
TR 16.7-9-13	Subject the SSS diesel generator to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.	18 months
TR 16.7-9-14	Verify that the SSS diesel starting 24-Volt batteries, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration.	18 months
TR 16.7-9-15	Verify that the SSS diesel starting 24-Volt battery-to-battery and terminal connections are clean, tight, and free of corrosion.	18 months
TR 16.7-9-16	Verify that the SSS 250/125-Volt batteries, cell plates, and battery racks show no visual indications of physical damage or abnormal deterioration.	18 months
TR 16.7-9-17	Verify that the SSS 250/125-Volt battery-to-battery and terminal connections are clean, tight, free of corrosion, and coated with anti-corrosion material.	18 months
TR 16.7-9-18	Verify that the steam turbine driven auxiliary feedwater pump and controls from the Standby Shutdown Facility function as designed from the SSS.	18 months
TR 16.7-9-19	Perform CHANNEL CALIBRATION of each SSS instrumentation device.	18 months

(continued)

TESTING REQUIREMENTS (continued)

TEST		FREQUENCY
TR 16.7-9-20	Verify proper installation of pressurizer insulation.	18 months
TR 16.7-9-21	Verify pressurizer heaters powered from the SSS have a capacity of ≥ 65 kW measured at motor control center SMXG.	18 months
TR 16.7-9-22	Verify flowpath from the reactor vessel head through the valves powered from the SSS is unobstructed.	18 months

BASES

The SSS is designed to mitigate the consequences of certain postulated fire, security, and station blackout incidents by providing capability to maintain MODE 3 conditions and by controlling and monitoring vital systems from locations external to the main control room. This capability is consistent with the requirements of 10 CFR Part 50.48(c).

The TESTING REQUIREMENTS ensure that the SSS systems and components are capable of performing their intended functions. The required level in the SSS diesel generator fuel storage tank ensures sufficient fuel for 72 hours uninterrupted operation. It is assumed that, within 72 hours, either offsite power can be restored or additional fuel can be added to the storage tank.

Although the standby makeup pump is not nuclear safety related and was not designed according to ASME Code requirements, it is tested quarterly to ensure its FUNCTIONALITY. The TESTING REQUIREMENT concerning the standby makeup pump water supply ensures that an adequate water volume is available to supply the pump continuously for 72 hours.

Total accumulative LEAKAGE is calculated in the NC System Leakage Calculation procedure as identified + unidentified + seal leakoff (References 2 and 3). The REMEDIAL ACTION limit of 20 gpm total accumulative LEAKAGE provides additional margin to allow for instrument inaccuracy, and for the predicted increase in seal leakoff rate due to heatup of the reactor coolant pump seal injection water supply temperature following the SSS event (due to spent fuel pool heatup). Following the increase in seal injection temperature, the standby makeup pump flow of 26 gpm is sufficient to provide in excess of this total accumulative LEAKAGE, thereby assuring that reactor coolant system inventory is maintained at MODE 3 conditions. The supporting evaluation is provided in CNC-1223.04-00-0072 (Ref. 4).

A visual inspection of the diesel starting 24-volt batteries, cell plates, and battery racks provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance. Since the battery cell jars are not transparent, a direct visual inspection of the cell plates cannot be performed. Instead, the cell plates are inspected for physical damage and abnormal deterioration by: 1) visually inspecting the jar sides of each cell for excessive bowing and/or deformation, and 2) visually inspecting the electrolyte of each cell for abnormal appearance.

Verifying individual cell voltage while on float charge for the SSS diesel starting 24-Volt batteries ensures that each cell is capable of supporting its intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery (or battery cell) and maintain the battery (or battery cell) in a fully charged state. The battery cell voltage limit of 1.36 volts is consistent with the nominal design voltage of the battery and is based on the manufacturer's recommended minimum float charge voltage for a fully charged cell with adequate capacity. The 24-Volt starting battery is designed with two battery banks, each battery bank contains 20 individual battery cells. The 24-Volt starting battery has sufficient capacity margin to maintain SSS diesel starting functionality with one cell in each battery bank to be fully degraded with a voltage < 1.36 volts. The 24-Volt starting battery also has sufficient capacity margin to maintain SSS diesel starting functionality with one cell bypassed in each battery bank. The 24-Volt starting battery is required to have 19 individual battery cells per battery bank to maintain SSS diesel starting functionality with sufficient capacity margin. The battery sizing calculation accounts for one degraded cell in each battery bank by assuming the degraded cells undergo a worst case polarity reversal during SSS diesel starting. The supporting evaluation is provided in CNC-1381.06-00-0056 (Ref. 12).

Verification of proper installation of pressurizer insulation ensures that pressurizer heat losses during an SSS event do not exceed the capacity of the pressurizer heaters powered from the SSS.

Testing of the pressurizer heater capacity ensures the full capacity of the heaters is available to maintain a steam bubble in the pressurizer during an SSS event. The acceptance criterion includes an allowance for the voltage drop in the power cables between the SSS and the pressurizer.

Testing of the flowpath from the reactor vessel head to the pressurizer relief tank ensures sufficient flow capacity for reactor coolant inventory control during an SSS event.

REFERENCES

1. Letter from NRC to Gary R. Peterson, Duke, Issuance of Improved Technical Specifications Amendments for Catawba, September 30, 1998.

2. PT/1(2)/A/4150/001D, NC System Leakage Calculation.

3. PT/1(2)/A/4150/001I, Manual NC Leakage Calculation.

4. CNC-1223.04-00-0072, Reactor Coolant Pumps No. 1 Seal Leakoff Annunciator Alarm Setpoint for Unit 1 and Unit 2.

5. CNS-1560.SS-00-0001, Design Basis Specification for the Standby Shutdown Facility.

6. Catawba Technical Specification Amendments 206/200, July 10, 2003.

7. Catawba UFSAR, Section 18.2.4.

8. Catawba License Renewal Commitments, CNS-1274.00-00-0016, Section 4.5.

9. CNC-1223.03-00-0033, Determination of Pressurizer Heater Capacity Powered from the SSF Diesel.

10. Catawba Nuclear Station 10 CFR 50.48(c) Fire Protection Safety Evaluation (SE).

11. 10 CFR 50.48(c), Fire Protection.

12. CNC-1381.06-00-0056, SSF Diesel Generator Battery Sizing Calculation.

16.9 AUXILIARY SYSTEMS

16.9-3 CO_2 Systems

COMMITMENT

The following High Pressure and Low Pressure CO_2 Systems shall be FUNCTIONAL:

a. Low Pressure CO_2 System –

Diesel Generator Building 1A
Diesel Generator Building 1B
Diesel Generator Building 2A
Diesel Generator Building 2B

b. High Pressure CO_2 System –

Description	Associated Fire Detection Instrument (EFA) Zone
U1 Turbine Driven CA Pump Pit	9
1A Motor Driven CA Pump Pit	9
1B Motor Driven CA Pump Pit	9
U2 Turbine Driven CA Pump Pit	19
2A Motor Driven CA Pump Pit	19
2B Motor Driven CA Pump Pit	19

APPLICABILITY:

Whenever equipment protected by the systems is required to be OPERABLE.

--NOTE--

Non-functional or breached fire barrier features (walls, floors, ceilings, doors, dampers, and penetration seals) affect CO_2 System FUNCTIONALITY, since openings affect the ability to achieve and maintain CO_2 concentrations within the room.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Low Pressure CO_2 Systems non-functional.	A.1.1 Verify backup fire suppression equipment is available.	1 hour
	AND	
	A.1.2 Establish continuous fire watch.	1 hour
	OR	
	A.2.1 Verify backup fire suppression equipment is available.	1 hour
	AND	
	A.2.2 Verify fire barrier between Train A and Train B diesel generator rooms is FUNCTIONAL.	1 hour
	AND	
	A.2.3 Establish hourly fire watch.	1 hour
	OR	
	A.3 Complete an evaluation as permitted by NRC RIS 2005-07 to implement Required Action(s)	Prior to terminating Required Action A.1 or A.2
B. One or more High Pressure CO_2 Systems non-functional.	B.1.1 Verify backup fire suppression equipment is available.	1 hour
	AND	
	B.1.2 Establish a continuous fire watch.	1 hour
	OR	(continued)

REMEDIAL ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2.1 Verify backup fire suppression equipment is available.	1 hour
	AND	
	B.2.2 Verify affected area has FUNCTIONAL fire detection instrumentation.	1 hour
	AND	
	B.2.3 Establish hourly fire watch.	1 hour
	OR	
	B.3 Complete an evaluation as permitted by NRC RIS 2005-07 to implement Required Action(s).	Prior to terminating Required Action B.1 or B.2

Testing Requirements

	TEST	FREQUENCY
TR 16.9-3-1	Verify that each manual, power operated, or automatic valve in the flow path is in the correct position.	In accordance with performance based criteria in BASES
TR 16.9-3-2	Verify that the Low Pressure CO_2 System storage tank level is > 44% of full capacity.	31 days
TR 16.9-3-3	Verify that the weight of each High Pressure CO_2 System storage cylinder is ≥ 90% full charge weight.	In accordance with performance based criteria in BASES
TR 16.9-3-4	Perform a visual inspection of all spot type heat detectors in each initiating device circuit.	12 months
TR 16.9-3-5	------------------------------NOTE------------------------------ Different detector(s) shall be selected for each test such that all detectors are tested within 5 years. --- Simulate system actuation by applying a heat source to one or more restorable spot type heat detector(s) in each initiating device circuit.	12 months
TR 16.9-3-6	Verify that the supervisory function of the alarm circuits operates properly and provides correct alarm response for each initiating device circuit.	12 months
TR 16.9-3-7	Verify each Low Pressure CO_2 System actuates manually and automatically, upon receipt of a simulated actuation signal.	18 months
TR 16.9-3-8	Verify that the Low Pressure CO_2 System normal and emergency ventilation system fans receive an “off” signal upon system operation.	18 months

(continued)

TESTING REQUIREMENTS (continued)

TEST		FREQUENCY
TR 16.9-3-9	Perform a visual inspection of the Low Pressure CO_2 System discharge nozzles to assure no blockage.	18 months
TR 16.9-3-10	Verify each High Pressure CO_2 System actuates manually and automatically, upon receipt of a simulated actuation signal.	18 months
TR 16.9-3-11	Verify that damper closure devices receive an actuation signal upon High Pressure CO_2 System operation.	18 months
TR 16.9-3-12	Perform a visual inspection of the High Pressure CO_2 System discharge nozzles to assure no blockage.	18 months
TR 16.9-3-13	Deleted.	

BASES

The FUNCTIONALITY of the Fire Suppression Systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The Fire Suppression System consists of the water supply/distribution system, sprinkler systems, fire hose stations, fire hydrants, and CO_2 systems. The collective capability of the Fire Suppression Systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility Fire Protection Program.

The proper positioning of RF/RY valves is critical to delivering fire suppression CO_2 at the fire source as quickly as possible. The option of increasing or decreasing the frequency of valve position verification allows the ability to optimize plant operational resources. Should an adverse trend develop with CO_2 Systems valve positions, the frequency of verification shall be increased. Similarly, if the CO_2 Systems valve position trends are positive, the frequency of verification could be decreased. Through programmed trending of CO_2 Systems as found valve positions, the CO_2 fire protection systems will be maintained at predetermined reliability standards. The Fire Protection Engineer is responsible for trending and determining verification frequencies based on the following:

BASES (continued)

Initially the frequency shall be monthly.

Annually review the results of the completed valve position verification procedures.

- If the results demonstrate that the valves are found in the correct position at least 99% of the time, the frequency of conducting the valve position verification may be decreased from monthly to quarterly or quarterly to semiannually or semiannually to annually as applicable. The frequency shall not be extended beyond annually (plus grace period).

- If the results demonstrate that the valves are not found in the correct position at least 99% of the time, the frequency of conducting the valve position verification shall be increased from annually to semiannually or semiannually to quarterly or quarterly to monthly as applicable. The valve position verification need not be conducted more often than monthly

The weight of the high pressure CO_2 cylinders is critical to ensuring proper volume of suppressant can be delivered to the hazard. The option of increasing or decreasing the frequency of weighing the cylinders allows the ability to optimize plant operational resources. Should an adverse trend develop with cylinder weights, the frequency of the inspection shall be increased. Similarly, if the cylinder weight trends are positive, the frequency of verification could be decreased. Through programmed trending of cylinder weights, the volume of high pressure CO_2 will be maintained at predetermined reliability standards. The Fire Protection Engineer is responsible for trending and determining inspection frequencies based on the following:

Initially the frequency shall be semi-annual.

Annually review the results of the completed cylinder weighing procedure.

If the results demonstrate that the cylinder weights are found acceptable at least 95% of the time, the frequency of conducting the cylinder weighing may be decreased from semi-annual to annually or from annually to 18 months. The frequency shall not be extended beyond 18 months (plus grace period).

If the results demonstrate that the cylinder weights are not found acceptable at least 95% of the time, the frequency of conducting the cylinder weighing shall be increased from 18 months to annually or from annually to semiannually as applicable. The verification need not be conducted more often than semi-annually.

The main bank (9 cylinders) or the reserve bank (9 cylinders) provides a sufficient quantity of CO_2 to totally flood any of the three auxiliary feedwater pump pits with the required design concentration. Therefore, the High Pressure CO_2 System is FUNCTIONAL with the system aligned to either the main or the reserve bank of cylinders. The system is aligned to the main or reserve bank of cylinders by means of a local manual toggle switch.

In the event that portions of the Fire Suppression Systems are non-functional, alternate backup fire fighting equipment is required to be made available in the affected areas until the non-functional equipment is restored to service. When the non-functional fire fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the non-functional equipment is the primary means of fire suppression.

When a required CO2 System is non-functional, the REMEDIAL ACTION is to establish an appropriate fire watch, and verify or establish backup fire suppression in the affected area. The REMEDIAL ACTION allows for the use of either a nearby fire hose station or fire extinguishers as an acceptable means of backup fire suppression. Typically, the preferred choice is to credit a nearby fire hose station as the backup means of suppression. In most cases, nearby fire hose stations exist in the areas affected by a non-functional CO2 System.

The Associated Fire Detection Instrument (EFA) Zone is for area detection associated with Required Actions for Condition B, the high pressure CO2 systems. The diesel generator rooms which are protected by low pressure CO2 systems do not have area detection or a Required Action dependent on EFA detection.

This SLC is part of the Catawba Fire Protection Program and therefore subject to the provisions of the Catawba Renewed Facility Operating License Conditions 2.C.(5).

REFERENCES

1. Catawba UFSAR, Section 9.5.1.
2. Catawba Nuclear Station 10CFR50.48(c) Fire Protection Safety Evaluation (SE).
3. Catawba Plant Design Basis Specification for Fire Protection, CNS-1465.00-00-0006, as revised.
4. Catawba Renewed Facility Operating License Condition 2.C.(5).
5. RIS 2005-07, Compensatory Measures to Satisfy the Fire Protection Program Requirements.

16.9 AUXILIARY SYSTEMS

16.9-5 Fire Rated Assemblies

COMMITMENT All required Fire Rated Assemblies (walls, floors/ceilings, cable enclosures and other fire barriers) and all sealing devices in fire rated assembly penetrations (fire doors, fire dampers, and penetration seals) as shown on the CN-1105 drawing series shall be FUNCTIONAL.

APPLICABILITY: At all times.

-----NOTE-----
Non-functional or breached fire barrier features (walls, floors, ceilings, doors, dampers, and penetration seals) in the diesel generator rooms and the auxiliary feedwater pump rooms may affect CO_2 System FUNCTIONALITY. See SLC 16.9-3, "CO_2 Systems".

REMEDIAL ACTIONS

IF the required Fire Rated Assembly sealing device is a Fire Door, see Table 16.9-5-1

IF the required Fire Rated Assembly sealing device is a Fire Damper see Table 16.9-5-2

IF required Fire Rated Assembly is a Fire Barrier or Penetration Seal:

1. Identify the location of the impaired fire protection feature by elevation, column, and building
2. Verify the wall, floor/ceiling is a committed boundary on the CN-1105 drawing series (if not a committed boundary, SLC 16.9-5 does not apply)
3. Refer to CN-1209-10 series drawings to identify the Fire Area on both sides of the impaired feature
4. IF either of the Fire Areas is identified as High Safety Significant (HSS) (see Table 16.9-5-3) then implement the REQUIRED ACTION **CONDITION A**
5. IF the Fire Areas are not HSS, then identify the associated shutdown trains/methods of the Fire Areas on each side of the barrier using Table 16.9-5-4 and implement the REQUIRED ACTION as identified in the following Chart:

Shutdown Train (Side 1 & Side 2)	*A*	*B*	*SSS*	*A or B*	*A and B*
A	**CONDITION C**	**CONDITION B**	**CONDITION B**	**CONDITION C**	**CONDITION B**
B	**CONDITION B**	**CONDITION C**	**CONDITION B**	**CONDITION C**	**CONDITION B**
SSS	**CONDITION B**	**CONDITION B**	**CONDITION C**	**CONDITION B**	**CONDITION B**
A or B	**CONDITION C**	**CONDITION C**	**CONDITION B**	**CONDITION C**	**CONDITION B**
A and B	**CONDITION B**	**CONDITION B**	**CONDITION B**	**CONDITION B**	**CONDITION C**

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more HSS* required Fire Rated Assemblies is non-functional.	A.1 Establish a continuous fire watch on at least one side of the assembly.	1 hour
	OR	
	A.2.1 Verify at least one side of the assembly has FUNCTIONAL fire detection instrumentation.	1 hour
	AND	
	A.2.2 Establish an hourly fire watch patrol on at least one side of the assembly.	1 hour
	OR	
	A.3 Complete an evaluation as permitted by NRC RIS 2005-07 to institute required action(s).	Prior to terminating Required Action A.1 or A.2

(continued)

REMEDIAL ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more LSS** required Fire Rated Assemblies is non-functional.	B.1 Establish an hourly fire watch on at least one side of the assembly.	1 hour
	OR	
	B.2.1 Verify at least one side of the assembly has FUNCTIONAL fire detection instrumentation.	1 hour
	AND	
	B.2.2 Establish a once per shift fire watch patrol on at least one side of the assembly.	1 hour
	OR	
	B.3 Complete an evaluation as permitted by NRC RIS 2005-07 to institute required action(s).	Prior to terminating Required Action B.1 or B.2

(continued)

REMEDIAL ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more DID*** required Fire Rated Assemblies is non-functional.	C.1 Establish a once per shift fire watch on at least one side of the assembly.	1 hour
	OR	
	C.2 Verify at least one side of the assembly has FUNCTIONAL fire detection instrumentation.	1 hour
	OR	
	C.3 Complete an evaluation as permitted by NRC RIS 2005-07 to institute required action(s).	Prior to terminating Required Action C.1

*High Safety Significant (HSS) Fire Areas containing required Fire Rated Assemblies are defined in Table 16.9-5-3.

**Low Safety Significant (LSS) Fire Areas containing required Fire Rated Assemblies are defined as those areas with a boundary between redundant shutdown trains.

***Defense-in-Depth (DID) Fire Areas containing required Fire Rated Assemblies are defined as analysis compartment boundaries or PRA compartment boundaries that do not meet the HSS or LSS definitions.

TESTING REQUIREMENTS

	TEST	FREQUENCY
TR 16.9-5-1	Verify each HSS and LSS interior unlocked fire door is closed.	24 hours
TR 16.9-5-2	Verify each HSS and LSS locked closed fire door is closed.	7 days
TR 16.9-5-3	Perform an inspection and functional test of the release and closing mechanism and latches for each swinging fire door shown in Table 16.9-5-1.	6 months
TR 16.9-5-4	Perform a visual inspection of the exposed surfaces of each required Fire Rated Assembly.	18 months
TR 16.9-5-5	------------------------------NOTE------------------------------ Any abnormal changes or degradation shall be identified and resolved via the corrective action program. Based on the investigation results, additional dampers may be selected for inspection. Samples will be grouped by unit, system, and train and shall be selected such that each damper is inspected every 15 years. --	
	Perform a visual inspection of fire dampers in each HSS and LSS required Fire Rated Assembly.	18 months, in accordance with the predefined inspection schedule

(continued)

TESTING REQUIREMENTS (continued)

	TEST	FREQUENCY
TR 16.9-5-6	-----NOTE----- Any abnormal changes or degradation shall be identified and resolved via the corrective action program. Based on the investigation results, additional Fire Rated Assemblies may be selected for inspection. Samples shall be selected such that each Fire Rated Assembly is inspected every 15 years. ----- Perform a visual inspection of penetration seals in each HSS AND LSS required Fire Rated Assembly.	18 months, in accordance with the predefined inspection schedule
TR 16.9-5-7	Perform an inspection and functional test of the automatic hold open, release and closing mechanism for each rolling fire door shown in Table 16.9-5-1.	18 Months

Table 16.9-5-1

Required Fire Doors

DOOR NUMBER	BLDG	LOCATION	ELEVATION	FIRE AREA INTERFACE	RISK CRITERIA	REMEDIAL ACTION CONDITION
AX500F	AUX	56, FF	522+0	1/4	DID	C
AX214A	AUX	54-55, FF-GG	543+0	1/4	DID	C
AX214B	AUX	58-59, FF-GG	543+0	1/4	DID	C
AX217D	AUX	52-53, BB	543+0	3/34	LSS	B
AX217F(1)	AUX	51, AA-BB	543+0	3/40	LSS	B
AX217G	AUX	52-53, BB	543+0	3/32	LSS	B
AX227D	AUX	54-55, MM-NN	543+0	4/22	DID	C
AX227E	AUX	59-60, MM-NN	543+0	4/22	DID	C
AX228A	AUX	56-57, EE	543+0	4/9	DID	C
AX228B	AUX	57-58, EE	543+0	4/10	DID	C
AX248	AUX	57-58, QQ	543+0	4/ASB	LSS	B
AX260B	AUX	61-62, BB-CC	543+0	2/36	LSS	B
AX260F(1)	AUX	62, AA-BB	543+0	2/39	LSS	B
AX260G	AUX	61-62, BB-CC	543+0	2/31	LSS	B
AX260H	AUX	61-62, BB-CC	543+0	2/33	LSS	B
T527#1	AUX	52-53, BB-CC	543+0	3/37	LSS	B
AX202	AUX	51, NN	543+0	4/STAIR	DID	C
AX253A	AUX	63, NN	543+0	4/STAIR	DID	C
AX227A	AUX	59, FF-GG	543+0	4/STAIR	DID	C
AX260E	AUX	52, CC	543+0	3/STAIR	DID	C
AX516M	AUX	62, CC	543+0	2/STAIR	DID	C
AX354A	AUX	55, DD-EE	554+0	22/45	LSS	B
AX354B	AUX	59, DD-EE	554+0	22/46	LSS	B
AX418	AUX	57, BB	554+0	9/10	DID	C
AX419	AUX	57, DD-EE	554+0	9/10	DID	C
AX420A	AUX	59, DD-EE	554+0	9/46	LSS	B
AX421A	AUX	55, DD-EE	554+0	10/45	LSS	B
S102A	AUX	53-54, AA	554+0	10/SRV	LSS	B
AX302	AUX	41, CC-DD	556+0	25/41	DID	C
AX304	AUX	41, AA-BB	556+0	26/42	DID	C
AX306	AUX	73, DD-EE	556+0	27/43	DID	C
AX308	AUX	73, BB-CC	556+0	28/44	DID	C
AX348B	AUX	54-55, MM-NN	560+0	11/22	DID	C
AX348C	AUX	53-54, HH	560+0	4/11	DID	C
AX348D	AUX	59-60, MM-NN	560+0	11/22	DID	C
AX348E	AUX	60-61, HH	560+0	4/11	DID	C
AX352B	AUX	53, CC-DD	560+0	6/STAIR	HSS	A
AX352C	AUX	53, CC-DD	560+0	10/STAIR	DID	C
AX352D	AUX	46-47, BB-CC	560+0	6/RB1	HSS	A
AX353	AUX	45, BB	560+0	6/8	HSS	A
AX353B	AUX	45, AA-BB	560+0	8/41	LSS	B
AX353C	AUX	45, AA-BB	560+0	8/42	DID	C

(continued)

Table 16.9-5-1

Required Fire Doors

DOOR NUMBER	BLDG	LOCATION	ELEVATION	FIRE AREA INTERFACE	RISK CRITERIA	REMEDIAL ACTION CONDITION
AX393B	AUX	61, CC-DD	560+0	9/STAIR	DID	C
AX393C	AUX	61, CC-DD	560+0	5/STAIR	DID	C
AX393D	AUX	67-68, BB-CC	560+0	5/RB2	LSS	B
AX394	AUX	69, BB	560+0	5/7	DID	C
AX394B	AUX	69, AA-BB	560+0	7/43	LSS	B
AX394C	AUX	69, AA-BB	560+0	7/44	DID	C
AX395	AUX	61, AA-BB	560+0	7/9	LSS	B
AX396	AUX	53, AA-BB	560+0	8/10	LSS	B
AX415	AUX	45-46, CC-DD	560+0	6/RB1	HSS	A
AX416	AUX	68-69, CC-DD	560+0	5/RB2	LSS	B
AX417	AUX	57, QQ	560+0	11/ASB	LSS	B
AX313D	AUX	51, NN	560+0	11/STAIR	DID	C
AX388B	AUX	63, NN	560+0	11/STAIR	DID	C
AX348	AUX	59, FF-GG	560+0	11/STAIR	DID	C
AX355A	AUX	53-54, FF	568+0	4/11	DID	C
AX355D	AUX	60, FF	568+0	4/11	DID	C
AX355E	AUX	60, FF	568+0	11/STAIR	DID	C
AX515	AUX	54, BB	574+0	17/45	HSS	A
AX516	AUX	56-57, DD	574+0	14/45	HSS	A
AX516A	AUX	57-58, DD	574+0	16/46	HSS	A
AX516K	AUX	57, AA-BB	574+0	16/17	HSS	A
AX517A	AUX	53-54, DD-EE	574+0	22/45	LSS	B
AX517B	AUX	60-61, DD-EE	574+0	22/46	LSS	B
AX517C	AUX	57, DD-EE	574+0	45/46	DID	C
AX517D	AUX	57, DD-EE	574+0	9/46	LSS	B
AX517E	AUX	56-57, DD-EE	574+0	10/46	LSS	B
AX518	AUX	60, BB	574+0	16/46	HSS	A
S303	SRV	36-37, 1N	574+0	45/SRV	DID	C
S303C	SRV	36-37, V	574+0	45/SRV	DID	C
S304A	AUX	60, AA	574+0	46/SRV	DID	C
AX500H	AUX	54-55, MM-NN	577+0	18/22	DID	C
AX500K	AUX	53-54, HH-GG	577+0	4/18	DID	C
AX500L	AUX	59-60, MM-NN	577+0	18/22	DID	C
AX500N	AUX	60-61, HH-GG	577+0	4/18	DID	C
AX513B	AUX	53, CC-DD	577+0	13/STAIR	HSS	A
AX514	AUX	45, BB	577+0	13/15	HSS	A
AX514B	AUX	45-46, AA-BB	577+0	6/13	HSS	A
AX517	AUX	57, EE	577+0	9/18	DID	C
AX525	AUX	55-56, QQ	577+0	18/ASB	LSS	B
AX525B	AUX	56, QQ	577+0	18/ASB	LSS	B
AX526D	AUX	58, QQ	577+0	18/ASB	LSS	B
A314#3	AUX	61, CC-DD	577+0	12/STAIR	HSS	A
AX533C	AUX	61, CC-DD	577+0	46/STAIR	DID	C
AX534	AUX	69, BB	577+0	12/14	HSS	A

(continued)

Table 16.9-5-1

Required Fire Doors

DOOR NUMBER	BLDG	LOCATION	ELEVATION	FIRE AREA INTERFACE	RISK CRITERIA	REMEDIAL ACTION CONDITION
AX534B	AUX	68-69, AA-BB	577+0	7/14	HSS	A
AX535A	AUX	61, AA-BB	577+0	14/46	HSS	A
AX536	AUX	53, AA-BB	577+0	15/45	HSS	A
AX656	AUX	53, CC-DD	577+0	45/STAIR	DID	C
AX500P	AUX	51, NN	577+0	18/STAIR	DID	C
AX500S	AUX	63, NN	577+0	18/STAIR	DID	C
AX338A	AUX	60, FF-GG	577+0	18/STAIR	DID	C
AX602	AUX	52, UU-VV	594+0	24/ASB	DID	C
AX627	AUX	62, UU-VV	594+0	23/ASB	DID	C
AX630	AUX	58, QQ	594+0	22/ASB	LSS	B
AX632	AUX	57, QQ	594+0	22/ASB	LSS	B
AX635	AUX	60-61, QQ	594+0	22/ASB	LSS	B
AX635E	AUX	53-54, QQ	594+0	22/ASB	LSS	B
AX635F	AUX	53-54, QQ	594+0	22/ASB	LSS	B
AX655	AUX	62-63, DD	594+0	19/48	LSS	B
AX656C	AUX	61, CC-DD	594+0	19/22	LSS	B
AX657	AUX	60-61, CC	594+0	19/22	LSS	B
AX657A(2)	AUX	54, BB	594+0	21/35	HSS	A
AX657B	AUX	52-53, CC-DD	594+0	20/22	LSS	B
AX657E(2)	AUX	53, BB	594+0	21/35	HSS	A
AX657F	AUX	60, DD-EE	594+0	21/22	HSS	A
AX657G	AUX	57-58, DD-EE	594+0	21/22	HSS	A
AX657H	AUX	54, DD-EE	594+0	21/22	HSS	A
AX657J	AUX	53, BB-CC	594+0	20/21	HSS	A
AX658B	AUX	51-52, DD	594+0	20/49	LSS	B
S400	AUX	55-56, AA	594+0	21/SRV	HSS	A
S406	AUX	58-59, AA	594+0	21/SRV	HSS	A
AX635G	AUX	51, NN	594+0	22/STAIR	DID	C
AX635H	AUX	63, NN	594+0	22/STAIR	DID	C
AX654A	AUX	60, FF	594+0	22/STAIR	DID	C
AX654B	AUX	61, CC-DD	594+0	19/STAIR	DID	C
AX665B	AUX	53, CC-DD	594+0	22/STAIR	DID	C
AX700B	AUX	50-51, JJ-KK	605+10	24/RB1	LSS	B
AX700D	AUX	63-64, KK	605+10	22/23	LSS	B
AX701	AUX	50-51, JJ-KK	605+10	22/RB1	LSS	B
AX714B	AUX	63-64, JJ-KK	605+10	23/RB2	LSS	B
AX720	AUX	50-51, HH-JJ	605+10	22/RB1	LSS	B
AX721	AUX	63-64, HH-JJ	605+10	22/RB2	LSS	B
AX714C	AUX	50-51, KK	605+10	22/24	LSS	B
AX715A	AUX	63-64, JJ-KK	605+10	22/RB2	LSS	B
S211(2)	TB1	17, V	568+0	SRV/TB1	DID	C
S212	TB1	19, V	568+0	SRV/TB1	DID	C
S210	TB1	21,V	568+0	SRV/TB1	DID	C

(continued)

Table 16.9-5-1

Required Fire Doors

DOOR NUMBER	BLDG	LOCATION	ELEVATION	FIRE AREA INTERFACE	RISK CRITERIA	REMEDIAL ACTION CONDITION
S206	TB1	22, V	568+0	SRV/TB1	DID	C
S201	TB1	33, V	568+0	SRV/TB1	DID	C
SR3(3)	TB1	30-31, V	568+0	SRV/TB1	DID	C
S201A	TB1	27, V	568+0	SRV/TB1	DID	C
T101	TB1	31, 1K	568+0	TB1/U1 OTT	DID	C
S424	TB1	24-25, V	594+0	SRV/TB1	DID	C
S425	TB1	23, V	594+0	SRV/TB1	DID	C
S426	TB1	22, V	594+0	SRV/TB1	DID	C
SR21(3)	TB1	24, V	594+0	SRV/TB1	DID	C
S472	TB1	27, V	594+0	SRV/TB1	DID	C
S423	TB1	29, V	594+0	SRV/TB1	DID	C
S422	TB1	29, V	594+0	SRV/TB1	DID	C
SR7(3)	TB1	29-30, V	594+0	SRV/TB1	DID	C
S416	TB1	32, V	594+0	SRV/TB1	DID	C
S444	TB1	15, V	594+0	SRV/TB1	DID	C
TR4(3)	TB1	15-16, V	594+0	SRV/TB1	DID	C
T200A	TB1	32, 1J-1K	594+0	TB1/U1 MTOT	DID	C
S701	TB1	22, 1L	619+6	SRV/TB1	DID	C
S704	TB1	33, 1L	619+6	SRV/TB1	DID	C
S209	TB2	20, P	568+0	SRV/TB2	DID	C
S208	TB2	22, P	568+0	SRV/TB2	DID	C
SR2(3)	TB2	32-33, P	568+0	SRV/TB2	DID	C
S462	TB2	32, P	568+0	SRV/TB2	DID	C
SR4(3)	TB2	30-31, P	568+0	SRV/TB2	DID	C
S1102	TB2	27, P	568+0	SRV/TB2	DID	C
T151	TB2	31, 2K	568+0	TB2/U2 OTT	DID	C
S423E	TB2	26, P-Q	594+0	SRV/TB2	DID	C
S416A	TB2	32, P	594+0	SRV/TB2	DID	C
SR8(3)	TB2	29-30, P	594+0	SRV/TB2	DID	C
S422A	TB2	29, P	594+0	SRV/TB2	DID	C
S423A	TB2	29, P	594+0	SRV/TB2	DID	C
S435	TB2	24-25, P	594+0	SRV/TB2	DID	C
S436	TB2	23, P	594+0	SRV/TB2	DID	C
S437	TB2	22, P	594+0	SRV/TB2	DID	C
SR22(3)	TB2	24, P	594+0	SRV/TB2	DID	C
S444A	TB2	15, P	594+0	SRV/TB2	DID	C
SR16(3)	TB2	15-16, P	594+0	SRV/TB2	DID	C
S472A	TB2	27, P	594+0	SRV/TB2	DID	C
T250A	TB2	32, 2J-2K	594+0	TB2/U2 MTOT	DID	C
S701A	TB2	22, 2L	619+6	SRV/TB2	DID	C
S704A	TB2	33, 2L	619+6	SRV/TB2	DID	C
AX662A	NSWPS	----	600+0	29/30	LSS	B

(1) These doors are not equipped with closing mechanisms or latches and are therefore exempt from TESTING REQUIREMENT 16.9-5-3.
(2) These doors are held open with a fusible link.
(3) Rolling Door.

Table 16.9-5-2

Required Fire Dampers

DAMPER NUMBER	BLDG	LOCATION	ELEVATION	FIRE AREA INTERFACE	RISK CRITERIA	REMEDIAL ACTION CONDITION
1VA-FD001	AUX	53/GG-FF	522+0	1/4	DID	C
1VA-FD002	AUX	53/GG-HH	522+0	1/4	DID	C
1VA-FD003	AUX	55-56/GG-HH	522+0	1/4	DID	C
1VA-FD004	AUX	55-56/GG-HH	522+0	1/4	DID	C
1VA-FD005	AUX	54-55/GG-HH	522+0	1/4	DID	C
1VA-FD006	AUX	54-55/GG-HH	522+0	1/4	DID	C
1VA-FD007	AUX	53/GG-FF	522+0	1/4	DID	C
1VA-FD008	AUX	53/GG-FF	522+0	1/4	DID	C
1VA-FD009	AUX	53-54/FF-GG	522+0	1/1 (ND PUMPS)	DID	C
1VA-FD010	AUX	56-57/ GG-HH,	522+0	1/4	DID	C
1VA-FD011	AUX	56-57/FF	522+0	1/4	DID	C
1VA-FD012	AUX	51/NN-PP	543+0	11/STAIR	DID	C
1VA-FD013	AUX	54/MM	543+0	4/22	DID	C
1VA-FD014	AUX	54/MM	543+0	4/22	DID	C
1VA-FD015	AUX	54-55/MM-NN	543+0	4/22	DID	C
1VA-FD016	AUX	54-55/MM-NN	543+0	4/22	DID	C
1VA-FD017	AUX	54-55/MM-NN	543+0	4/22	DID	C
1VA-FD020	AUX	55/JJ-KK	543+0	4/4 (NV PUMPS)	DID	C
1VA-FD033	AUX	51-52/AA-BB	543+0	3/40	LSS	B
1VA-FD034	AUX	51-52/AA-BB	543+0	3/40	LSS	B
1VA-FD035	AUX	52/AA-BB	543+0	3/32	LSS	B
1VA-FD036	AUX	52-53/BB	543+0	3/32	LSS	B
1VA-FD038	AUX	52-53/BB	543+0	3/34	LSS	B
1VA-FD039	AUX	52-53/BB	543+0	3/34	LSS	B
1VA-FD040	AUX	52-53/BB	543+0	3/32	LSS	B
1VA-FD041	AUX	52-53/BB	543+0	3/32	LSS	B
1VA-FD042	AUX	53/CC	543+0	3/STAIR	DID	C
1VA-FD043	AUX	53/CC-DD	543+0	3/STAIR	DID	C
1VA-FD045	AUX	52-53/DD	560+0	3/6	HSS	A
1VA-FD046	AUX	52-53/CC-DD	577+0	6/13	HSS	A
1VA-FD047	AUX	52-53/CC-DD	577+0	6/13	HSS	A
1VA-FD048	AUX	54/MM-NN	560+0	11/22	DID	C
1VA-FD049	AUX	54/MM	560+0	11/22	DID	C
1VA-FD050	AUX	54-55/MM	560+0	4/22	DID	C
1VA-FD051	AUX	54-55/MM	560+0	4/22	DID	C
1VA-FD052	AUX	55/MM-NN	560+0	11/22	DID	C
1VA-FD053	AUX	55/MM	560+0	11/22	DID	C
1VA-FD054	AUX	53/GG-HH	560+0	4/11	DID	C
1VA-FD055	AUX	53/GG-HH	560+0	4/11	DID	C
1VA-FD056	AUX	53/KK	560+0	4/11	DID	C
1VA-FD057	AUX	53/GG-HH	560+0	4/11	DID	C

(continued)

Table 16.9-5-2

REQUIRED FIRE DAMPERS

DAMPER NUMBER	BLDG	LOCATION	ELEVATION	FIRE AREA INTERFACE	RISK CRITERIA	REMEDIAL ACTION CONDITION
1VA-FD058	AUX	53-54/HH	560+0	4/11	DID	C
1VA-FD059	AUX	54/GG-HH	560+0	4/11	DID	C
1VA-FD060	AUX	54/HH	560+0	4/11	DID	C
1VA-FD061	AUX	56-57/QQ	577+0	18/ASB	LSS	B
1VA-FD062	AUX	55-56/QQ	577+0	18/ASB	LSS	B
1VA-FD063	AUX	55/MM-NN	577+0	18/22	DID	C
1VA-FD064	AUX	55/MM	577+0	18/22	DID	C
1VA-FD065	AUX	54/MM	577+0	18/22	DID	C
1VA-FD066	AUX	54/MM	577+0	18/22	DID	C
1VA-FD067	AUX	54/HH	577+0	4/18	DID	C
1VA-FD068	AUX	53-54/HH	577+0	4/18	DID	C
1VA-FD069	AUX	54/GG-HH	577+0	4/18	DID	C
1VA-FD070	AUX	53-54/HH	577+0	4/18	DID	C
1VA-FD071	AUX	53-54/HH	577+0	4/18	DID	C
1VA-FD072	AUX	53/HH	577+0	4/18	DID	C
1VA-FD073	AUX	53/HH	577+0	4/18	DID	C
1VA-FD074	AUX	53/GG-HH	577+0	4/18	DID	C
1VA-FD075	AUX	53-54/KK-LL	594+0	18/22	DID	C
1VA-FD076	AUX	53-54/KK-LL	594+0	18/22	DID	C
1VA-FD078	AUX	57/NN	594+0	22/STAIR	DID	C
1VA-FD087	AUX	55-56/QQ	594+0	22/ASB	LSS	B
1VA-FD088	AUX	53-54/QQ	594+0	22/ASB	LSS	B
1VA-FD133	AUX	53/CC-DD	594+0	22/STAIR	DID	C
1VA-FD139	AUX	51-52/DD	543+0	3/4	DID	C
1VA-FD140	AUX	53-54/FF-GG	560+0	4/11	DID	C
1VA-FD141	AUX	53-54/FF-GG	560+0	4/11	DID	C
1VA-FD142	AUX	53/GG	560+0	4/11	DID	C
1VA-FD143	AUX	53/JJ-HH	560+0	4/11	DID	C
1VA-FD144	AUX	53/KK	560+0	4/11	DID	C
1VA-FD145	AUX	51/KK	560+0	11/18	DID	C
1VA-FD146	AUX	51/KK	560+0	11/18	DID	C
1VA-FD147	AUX	52/MM	560+0	11/18	DID	C
1VA-FD148	AUX	52/MM-NN	560+0	4/11	DID	C
1VA-FD149	AUX	52-53/DD	560+0	3/6	HSS	A
1VA-FD150	AUX	52-53/DD	560+0	3/6	HSS	A
1VA-FD152	AUX	52-53/BB-CC	543+0	3/37	LSS	B
1VA-FD153	AUX	52-53/CC	543+0	3/37	LSS	B
1VA-FD154	AUX	53-54/GG-HH	594+0	4/22	DID	C
1VA-FD155	AUX	53-54/GG-HH	594+0	4/22	DID	C
1VA-FD159	AUX	49-50/AA-BB	543+0	CO2	HSS	A
1VA-FD160	AUX	50-51/AA-BB	543+0	CO2	HSS	A
1VA-FD163	AUX	56/EE	543+0	10/45	LSS	B
1VA-FD164	AUX	56-57/EE	543+0	4/10	DID	C
2VA-FD001	AUX	61/GG-FF	522+0	1/4	DID	C
2VA-FD002	AUX	61/GG-FF	522+0	1/4	DID	C

(continued)

Table 16.9-5-2

REQUIRED FIRE DAMPERS

DAMPER NUMBER	BLDG	LOCATION	ELEVATION	FIRE AREA INTERFACE	RISK CRITERIA	REMEDIAL ACTION CONDITION
2VA-FD003	AUX	60-61/FF-GG	522+0	1/1 (ND PUMPS)	DID	C
2VA-FD004	AUX	61/GG-FF	522+0	1/4	DID	C
2VA-FD005	AUX	60-61/GG-HH	522+0	1/4	DID	C
2VA-FD006	AUX	59-60/GG-HH	522+0	1/4	DID	C
2VA-FD007	AUX	59-60/GG-HH	522+0	1/4	DID	C
2VA-FD008	AUX	58-59/GG-HH	522+0	1/4	DID	C
2VA-FD009	AUX	58-59/GG-HH	522+0	1/4	DID	C
2VA-FD010	AUX	57-58/GG-HH	522+0	1/4	DID	C
2VA-FD011	AUX	57-58/FF	522+0	1/4	DID	C
2VA-FD012	AUX	59-60/MM-NN	543+0	4/22	DID	C
2VA-FD013	AUX	59/MM	543+0	4/22	DID	C
2VA-FD014	AUX	59/MM	543+0	4/22	DID	C
2VA-FD015	AUX	59-60/MM-NN	543+0	4/22	DID	C
2VA-FD020	AUX	63/NN	534+0	4/STAIR	DID	C
2VA-FD023	AUX	59/JJ-KK	543+0	4/4 (NV PUMPS)	DID	C
2VA-FD036	AUX	61-62/DD	560+0	2/5	LSS	B
2VA-FD037	AUX	61-62/CC-DD	577+0	5/12	HSS	A
2VA-FD038	AUX	61-62/CC-DD	577+0	5/12	HSS	A
2VA-FD040	AUX	62-63/AA-BB	543+0	2/39	LSS	B
2VA-FD041	AUX	62-63/AA-BB	543+0	2/39	LSS	B
2VA-FD042	AUX	62/AA-BB	543+0	2/31	LSS	B
2VA-FD043	AUX	61-62/BB	543+0	2/31	LSS	B
2VA-FD045	AUX	61/CC	543+0	2/STAIR	DID	C
2VA-FD046	AUX	61/CC-DD	543+0	2/STAIR	DID	C
2VA-FD048	AUX	61-62/BB	543+0	2/33	LSS	B
2VA-FD049	AUX	61-62/BB	543+0	2/33	LSS	B
2VA-FD050	AUX	61-62/BB	543+0	2/31	LSS	B
2VA-FD051	AUX	61-62/BB	543+0	2/31	LSS	B
2VA-FD053	AUX	60/MM	560+0	11/22	DID	C
2VA-FD054	AUX	59/MM-NN	560+0	11/22	DID	C
2VA-FD056	AUX	60/MM-NN	560+0	11/22	DID	C
2VA-FD057	AUX	59-60/MM	560+0	11/22	DID	C
2VA-FD058	AUX	59-60/MM	560+0	4/22	DID	C
2VA-FD059	AUX	60-61/HH	560+0	4/11	DID	C
2VA-FD060	AUX	61/HH-JJ	560+0	4/11	DID	C
2VA-FD061	AUX	60-61/GG-HH	560+0	4/11	DID	C
2VA-FD062	AUX	61/GG-HH	560+0	4/11	DID	C
2VA-FD063	AUX	61/GG-HH	560+0	4/11	DID	C
2VA-FD064	AUX	60-61/GG-HH	560+0	4/11	DID	C
2VA-FD065	AUX	61/HH	560+0	4/11	DID	C
2VA-FD069	AUX	58-59/QQ	577+0	18/ASB	LSS	B
2VA-FD070	AUX	59-60/QQ	577+0	18/ASB	LSS	B
2VA-FD071	AUX	59-60/MM-NN	577+0	18/22	DID	C
2VA-FD072	AUX	59-60/MM	577+0	18/22	DID	C

(continued)

Table 16.9-5-2

REQUIRED FIRE DAMPERS

DAMPER NUMBER	BLDG	LOCATION	ELEVATION	FIRE AREA INTERFACE	RISK CRITERIA	REMEDIAL ACTION CONDITION
2VA-FD073	AUX	60/MM	577+0	18/22	DID	C
2VA-FD074	AUX	60/MM	577+0	18/22	DID	C
2VA-FD075	AUX	60/HH	577+0	4/22	DID	C
2VA-FD076	AUX	60/HH	577+0	4/22	DID	C
2VA-FD077	AUX	60-61/HH	577+0	4/22	DID	C
2VA-FD078	AUX	60-61/HH	577+0	4/22	DID	C
2VA-FD079	AUX	61/HH	577+0	4/22	DID	C
2VA-FD080	AUX	61/GG-HH	577+0	4/22	DID	C
2VA-FD081	AUX	61/HH	577+0	4/22	DID	C
2VA-FD083	AUX	63/NN	594+0	22/STAIR	DID	C
2VA-FD086	AUX	60/FF	594+0	22/STAIR	DID	C
2VA-FD087	AUX	59-60/QQ	594+0	22/ASB	LSS	B
2VA-FD088	AUX	60-61/QQ	594+0	22/ASB	LSS	B
2VA-FD093	AUX	58-59/QQ	594+0	22/ASB	LSS	B
2VA-FD097	AUX	61/CC-DD	594+0	22/STAIR	DID	C
2VA-FD108A	AUX	57-59/QQ	611+0	22/ASB	LSS	B
2VA-FD108B	AUX	57-59/QQ	611+0	22/ASB	LSS	B
2VA-FD114	AUX	59-60/KK-LL	594+0	18/22	DID	C
2VA-FD115	AUX	59-60/KK-LL	594+0	18/22	DID	C
2VA-FD137	AUX	60-61/FF-GG	560+0	4/18	DID	C
2VA-FD138	AUX	60-61/FF-GG	560+0	4/18	DID	C
2VA-FD139	AUX	61/GG	560+0	4/11	DID	C
2VA-FD141	AUX	62-63/DD	543+0	2/4	DID	C
2VA-FD142	AUX	60-61/KK	560+0	4/11	DID	C
2VA-FD143	AUX	62-63/KK	560+0	4/18	DID	C
2VA-FD144	AUX	63/KK	560+0	4/18	DID	C
2VA-FD145	AUX	61-62/MM-NN	560+0	4/11	DID	C
2VA-FD146	AUX	61-62/DD	560+0	2/5	LSS	B
2VA-FD147	AUX	61-62/DD	560+0	2/5	LSS	B
2VA-FD151	AUX	61-62/BB-CC	543+0	2/36	LSS	B
2VA-FD152	AUX	61-62/CC	543+0	2/36	LSS	B
2VA-FD153	AUX	60-61/GG-HH	594+0	4/22	DID	C
2VA-FD154	AUX	60-61/GG-HH	594+0	4/22	DID	C
2VA-FD157	AUX	63-64/AA-BB	543+0	CO2	HSS	A
2VA-FD158	AUX	64-65/AA-BB	543+0	CO2	HSS	A
2VA-FD160	AUX	57-58/QQ	543+0	4/ASB	LSS	B
2VA-FD161	AUX	57-58/QQ	543+0	4/ASB	LSS	B
2VA-FD163	AUX	58/EE	543+0	9/46	LSS	B
2VA-FD164	AUX	57-58/EE	543+0	4/9	DID	C
0BRS-FD001	AUX	54-55/DD-EE	554+0	10/22	DID	C
0BRS-FD010	AUX	57/DD-EE	554+0	9/10	DID	C
0BRS-FD019	AUX	59/DD-EE	554+0	9/22	DID	C
0BRX-FD001A	AUX	54-55/DD-EE	554+0	10/22	DID	C
0BRX-FD001B	AUX	54-55/DD-EE	554+0	10/22	DID	C

(continued)

Table 16.9-5-2

REQUIRED FIRE DAMPERS

DAMPER NUMBER	BLDG	LOCATION	ELEVATION	FIRE AREA INTERFACE	RISK CRITERIA	REMEDIAL ACTION CONDITION
0BRX-FD001C	AUX	54-55/DD-EE	554+0	10/22	DID	C
0BRX-FD001D	AUX	54-55/DD-EE	554+0	10/22	DID	C
0BRX-FD001E	AUX	54-55/DD-EE	554+0	10/22	DID	C
0BRX-FD001F	AUX	54-55/DD-EE	554+0	10/22	DID	C
0BRX-FD001G	AUX	54-55/DD-EE	554+0	10/22	DID	C
0BRX-FD001H	AUX	54-55/DD-EE	554+0	10/22	DID	C
0BRX-FD002	AUX	54-55/DD-EE	554+0	10/22	DID	C
0BRX-FD009	AUX	57/AA-BB	554+0	9/10	DID	C
0BRX-FD010	AUX	57/AA-BB	554+0	9/10	DID	C
0BRX-FD011	AUX	57/BB-CC	554+0	9/10	DID	C
0BRX-FD012	AUX	57/CC-DD	554+0	9/10	DID	C
0BRX-FD013	AUX	57/CC-DD	554+0	9/10	DID	C
0BRX-FD014	AUX	57/DD-EE	554+0	9/10	DID	C
0BRX-FD021	AUX	60/DD-EE	554+0	9/22	DID	C
0BRX-FD022A	AUX	60/DD-EE	554+0	9/22	DID	C
0BRX-FD022B	AUX	60/DD-EE	554+0	9/22	DID	C
0BRX-FD022C	AUX	60/DD-EE	554+0	9/22	DID	C
0BRX-FD022D	AUX	60/DD-EE	554+0	9/22	DID	C
0BRX-FD022E	AUX	60/DD-EE	554+0	9/22	DID	C
0BRX-FD022F	AUX	60/DD-EE	554+0	9/22	DID	C
0BRX-FD022G	AUX	60/DD-EE	554+0	9/22	DID	C
0BRX-FD022H	AUX	60/DD-EE	554+0	9/22	DID	C
0BRX-FD023	AUX	57/BB-CC	554+0	9/10	DID	C
1CRA-FD005A	AUX	54-55/DD-EE	594+0	21/22	HSS	A
1CRA-FD005B	AUX	54-55/DD-EE	594+0	21/22	HSS	A
1CRA-FD008	AUX	54/AA	594+0	21/35	HSS	A
1CRA-FD009	AUX	53-54/CC-DD	594+0	22/STAIR	DID	C
1CRA-FD010	AUX	53-54/CC	594+0	21/STAIR	HSS	A
1CRA-FD011	AUX	53/AA-BB	594+0	20/35	DID	C
1CRA-FD012	AUX	53/BB-CC	594+0	20/21	HSS	A

(continued)

Table 16.9-5-2

REQUIRED FIRE DAMPERS

DAMPER NUMBER	BLDG	LOCATION	ELEVATION	FIRE AREA INTERFACE	RISK CRITERIA	REMEDIAL ACTION CONDITION
1CRA-FD013	AUX	52/CC-DD	594+0	20/22	LSS	B
1CRA-FD016	AUX	54-55/DD-EE	574+0	22/45	LSS	B
1CRA-FD017	AUX	54-55/DD	574+0	17/45	HSS	A
1CRA-FD018	AUX	54-55/DD	574+0	17/45	HSS	A
1CRA-FD019	AUX	54/AA-BB	574+0	17/45	HSS	A
1CRA-FD020	AUX	57/CC-DD	574+0	16/17	HSS	A
1CRA-FD021	AUX	53-54/DD-EE	574+0	22/45	LSS	B
1CRA-FD022	AUX	55-56/DD	574+0	17/45	HSS	A
1CRA-FD023	AUX	56-57/DD	574+0	17/45	HSS	A
1CRA-FD024A	AUX	57/DD-EE	574+0	45/46	DID	C
1CRA-FD024B	AUX	57/DD-EE	574+0	45/46	DID	C
1CRA-FD025A	AUX	54-55/DD-EE	574+0	22/45	LSS	B
1CRA-FD025B	AUX	54-55/DD-EE	574+0	22/45	LSS	B
1CRA-FD026	AUX	54-55/EE	577+0	18/22	DID	C
1CRA-FD028	AUX	53-54/EE	577+0	18/22	DID	C
1CRA-FD029	AUX	54-55/EE	568+0	11/22	DID	C
1CRA-FD030	AUX	54-55/EE	568+0	11/22	DID	C
1CRA-FD039	AUX	57/EE-FF	577+0	18/18 (KC PUMPS)	DID	C
1CR-FD001	AUX	55-56/DD-EE	594+0	21/22	HSS	A
1CR-FD002	AUX	55-56/DD-EE	594+0	21/22	HSS	A
1CR-FD003	AUX	54/AA-BB	594+0	21/35	HSS	A
1CR-FD004	AUX	53-54/BB	594+0	21/35	HSS	A
1CR-FD005	AUX	53-54/BB	594+0	21/35	HSS	A
1CR-FD007	AUX	51/CC-DD	594+0	13/20	HSS	A
2CRA-FD005A	AUX	59-60/DD-EE	594+0	21/22	HSS	A
2CRA-FD005B	AUX	59-60/DD-EE	594+0	21/22	HSS	A
2CRA-FD008	AUX	60/AA-BB	594+0	19/21	HSS	A
2CRA-FD009	AUX	60-61/CC	594+0	19/22	LSS	B
2CRA-FD012	AUX	61/CC-DD	594+0	19/22	LSS	B
2CRA-FD015	AUX	59-60/DD-EE	574+0	22/46	LSS	B
2CRA-FD016	AUX	59-60/DD	574+0	16/46	HSS	A
2CRA-FD017	AUX	59-60/DD	574+0	16/46	HSS	A
2CRA-FD018	AUX	60/AA-BB	574+0	16/46	HSS	A
2CRA-FD019	AUX	58-59/DD	574+0	16/46	HSS	A
2CRA-FD020	AUX	57-58/DD	574+0	16/46	HSS	A
2CRA-FD021	AUX	60-61/DD-EE	574+0	22/46	LSS	B
2CRA-FD022A	AUX	59-60/DD-EE	574+0	22/46	LSS	B

(continued)

Table 16.9-5-2

REQUIRED FIRE DAMPERS

DAMPER NUMBER	BLDG	LOCATION	ELEVATION	FIRE AREA INTERFACE	RISK CRITERIA	REMEDIAL ACTION CONDITION
2CRA-FD022B	AUX	59-60/DD-EE	574+0	22/46	LSS	B
2CRA-FD023	AUX	59-60/EE	577+0	18/22	DID	C
2CRA-FD025	AUX	60-61/EE	577+0	18/22	DID	C
2CRA-FD026	AUX	59-60/EE	568+0	11/22	DID	C
2CRA-FD027	AUX	59-60/EE	568+0	11/22	DID	C
2CR-FD001	AUX	58-59/DD-EE	594+0	21/22	HSS	A
2CR-FD002	AUX	58-59/DD	594+0	21/22	HSS	A
2CR-FD003	AUX	63-64/CC	594+0	12/19	HSS	A
1VF-FD001A	AUX	51, NN-PP	605+10	22/24	LSS	B
1VF-FD001B	AUX	51, NN-PP	605+10	22/24	LSS	B
1VF-FD002A	AUX	50-51/NN-PP	631+6	24/38	DID	C
1VF-FD002B	AUX	50-51/NN-PP	631+6	24/38	DID	C
1VF-FD004	AUX	49/PP-QQ	631+6	24/38	DID	C
1VF-FD005	AUX	49-50/PP-QQ	631+6	24/38	DID	C
1VF-FD006	AUX	50-51	631+6	24/38	DID	C
1VF-FD007	AUX	50-51/KK-LL	605+10	22/24	LSS	B
1VF-FD010	AUX	50-51/KK	605+10	22/24	LSS	B
1VF-FD011	AUX	50-51/JJ-KK	631+6	22/38	LSS	B
1VF-FD013	AUX	50-51/JJ-KK	616+10	22/24	LSS	B
1VF-FD014	AUX	50-51/JJ-KK	616+10	22/24	LSS	B
2VF-FD001A	AUX	63, NN-PP	605+10	22/23	LSS	B
2VF-FD001B	AUX	63, NN-PP	605+10	22/23	LSS	B
2VF-FD002A	AUX	63-64/NN-PP	631+6	23/47	DID	C
2VF-FD002B	AUX	63-64/NN-PP	631+6	23/47	DID	C
2VF-FD004	AUX	65/PP-QQ	631+6	23/47	DID	C
2VF-FD005	AUX	64-65/PP-QQ	631+6	23/47	DID	C
2VF-FD006	AUX	63-64/PP-QQ	631+6	23/47	DID	C
2VF-FD007	AUX	63-64/KK-LL	605+10	22/23	LSS	B
2VF-FD010	AUX	63-64/KK	605+10	22/23	LSS	B
2VF-FD011	AUX	64-64/JJ-KK	631+6	22/47	LSS	B
2VF-FD013	AUX	63-64/JJ-KK	616+10	22/23	LSS	B
2VF-FD014	AUX	63-64/JJ-KK	616+10	22/23	LSS	B
1TB-FD001	TB1	18-19/V	594+0	TB1/SRV	DID	C
1TB-FD002	TB1	18-19/V	594+0	TB1/SRV	DID	C
1TB-FD003	TB1	18-19/V	594+0	TB1/SRV	DID	C
1TB-FD004	TB1	18-19/V	594+0	TB1/SRV	DID	C
1TB-FD005	TB1	18-19/V	594+0	TB1/SRV	DID	C
1TB-FD006	TB1	18-19/V	594+0	TB1/SRV	DID	C
1TB-FD007	TB1	21-22/V	594+0	TB1/SRV	DID	C
1TB-FD008	TB1	21-22/V	594+0	TB1/SRV	DID	C
1TB-FD009	TB1	21-22/V	594+0	TB1/SRV	DID	C
1TB-FD010	TB1	21-22/V	594+0	TB1/SRV	DID	C
1TB-FD011	TB1	21-22/V	594+0	TB1/SRV	DID	C
1TB-FD012	TB1	21-22/V	594+0	TB1/SRV	DID	C
1TB-FD032	TB1	18-19/V	594+0	TB1/SRV	DID	C

(continued)

Table 16.9-5-2

REQUIRED FIRE DAMPERS

DAMPER NUMBER	BLDG	LOCATION	ELEVATION	FIRE AREA INTERFACE	RISK CRITERIA	REMEDIAL ACTION CONDITION
1TB-FD038	TB1	16-17/V	594+0	TB1/SRV	DID	C
1TB-FD039	TB1	16-17/V	594+0	TB1/SRV	DID	C
1TB-FD040	TB1	16/V	594+0	TB1/SRV	DID	C
1TB-FD043	TB1	30-31/1J-1K	568+0	TB1/OTT	DID	C
1TB-FD044	TB1	32/1J-1K	594+0	TB1/MTOT	DID	C
1TB-FD045	TB1	30/1J-1K	594+0	TB1/MTOT	DID	C
1TB-FD046	TB1	32/1K-1L	568+0	TB1/OTT	DID	C
2TB-FD013	TB2	21-22/P	594+0	TB2/SRV	DID	C
2TB-FD014	TB2	21-22/P	594+0	TB2/SRV	DID	C
2TB-FD015	TB2	21-22/P	594+0	TB2/SRV	DID	C
2TB-FD016	TB2	21-22/P	594+0	TB2/SRV	DID	C
2TB-FD017	TB2	21-22/P	594+0	TB2/SRV	DID	C
2TB-FD018	TB2	21-22/P	594+0	TB2/SRV	DID	C
2TB-FD019	TB2	18-19/P	594+0	TB2/SRV	DID	C
2TB-FD020	TB2	18-19/P	594+0	TB2/SRV	DID	C
2TB-FD021	TB2	18-19/P	594+0	TB2/SRV	DID	C
2TB-FD022	TB2	18-19/P	594+0	TB2/SRV	DID	C
2TB-FD023	TB2	18-19/P	594+0	TB2/SRV	DID	C
2TB-FD024	TB2	18-19/P	594+0	TB2/SRV	DID	C
2TB-FD031	TB2	32/2K-2L	568+0	TB2/OTT	DID	C
2TB-FD032	TB2	18/P	594+0	TB2/SRV	DID	C
2TB-FD036	TB2	16-17/P	594+0	TB2/SRV	DID	C
2TB-FD038	TB2	17-18/P	594+0	TB2/SRV	DID	C
2TB-FD039	TB2	32/2J-2K	594+0	TB2/MTOT	DID	C
2TB-FD040	TB2	30/2J-2K	594+0	TB2/MTOT	DID	C
2TB-FD041	TB2	30-31/2J/2K	568+0	TB2/OTT	DID	C
SB-FD158	SRV	18-19/V	568+0	TB1/SRV	DID	C

Table 16.9-5-3

HIGH SAFETY SIGNIFICANT (HSS) FIRE AREAS*

FIRE AREA	BLDG	ELEVATION	DESCRIPTION
6	AUX	560+0	Unit 1 Electrical Pen Room El 560
12	AUX	577+0	Unit 2 Electrical Pen Room El 577
13	AUX	577+0	Unit 1 Electrical Pen Room El 577
14	AUX	577+0	Unit 2 4160V Essential Swgr Room (2ETA)
15	AUX	577+0	Unit 1 4160V Essential Swgr Room (1ETA)
16	AUX	574+0	Unit 2 Cable Room El 574
17	AUX	574+0	Unit 1 Cable Room El 574
21	AUX	594+0	Main Control Room El 594

*High Safety Significant (HSS) Fire Areas are defined as the areas with HSS fire barrier features in accordance with the Catawba NFPA 805 Monitoring Program.

Table 16.9-5-4

FIRE AREAS AND SHUTDOWN TRAIN / METHOD

FIRE AREA	FIRE AREA DESCRIPTIONS	ASSURED SHUTDOWN TRAIN / METHOD
1	ND & NS Pump Room El 522 (Common)	SSS
2	Unit 2 CA Pump Room El 543	SSS
3	Unit 1 CA Pump Room El 543	SSS
4	Aux Bldg. Gen Area & NV Pump Room El 543 (Common)	SSS
5	Unit 2 Electrical Pen Room El 560	A
6	Unit 1 Electrical Pen Room El 560	A
7	Unit 2 4160V Essential SWGR Room El 560	A
8	Unit 1 4160V Essential SWGR Room El 560	A
9	Unit 2 Battery Room El 554	SSS
10	Unit 1 Battery Room El 554	SSS
11	Aux Bldg. Gen Area & U1 KC Pump Room El 560 (Common)	SSS
12	Unit 2 Electrical Pen Room El 577	B
13	Unit 1 Electrical Pen Room El 577	B
14	Unit 2 4160V Essential SWGR Room El 577	B
15	Unit 1 4160V Essential SWGR Room El 577	B
16	Unit 2 Cable Room El 574	SSS
17	Unit 1 Cable Room El 574	SSS
18	Aux Bldg. Gen Area & U2 KC Pump Room El 577 (Common)	SSS
19	Unit 2 Electrical Pen Room El 594	A
20	Unit 1 Electrical Pen Room El 594	A
21	Control Room El 594 (Common)	SSS
22	Aux Bldg. Gen Area El 594 (Common)	SSS
23	Unit 2 Fuel Storage Area El 605	A
24	Unit 1 Fuel Storage Area El 605	A
25	Diesel Generator Bldg. 1A El 556	B
25A	Diesel Generator Bldg. 1A Stairwell	B
26	Diesel Generator Bldg. 1B El 556	A
26B	Diesel Generator Bldg. 1B Stairwell	A
27	Diesel Generator Bldg. 2A El 556	B
27A	Diesel Generator Bldg. 2A Stairwell	B
28	Diesel Generator Bldg. 2B El 556	A
28B	Diesel Generator Bldg. 2B Stairwell	A
29	Train A RN Pump Structure El 600 (Common)	B
30	Train B RN Pump Structure El 600 (Common)	A
31	Unit 2 Train A Aux Shutdown Panel El 543	B
32	Unit 1 Train A Aux Shutdown Panel El 543	B
33	Unit 2 Train B Aux Shutdown Panel El 543	A
34	Unit 1 Train B Aux Shutdown Panel El 543	A
35	Control Room Tagout Area El 594	A or B
36	Unit 2 Turbine Driven CA Pump Control Panel Room El 543	B
37	Unit 1 Turbine Driven CA Pump Control Panel Room El 543	B
38	Unit 1 Fuel Storage Area HVAC Room El 631	A or B

(continued)

Table 16.9-5-4

FIRE AREAS AND SHUTDOWN TRAIN / METHOD

FIRE AREA	FIRE AREA DESCRIPTIONS	ASSURED SHUTDOWN TRAIN / METHOD
39	Unit 2 Turbine Driven CA Pump Pit El 543	B
40	Unit 1 Turbine Driven CA Pump Pit El 543	B
41	DG1A Sequencer Tunnel El 556	B
42	DG1B Sequencer Tunnel El 556	A
43	DG2A Sequencer Tunnel El 556	B
44	DG2B Sequencer Tunnel El 556	A
45	Unit 1 Cable Room Corridor El 574	B
46	Unit 2 Cable Room Corridor El 574	B
47	Unit 2 Fuel Storage Area HVAC Room El 631	A or B
48	Unit 2 Interior Doghouse	A and B
49	Unit 1 Interior Doghouse	A and B
50	Unit 2 Exterior Doghouse	A and B
51	Unit 1 Exterior Doghouse	A and B
ASB	Auxiliary Service Building	A or B
RB1	Unit 1 Reactor Building	A and B
RB2	Unit 2 Reactor Building	A and B
SRV	Service Building	B
SSF	Standby Shutdown Facility	A or B
STAIR*	Stairway	See Note
TB1	Unit 1 Turbine Building	A or B
TB2	Unit 2 Turbine Building	A or B
YRD**	Yard Area	A or B

* IF the barrier in a stairway is adjacent to a HSS Fire Area (see Table 16.9-5-3), enter CONDITION A; otherwise enter CONDITION C.
** Exterior walls that interface with the YRD do not require entry into a CONDITION statement and therefore do not have a REQUIRED ACTION.

A = A TRAIN
B = B TRAIN
SSS = STANDBY SHUTDOWN SYSTEM

BASES

The functional integrity of the Fire Rated Assemblies and associated sealing devices ensures that fires will be confined or adequately retarded so as not to spread between fire areas/compartments.

The fire barriers and associated penetration seals are passive elements in the facility fire protection program and are subject to periodic inspections.

Risk-informed insights from the Fire PRA process can apply to compensatory actions. The safety significance of the fire area can provide relief for required compensatory actions. In addition, the presence of functional fire detection can reduce the required compensatory actions. Functional fire detection in the area provides early warning of a fire for fire brigade response. Fire detection can provide a compensatory action equivalent to or better than fire watch.

Fire barrier penetration seals, including cable/pipe penetration seals, fire doors, and fire dampers, are considered FUNCTIONAL when the visually observed condition indicates no abnormal change in appearance or abnormal degradation. An evaluation is performed to determine the cause of any identified fire barrier penetration seal abnormal change in appearance or abnormal degradation and the effect of this change on the ability of the fire barrier penetration seal to perform its function. Based on this evaluation additional inspections may be performed.

During periods of time when a barrier is not FUNCTIONAL, either:

(1) Perform the recommended fire watch in accordance with the criteria in the remedial actions, or

(2) a licensee may choose to implement a different required action or a combination of actions (e.g., additional administrative controls, operator briefings, temporary procedures, interim shutdown strategies, operator manual actions, temporary fire barriers, temporary detection or suppression systems). Such a change must be made to the approved Fire Protection Plan (FPP). However, the licensee must complete a documented evaluation of the impact of the proposed required action to the FPP. The evaluation must demonstrate that the required actions would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. Any change to the FPP must maintain compliance with the General Design Criteria and 10 CFR 50.48(a).

The evaluation of the required action should incorporate risk insights regarding the location, quantity, and type of combustible material in the fire area; the presence of ignition sources and their likelihood of occurrence; the automatic fire suppression and the fire detection capability in the fire area; the manual fire suppression capability in the fire area; and the human error probability where applicable.

BASES (continued)

The expectation is to promptly complete the corrective action at the first available opportunity and eliminate the reliance on the required action.

This SLC is part of the Catawba Fire Protection Program and therefore subject to the provisions of Section 2.C.(5) of the Catawba Renewed Facility Operating Licenses.

REFERENCES

1. Catawba UFSAR, Section 9.5.1.
2. Catawba Nuclear Station 10 CFR 50.48(c) Fire Protection Safety Evaluation (SE).
3. Catawba Plant Design Basis Specification for Fire Protection, CNS-1465.00-00-0006, as revised.
4. Catawba UFSAR, Section 18.2.8.
5. Catawba License Renewal Commitments, CNS-1274.00-00-0016, Section 4.12.2.
6. NRC Regulatory Issue Summary 2005-07, Compensatory Measures to Satisfy the Fire Protection Program Requirements, April 19, 2005.
7. Catawba Renewed Facility Operating License Conditions 2.C.(5).
8. CNC-1435.00-00-0084, Catawba NFPA 805 Monitoring Program.
9. CNC-1435.00-00-0044, Fire Protection Nuclear Safety Capability Assessment.
10. CN-1209.10 series drawings.
11. CN-1105 series drawings.

16.9 AUXILIARY SYSTEMS

16.9-5 Fire Rated Assemblies

COMMITMENT All required Fire Rated Assemblies (walls, floors/ceilings, cable enclosures and other fire barriers) and all sealing devices in fire rated assembly penetrations (fire doors, fire dampers, and penetration seals) as shown on the CN-1105 drawing series shall be FUNCTIONAL.

APPLICABILITY: At all times.

--NOTE--
Non-functional or breached fire barrier features (walls, floors, ceilings, doors, dampers, and penetration seals) in the diesel generator rooms and the auxiliary feedwater pump rooms may affect CO_2 System FUNCTIONALITY. See SLC 16.9-3, "CO_2 Systems".

REMEDIAL ACTIONS

IF the required Fire Rated Assembly sealing device is a Fire Door, see Table 16.9-5-1

IF the required Fire Rated Assembly sealing device is a Fire Damper see Table 16.9-5-2

IF required Fire Rated Assembly is a Fire Barrier or Penetration Seal:
1. Identify the location of the impaired fire protection feature by elevation, column, and building
2. Verify the wall, floor/ceiling is a committed boundary on the CN-1105 drawing series (if not a committed boundary, SLC 16.9-5 does not apply)
3. Refer to CN-1209-10 series drawings to identify the Fire Area on <u>both</u> sides of the impaired feature
4. IF either of the Fire Areas is identified as High Safety Significant (HSS) (see Table 16.9-5-3) then implement the REQUIRED ACTION **CONDITION A**
5. IF the Fire Areas are not HSS, then identify the associated shutdown trains/methods of the Fire Areas on each side of the barrier using Table 16.9-5-4 and implement the REQUIRED ACTION as identified in the following Chart:

Shutdown Train (Side 1 & Side 2)	*A*	*B*	*SSS*	*A or B*	*A and B*
A	**CONDITION C**	**CONDITION B**	**CONDITION B**	**CONDITION C**	**CONDITION B**
B	**CONDITION B**	**CONDITION C**	**CONDITION B**	**CONDITION C**	**CONDITION B**
SSS	**CONDITION B**	**CONDITION B**	**CONDITION C**	**CONDITION B**	**CONDITION B**
A or B	**CONDITION C**	**CONDITION C**	**CONDITION B**	**CONDITION C**	**CONDITION B**
A and B	**CONDITION B**	**CONDITION B**	**CONDITION B**	**CONDITION B**	**CONDITION C**

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more HSS* required Fire Rated Assemblies is non-functional.	A.1 Establish a continuous fire watch on at least one side of the assembly.	1 hour
	<u>OR</u>	
	A.2.1 Verify at least one side of the assembly has FUNCTIONAL fire detection instrumentation.	1 hour
	<u>AND</u>	
	A.2.2 Establish an hourly fire watch patrol on at least one side of the assembly.	1 hour
	<u>OR</u>	
	A.3 Complete an evaluation as permitted by NRC RIS 2005-07 to institute required action(s).	Prior to terminating Required Action A.1 or A.2

(continued)

REMEDIAL ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more LSS** required Fire Rated Assemblies is non-functional.	B.1 Establish an hourly fire watch on at least one side of the assembly.	1 hour
	<u>OR</u>	
	B.2.1 Verify at least one side of the assembly has FUNCTIONAL fire detection instrumentation.	1 hour
	<u>AND</u>	
	B.2.2 Establish a once per shift fire watch patrol on at least one side of the assembly.	1 hour
	<u>OR</u>	
	B.3 Complete an evaluation as permitted by NRC RIS 2005-07 to institute required action(s).	Prior to terminating Required Action B.1 or B.2

(continued)

REMEDIAL ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more DID*** required Fire Rated Assemblies is non-functional.	C.1 Establish a once per shift fire watch on at least one side of the assembly.	1 hour
	OR	
	C.2 Verify at least one side of the assembly has FUNCTIONAL fire detection instrumentation.	1 hour
	OR	
	C.3 Complete an evaluation as permitted by NRC RIS 2005-07 to institute required action(s).	Prior to terminating Required Action C.1

*High Safety Significant (HSS) Fire Areas containing required Fire Rated Assemblies are defined in Table 16.9-5-3.

**Low Safety Significant (LSS) Fire Areas containing required Fire Rated Assemblies are defined as those areas with a boundary between redundant shutdown trains.

***Defense-in-Depth (DID) Fire Areas containing required Fire Rated Assemblies are defined as analysis compartment boundaries or PRA compartment boundaries that do not meet the HSS or LSS definitions.

TESTING REQUIREMENTS

TEST		FREQUENCY
TR 16.9-5-1	Verify each HSS and LSS interior unlocked fire door is closed.	24 hours
TR 16.9-5-2	Verify each HSS and LSS locked closed fire door is closed.	7 days
TR 16.9-5-3	Perform an inspection and functional test of the release and closing mechanism and latches for each swinging fire door shown in Table 16.9-5-1.	6 months
TR 16.9-5-4	Perform a visual inspection of the exposed surfaces of each required Fire Rated Assembly.	18 months
TR 16.9-5-5	------------------------------NOTE------------------------------ Any abnormal changes or degradation shall be identified and resolved via the corrective action program. Based on the investigation results, additional dampers may be selected for inspection. Samples will be grouped by unit, system, and train and shall be selected such that each damper is inspected every 15 years. --- Perform a visual inspection of fire dampers in each HSS and LSS required Fire Rated Assembly.	18 months, in accordance with the predefined inspection schedule

(continued)

TESTING REQUIREMENTS (continued)

	TEST	FREQUENCY
TR 16.9-5-6	---------------------------NOTE--------------------------- Any abnormal changes or degradation shall be identified and resolved via the corrective action program. Based on the investigation results, additional Fire Rated Assemblies may be selected for inspection. Samples shall be selected such that each Fire Rated Assembly is inspected every 15 years. --- Perform a visual inspection of penetration seals in each HSS AND LSS required Fire Rated Assembly.	18 months, in accordance with the predefined inspection schedule
TR 16.9-5-7	Perform an inspection and functional test of the automatic hold open, release and closing mechanism for each rolling fire door shown in Table 16.9-5-1.	18 Months

Table 16.9-5-1

Required Fire Doors

DOOR NUMBER	BLDG	LOCATION	ELEVATION	FIRE AREA INTERFACE	RISK CRITERIA	REMEDIAL ACTION CONDITION
AX500F	AUX	56, FF	522+0	1/4	DID	C
AX214A	AUX	54-55, FF-GG	543+0	1/4	DID	C
AX214B	AUX	58-59, FF-GG	543+0	1/4	DID	C
AX217D	AUX	52-53, BB	543+0	3/34	LSS	B
AX217F(1)	AUX	51, AA-BB	543+0	3/40	LSS	B
AX217G	AUX	52-53, BB	543+0	3/32	LSS	B
AX227D	AUX	54-55, MM-NN	543+0	4/22	DID	C
AX227E	AUX	59-60, MM-NN	543+0	4/22	DID	C
AX228A	AUX	56-57, EE	543+0	4/9	DID	C
AX228B	AUX	57-58, EE	543+0	4/10	DID	C
AX248	AUX	57-58, QQ	543+0	4/ASB	LSS	B
AX260B	AUX	61-62, BB-CC	543+0	2/36	LSS	B
AX260F(1)	AUX	62, AA-BB	543+0	2/39	LSS	B
AX260G	AUX	61-62, BB-CC	543+0	2/31	LSS	B
AX260H	AUX	61-62, BB-CC	543+0	2/33	LSS	B
T527#1	AUX	52-53, BB-CC	543+0	3/37	LSS	B
AX202	AUX	51, NN	543+0	4/STAIR	DID	C
AX253A	AUX	63, NN	543+0	4/STAIR	DID	C
AX227A	AUX	59, FF-GG	543+0	4/STAIR	DID	C
AX260E	AUX	52, CC	543+0	3/STAIR	DID	C
AX516M	AUX	62, CC	543+0	2/STAIR	DID	C
AX354A	AUX	55, DD-EE	554+0	22/45	LSS	B
AX354B	AUX	59, DD-EE	554+0	22/46	LSS	B
AX418	AUX	57, BB	554+0	9/10	DID	C
AX419	AUX	57, DD-EE	554+0	9/10	DID	C
AX420A	AUX	59, DD-EE	554+0	9/46	LSS	B
AX421A	AUX	55, DD-EE	554+0	10/45	LSS	B
S102A	AUX	53-54, AA	554+0	10/SRV	LSS	B
AX302	AUX	41, CC-DD	556+0	25/41	DID	C
AX304	AUX	41, AA-BB	556+0	26/42	DID	C
AX306	AUX	73, DD-EE	556+0	27/43	DID	C
AX308	AUX	73, BB-CC	556+0	28/44	DID	C
AX348B	AUX	54-55, MM-NN	560+0	11/22	DID	C
AX348C	AUX	53-54, HH	560+0	4/11	DID	C
AX348D	AUX	59-60, MM-NN	560+0	11/22	DID	C
AX348E	AUX	60-61, HH	560+0	4/11	DID	C
AX352B	AUX	53, CC-DD	560+0	6/STAIR	HSS	A
AX352C	AUX	53, CC-DD	560+0	10/STAIR	DID	C
AX352D	AUX	46-47, BB-CC	560+0	6/RB1	HSS	A
AX353	AUX	45, BB	560+0	6/8	HSS	A
AX353B	AUX	45, AA-BB	560+0	8/41	LSS	B
AX353C	AUX	45, AA-BB	560+0	8/42	DID	C

(continued)

Table 16.9-5-1

Required Fire Doors

DOOR NUMBER	BLDG	LOCATION	ELEVATION	FIRE AREA INTERFACE	RISK CRITERIA	REMEDIAL ACTION CONDITION
AX393B	AUX	61, CC-DD	560+0	9/STAIR	DID	C
AX393C	AUX	61, CC-DD	560+0	5/STAIR	DID	C
AX393D	AUX	67-68, BB-CC	560+0	5/RB2	LSS	B
AX394	AUX	69, BB	560+0	5/7	DID	C
AX394B	AUX	69, AA-BB	560+0	7/43	LSS	B
AX394C	AUX	69, AA-BB	560+0	7/44	DID	C
AX395	AUX	61, AA-BB	560+0	7/9	LSS	B
AX396	AUX	53, AA-BB	560+0	8/10	LSS	B
AX415	AUX	45-46, CC-DD	560+0	6/RB1	HSS	A
AX416	AUX	68-69, CC-DD	560+0	5/RB2	LSS	B
AX417	AUX	57, QQ	560+0	11/ASB	LSS	B
AX313D	AUX	51, NN	560+0	11/STAIR	DID	C
AX388B	AUX	63, NN	560+0	11/STAIR	DID	C
AX348	AUX	59, FF-GG	560+0	11/STAIR	DID	C
AX355A	AUX	53-54, FF	568+0	4/11	DID	C
AX355D	AUX	60, FF	568+0	4/11	DID	C
AX355E	AUX	60, FF	568+0	11/STAIR	DID	C
AX515	AUX	54, BB	574+0	17/45	HSS	A
AX516	AUX	56-57, DD	574+0	14/45	HSS	A
AX516A	AUX	57-58, DD	574+0	16/46	HSS	A
AX516K	AUX	57, AA-BB	574+0	16/17	HSS	A
AX517A	AUX	53-54, DD-EE	574+0	22/45	LSS	B
AX517B	AUX	60-61, DD-EE	574+0	22/46	LSS	B
AX517C	AUX	57, DD-EE	574+0	45/46	DID	C
AX517D	AUX	57, DD-EE	574+0	9/46	LSS	B
AX517E	AUX	56-57, DD-EE	574+0	10/46	LSS	B
AX518	AUX	60, BB	574+0	16/46	HSS	A
S303	SRV	36-37, 1N	574+0	45/SRV	DID	C
S303C	SRV	36-37, V	574+0	45/SRV	DID	C
S304A	AUX	60, AA	574+0	46/SRV	DID	C
AX500H	AUX	54-55, MM-NN	577+0	18/22	DID	C
AX500K	AUX	53-54, HH-GG	577+0	4/18	DID	C
AX500L	AUX	59-60, MM-NN	577+0	18/22	DID	C
AX500N	AUX	60-61, HH-GG	577+0	4/18	DID	C
AX513B	AUX	53, CC-DD	577+0	13/STAIR	HSS	A
AX514	AUX	45, BB	577+0	13/15	HSS	A
AX514B	AUX	45-46, AA-BB	577+0	6/13	HSS	A
AX517	AUX	57, EE	577+0	9/18	DID	C
AX525	AUX	55-56, QQ	577+0	18/ASB	LSS	B
AX525B	AUX	56, QQ	577+0	18/ASB	LSS	B
AX526D	AUX	58, QQ	577+0	18/ASB	LSS	B
A314#3	AUX	61, CC-DD	577+0	12/STAIR	HSS	A
AX533C	AUX	61, CC-DD	577+0	46/STAIR	DID	C
AX534	AUX	69, BB	577+0	12/14	HSS	A

(continued)

Table 16.9-5-1

Required Fire Doors

DOOR NUMBER	BLDG	LOCATION	ELEVATION	FIRE AREA INTERFACE	RISK CRITERIA	REMEDIAL ACTION CONDITION
AX534B	AUX	68-69, AA-BB	577+0	7/14	HSS	A
AX535A	AUX	61, AA-BB	577+0	14/46	HSS	A
AX536	AUX	53, AA-BB	577+0	15/45	HSS	A
AX656	AUX	53, CC-DD	577+0	45/STAIR	DID	C
AX500P	AUX	51, NN	577+0	18/STAIR	DID	C
AX500S	AUX	63, NN	577+0	18/STAIR	DID	C
AX338A	AUX	60, FF-GG	577+0	18/STAIR	DID	C
AX602	AUX	52, UU-VV	594+0	24/ASB	DID	C
AX627	AUX	62, UU-VV	594+0	23/ASB	DID	C
AX630	AUX	58, QQ	594+0	22/ASB	LSS	B
AX632	AUX	57, QQ	594+0	22/ASB	LSS	B
AX635	AUX	60-61, QQ	594+0	22/ASB	LSS	B
AX635E	AUX	53-54, QQ	594+0	22/ASB	LSS	B
AX635F	AUX	53-54, QQ	594+0	22/ASB	LSS	B
AX655	AUX	62-63, DD	594+0	19/48	LSS	B
AX656C	AUX	61, CC-DD	594+0	19/22	LSS	B
AX657	AUX	60-61, CC	594+0	19/22	LSS	B
AX657A(2)	AUX	54, BB	594+0	21/35	HSS	A
AX657B	AUX	52-53, CC-DD	594+0	20/22	LSS	B
AX657E(2)	AUX	53, BB	594+0	21/35	HSS	A
AX657F	AUX	60, DD-EE	594+0	21/22	HSS	A
AX657G	AUX	57-58, DD-EE	594+0	21/22	HSS	A
AX657H	AUX	54, DD-EE	594+0	21/22	HSS	A
AX657J	AUX	53, BB-CC	594+0	20/21	HSS	A
AX658B	AUX	51-52, DD	594+0	20/49	LSS	B
S400	AUX	55-56, AA	594+0	21/SRV	HSS	A
S406	AUX	58-59, AA	594+0	21/SRV	HSS	A
AX635G	AUX	51, NN	594+0	22/STAIR	DID	C
AX635H	AUX	63, NN	594+0	22/STAIR	DID	C
AX654A	AUX	60, FF	594+0	22/STAIR	DID	C
AX654B	AUX	61, CC-DD	594+0	19/STAIR	DID	C
AX665B	AUX	53, CC-DD	594+0	22/STAIR	DID	C
AX700B	AUX	50-51, JJ-KK	605+10	24/RB1	LSS	B
AX700D	AUX	63-64, KK	605+10	22/23	LSS	B
AX701	AUX	50-51, JJ-KK	605+10	22/RB1	LSS	B
AX714B	AUX	63-64, JJ-KK	605+10	23/RB2	LSS	B
AX720	AUX	50-51, HH-JJ	605+10	22/RB1	LSS	B
AX721	AUX	63-64, HH-JJ	605+10	22/RB2	LSS	B
AX714C	AUX	50-51, KK	605+10	22/24	LSS	B
AX715A	AUX	63-64, JJ-KK	605+10	22/RB2	LSS	B
S211(2)	TB1	17, V	568+0	SRV/TB1	DID	C
S212	TB1	19, V	568+0	SRV/TB1	DID	C
S210	TB1	21,V	568+0	SRV/TB1	DID	C

(continued)

Table 16.9-5-1

Required Fire Doors

DOOR NUMBER	BLDG	LOCATION	ELEVATION	FIRE AREA INTERFACE	RISK CRITERIA	REMEDIAL ACTION CONDITION
S206	TB1	22, V	568+0	SRV/TB1	DID	C
S201	TB1	33, V	568+0	SRV/TB1	DID	C
SR3[(3)]	TB1	30-31, V	568+0	SRV/TB1	DID	C
S201A	TB1	27, V	568+0	SRV/TB1	DID	C
T101	TB1	31, 1K	568+0	TB1/U1 OTT	DID	C
S424	TB1	24-25, V	594+0	SRV/TB1	DID	C
S425	TB1	23, V	594+0	SRV/TB1	DID	C
S426	TB1	22, V	594+0	SRV/TB1	DID	C
SR21[(3)]	TB1	24, V	594+0	SRV/TB1	DID	C
S472	TB1	27, V	594+0	SRV/TB1	DID	C
S423	TB1	29, V	594+0	SRV/TB1	DID	C
S422	TB1	29, V	594+0	SRV/TB1	DID	C
SR7[(3)]	TB1	29-30, V	594+0	SRV/TB1	DID	C
S416	TB1	32, V	594+0	SRV/TB1	DID	C
S444	TB1	15, V	594+0	SRV/TB1	DID	C
TR4[(3)]	TB1	15-16, V	594+0	SRV/TB1	DID	C
T200A	TB1	32, 1J-1K	594+0	TB1/U1 MTOT	DID	C
S701	TB1	22, 1L	619+6	SRV/TB1	DID	C
S704	TB1	33, 1L	619+6	SRV/TB1	DID	C
S209	TB2	20, P	568+0	SRV/TB2	DID	C
S208	TB2	22, P	568+0	SRV/TB2	DID	C
SR2[(3)]	TB2	32-33, P	568+0	SRV/TB2	DID	C
S462	TB2	32, P	568+0	SRV/TB2	DID	C
SR4[(3)]	TB2	30-31, P	568+0	SRV/TB2	DID	C
S1102	TB2	27, P	568+0	SRV/TB2	DID	C
T151	TB2	31, 2K	568+0	TB2/U2 OTT	DID	C
S423E	TB2	26, P-Q	594+0	SRV/TB2	DID	C
S416A	TB2	32, P	594+0	SRV/TB2	DID	C
SR8[(3)]	TB2	29-30, P	594+0	SRV/TB2	DID	C
S422A	TB2	29, P	594+0	SRV/TB2	DID	C
S423A	TB2	29, P	594+0	SRV/TB2	DID	C
S435	TB2	24-25, P	594+0	SRV/TB2	DID	C
S436	TB2	23, P	594+0	SRV/TB2	DID	C
S437	TB2	22, P	594+0	SRV/TB2	DID	C
SR22[(3)]	TB2	24, P	594+0	SRV/TB2	DID	C
S444A	TB2	15, P	594+0	SRV/TB2	DID	C
SR16[(3)]	TB2	15-16, P	594+0	SRV/TB2	DID	C
S472A	TB2	27, P	594+0	SRV/TB2	DID	C
T250A	TB2	32, 2J-2K	594+0	TB2/U2 MTOT	DID	C
S701A	TB2	22, 2L	619+6	SRV/TB2	DID	C
S704A	TB2	33, 2L	619+6	SRV/TB2	DID	C
AX662A	NSWPS	----	600+0	29/30	LSS	B

(1) These doors are not equipped with closing mechanisms or latches and are therefore exempt from TESTING REQUIREMENT 16.9-5-3.
(2) These doors are held open with a fusible link.
(3) Rolling Door.

Table 16.9-5-2

Required Fire Dampers

DAMPER NUMBER	BLDG	LOCATION	ELEVATION	FIRE AREA INTERFACE	RISK CRITERIA	REMEDIAL ACTION CONDITION
1VA-FD001	AUX	53/GG-FF	522+0	1/4	DID	C
1VA-FD002	AUX	53/GG-HH	522+0	1/4	DID	C
1VA-FD003	AUX	55-56/GG-HH	522+0	1/4	DID	C
1VA-FD004	AUX	55-56/GG-HH	522+0	1/4	DID	C
1VA-FD005	AUX	54-55/GG-HH	522+0	1/4	DID	C
1VA-FD006	AUX	54-55/GG-HH	522+0	1/4	DID	C
1VA-FD007	AUX	53/GG-FF	522+0	1/4	DID	C
1VA-FD008	AUX	53/GG-FF	522+0	1/4	DID	C
1VA-FD009	AUX	53-54/FF-GG	522+0	1/1 (ND PUMPS)	DID	C
1VA-FD010	AUX	56-57/ GG-HH,	522+0	1/4	DID	C
1VA-FD011	AUX	56-57/FF	522+0	1/4	DID	C
1VA-FD012	AUX	51/NN-PP	543+0	11/STAIR	DID	C
1VA-FD013	AUX	54/MM	543+0	4/22	DID	C
1VA-FD014	AUX	54/MM	543+0	4/22	DID	C
1VA-FD015	AUX	54-55/MM-NN	543+0	4/22	DID	C
1VA-FD016	AUX	54-55/MM-NN	543+0	4/22	DID	C
1VA-FD017	AUX	54-55/MM-NN	543+0	4/22	DID	C
1VA-FD020	AUX	55/JJ-KK	543+0	4/4 (NV PUMPS)	DID	C
1VA-FD033	AUX	51-52/AA-BB	543+0	3/40	LSS	B
1VA-FD034	AUX	51-52/AA-BB	543+0	3/40	LSS	B
1VA-FD035	AUX	52/AA-BB	543+0	3/32	LSS	B
1VA-FD036	AUX	52-53/BB	543+0	3/32	LSS	B
1VA-FD038	AUX	52-53/BB	543+0	3/34	LSS	B
1VA-FD039	AUX	52-53/BB	543+0	3/34	LSS	B
1VA-FD040	AUX	52-53/BB	543+0	3/32	LSS	B
1VA-FD041	AUX	52-53/BB	543+0	3/32	LSS	B
1VA-FD042	AUX	53/CC	543+0	3/STAIR	DID	C
1VA-FD043	AUX	53/CC-DD	543+0	3/STAIR	DID	C
1VA-FD045	AUX	52-53/DD	560+0	3/6	HSS	A
1VA-FD046	AUX	52-53/CC-DD	577+0	6/13	HSS	A
1VA-FD047	AUX	52-53/CC-DD	577+0	6/13	HSS	A
1VA-FD048	AUX	54/MM-NN	560+0	11/22	DID	C
1VA-FD049	AUX	54/MM	560+0	11/22	DID	C
1VA-FD050	AUX	54-55/MM	560+0	4/22	DID	C
1VA-FD051	AUX	54-55/MM	560+0	4/22	DID	C
1VA-FD052	AUX	55/MM-NN	560+0	11/22	DID	C
1VA-FD053	AUX	55/MM	560+0	11/22	DID	C
1VA-FD054	AUX	53/GG-HH	560+0	4/11	DID	C
1VA-FD055	AUX	53/GG-HH	560+0	4/11	DID	C
1VA-FD056	AUX	53/KK	560+0	4/11	DID	C
1VA-FD057	AUX	53/GG-HH	560+0	4/11	DID	C

(continued)

Table 16.9-5-2

REQUIRED FIRE DAMPERS

DAMPER NUMBER	BLDG	LOCATION	ELEVATION	FIRE AREA INTERFACE	RISK CRITERIA	REMEDIAL ACTION CONDITION
1VA-FD058	AUX	53-54/HH	560+0	4/11	DID	C
1VA-FD059	AUX	54/GG-HH	560+0	4/11	DID	C
1VA-FD060	AUX	54/HH	560+0	4/11	DID	C
1VA-FD061	AUX	56-57/QQ	577+0	18/ASB	LSS	B
1VA-FD062	AUX	55-56/QQ	577+0	18/ASB	LSS	B
1VA-FD063	AUX	55/MM-NN	577+0	18/22	DID	C
1VA-FD064	AUX	55/MM	577+0	18/22	DID	C
1VA-FD065	AUX	54/MM	577+0	18/22	DID	C
1VA-FD066	AUX	54/MM	577+0	18/22	DID	C
1VA-FD067	AUX	54/HH	577+0	4/18	DID	C
1VA-FD068	AUX	53-54/HH	577+0	4/18	DID	C
1VA-FD069	AUX	54/GG-HH	577+0	4/18	DID	C
1VA-FD070	AUX	53-54/HH	577+0	4/18	DID	C
1VA-FD071	AUX	53-54/HH	577+0	4/18	DID	C
1VA-FD072	AUX	53/HH	577+0	4/18	DID	C
1VA-FD073	AUX	53/HH	577+0	4/18	DID	C
1VA-FD074	AUX	53/GG-HH	577+0	4/18	DID	C
1VA-FD075	AUX	53-54/KK-LL	594+0	18/22	DID	C
1VA-FD076	AUX	53-54/KK-LL	594+0	18/22	DID	C
1VA-FD078	AUX	57/NN	594+0	22/STAIR	DID	C
1VA-FD087	AUX	55-56/QQ	594+0	22/ASB	LSS	B
1VA-FD088	AUX	53-54/QQ	594+0	22/ASB	LSS	B
1VA-FD133	AUX	53/CC-DD	594+0	22/STAIR	DID	C
1VA-FD139	AUX	51-52/DD	543+0	3/4	DID	C
1VA-FD140	AUX	53-54/FF-GG	560+0	4/11	DID	C
1VA-FD141	AUX	53-54/FF-GG	560+0	4/11	DID	C
1VA-FD142	AUX	53/GG	560+0	4/11	DID	C
1VA-FD143	AUX	53/JJ-HH	560+0	4/11	DID	C
1VA-FD144	AUX	53/KK	560+0	4/11	DID	C
1VA-FD145	AUX	51/KK	560+0	11/18	DID	C
1VA-FD146	AUX	51/KK	560+0	11/18	DID	C
1VA-FD147	AUX	52/MM	560+0	11/18	DID	C
1VA-FD148	AUX	52/MM-NN	560+0	4/11	DID	C
1VA-FD149	AUX	52-53/DD	560+0	3/6	HSS	A
1VA-FD150	AUX	52-53/DD	560+0	3/6	HSS	A
1VA-FD152	AUX	52-53/BB-CC	543+0	3/37	LSS	B
1VA-FD153	AUX	52-53/CC	543+0	3/37	LSS	B
1VA-FD154	AUX	53-54/GG-HH	594+0	4/22	DID	C
1VA-FD155	AUX	53-54/GG-HH	594+0	4/22	DID	C
1VA-FD159	AUX	49-50/AA-BB	543+0	CO2	HSS	A
1VA-FD160	AUX	50-51/AA-BB	543+0	CO2	HSS	A
1VA-FD163	AUX	56/EE	543+0	10/45	LSS	B
1VA-FD164	AUX	56-57/EE	543+0	4/10	DID	C
2VA-FD001	AUX	61/GG-FF	522+0	1/4	DID	C
2VA-FD002	AUX	61/GG-FF	522+0	1/4	DID	C

(continued)

Table 16.9-5-2

REQUIRED FIRE DAMPERS

DAMPER NUMBER	BLDG	LOCATION	ELEVATION	FIRE AREA INTERFACE	RISK CRITERIA	REMEDIAL ACTION CONDITION
2VA-FD003	AUX	60-61/FF-GG	522+0	1/1 (ND PUMPS)	DID	C
2VA-FD004	AUX	61/GG-FF	522+0	1/4	DID	C
2VA-FD005	AUX	60-61/GG-HH	522+0	1/4	DID	C
2VA-FD006	AUX	59-60/GG-HH	522+0	1/4	DID	C
2VA-FD007	AUX	59-60/GG-HH	522+0	1/4	DID	C
2VA-FD008	AUX	58-59/GG-HH	522+0	1/4	DID	C
2VA-FD009	AUX	58-59/GG-HH	522+0	1/4	DID	C
2VA-FD010	AUX	57-58/GG-HH	522+0	1/4	DID	C
2VA-FD011	AUX	57-58/FF	522+0	1/4	DID	C
2VA-FD012	AUX	59-60/MM-NN	543+0	4/22	DID	C
2VA-FD013	AUX	59/MM	543+0	4/22	DID	C
2VA-FD014	AUX	59/MM	543+0	4/22	DID	C
2VA-FD015	AUX	59-60/MM-NN	543+0	4/22	DID	C
2VA-FD020	AUX	63/NN	534+0	4/STAIR	DID	C
2VA-FD023	AUX	59/JJ-KK	543+0	4/4 (NV PUMPS)	DID	C
2VA-FD036	AUX	61-62/DD	560+0	2/5	LSS	B
2VA-FD037	AUX	61-62/CC-DD	577+0	5/12	HSS	A
2VA-FD038	AUX	61-62/CC-DD	577+0	5/12	HSS	A
2VA-FD040	AUX	62-63/AA-BB	543+0	2/39	LSS	B
2VA-FD041	AUX	62-63/AA-BB	543+0	2/39	LSS	B
2VA-FD042	AUX	62/AA-BB	543+0	2/31	LSS	B
2VA-FD043	AUX	61-62/BB	543+0	2/31	LSS	B
2VA-FD045	AUX	61/CC	543+0	2/STAIR	DID	C
2VA-FD046	AUX	61/CC-DD	543+0	2/STAIR	DID	C
2VA-FD048	AUX	61-62/BB	543+0	2/33	LSS	B
2VA-FD049	AUX	61-62/BB	543+0	2/33	LSS	B
2VA-FD050	AUX	61-62/BB	543+0	2/31	LSS	B
2VA-FD051	AUX	61-62/BB	543+0	2/31	LSS	B
2VA-FD053	AUX	60/MM	560+0	11/22	DID	C
2VA-FD054	AUX	59/MM-NN	560+0	11/22	DID	C
2VA-FD056	AUX	60/MM-NN	560+0	11/22	DID	C
2VA-FD057	AUX	59-60/MM	560+0	11/22	DID	C
2VA-FD058	AUX	59-60/MM	560+0	4/22	DID	C
2VA-FD059	AUX	60-61/HH	560+0	4/11	DID	C
2VA-FD060	AUX	61/HH-JJ	560+0	4/11	DID	C
2VA-FD061	AUX	60-61/GG-HH	560+0	4/11	DID	C
2VA-FD062	AUX	61/GG-HH	560+0	4/11	DID	C
2VA-FD063	AUX	61/GG-HH	560+0	4/11	DID	C
2VA-FD064	AUX	60-61/GG-HH	560+0	4/11	DID	C
2VA-FD065	AUX	61/HH	560+0	4/11	DID	C
2VA-FD069	AUX	58-59/QQ	577+0	18/ASB	LSS	B
2VA-FD070	AUX	59-60/QQ	577+0	18/ASB	LSS	B
2VA-FD071	AUX	59-60/MM-NN	577+0	18/22	DID	C
2VA-FD072	AUX	59-60/MM	577+0	18/22	DID	C

(continued)

Table 16.9-5-2

REQUIRED FIRE DAMPERS

DAMPER NUMBER	BLDG	LOCATION	ELEVATION	FIRE AREA INTERFACE	RISK CRITERIA	REMEDIAL ACTION CONDITION
2VA-FD073	AUX	60/MM	577+0	18/22	DID	C
2VA-FD074	AUX	60/MM	577+0	18/22	DID	C
2VA-FD075	AUX	60/HH	577+0	4/22	DID	C
2VA-FD076	AUX	60/HH	577+0	4/22	DID	C
2VA-FD077	AUX	60-61/HH	577+0	4/22	DID	C
2VA-FD078	AUX	60-61/HH	577+0	4/22	DID	C
2VA-FD079	AUX	61/HH	577+0	4/22	DID	C
2VA-FD080	AUX	61/GG-HH	577+0	4/22	DID	C
2VA-FD081	AUX	61/HH	577+0	4/22	DID	C
2VA-FD083	AUX	63/NN	594+0	22/STAIR	DID	C
2VA-FD086	AUX	60/FF	594+0	22/STAIR	DID	C
2VA-FD087	AUX	59-60/QQ	594+0	22/ASB	LSS	B
2VA-FD088	AUX	60-61/QQ	594+0	22/ASB	LSS	B
2VA-FD093	AUX	58-59/QQ	594+0	22/ASB	LSS	B
2VA-FD097	AUX	61/CC-DD	594+0	22/STAIR	DID	C
2VA-FD108A	AUX	57-59/QQ	611+0	22/ASB	LSS	B
2VA-FD108B	AUX	57-59/QQ	611+0	22/ASB	LSS	B
2VA-FD114	AUX	59-60/KK-LL	594+0	18/22	DID	C
2VA-FD115	AUX	59-60/KK-LL	594+0	18/22	DID	C
2VA-FD137	AUX	60-61/FF-GG	560+0	4/18	DID	C
2VA-FD138	AUX	60-61/FF-GG	560+0	4/18	DID	C
2VA-FD139	AUX	61/GG	560+0	4/11	DID	C
2VA-FD141	AUX	62-63/DD	543+0	2/4	DID	C
2VA-FD142	AUX	60-61/KK	560+0	4/11	DID	C
2VA-FD143	AUX	62-63/KK	560+0	4/18	DID	C
2VA-FD144	AUX	63/KK	560+0	4/18	DID	C
2VA-FD145	AUX	61-62/MM-NN	560+0	4/11	DID	C
2VA-FD146	AUX	61-62/DD	560+0	2/5	LSS	B
2VA-FD147	AUX	61-62/DD	560+0	2/5	LSS	B
2VA-FD151	AUX	61-62/BB-CC	543+0	2/36	LSS	B
2VA-FD152	AUX	61-62/CC	543+0	2/36	LSS	B
2VA-FD153	AUX	60-61/GG-HH	594+0	4/22	DID	C
2VA-FD154	AUX	60-61/GG-HH	594+0	4/22	DID	C
2VA-FD157	AUX	63-64/AA-BB	543+0	CO2	HSS	A
2VA-FD158	AUX	64-65/AA-BB	543+0	CO2	HSS	A
2VA-FD160	AUX	57-58/QQ	543+0	4/ASB	LSS	B
2VA-FD161	AUX	57-58/QQ	543+0	4/ASB	LSS	B
2VA-FD163	AUX	58/EE	543+0	9/46	LSS	B
2VA-FD164	AUX	57-58/EE	543+0	4/9	DID	C
0BRS-FD001	AUX	54-55/DD-EE	554+0	10/22	DID	C
0BRS-FD010	AUX	57/DD-EE	554+0	9/10	DID	C
0BRS-FD019	AUX	59/DD-EE	554+0	9/22	DID	C
0BRX-FD001A	AUX	54-55/DD-EE	554+0	10/22	DID	C
0BRX-FD001B	AUX	54-55/DD-EE	554+0	10/22	DID	C

(continued)

Table 16.9-5-2

REQUIRED FIRE DAMPERS

DAMPER NUMBER	BLDG	LOCATION	ELEVATION	FIRE AREA INTERFACE	RISK CRITERIA	REMEDIAL ACTION CONDITION
0BRX-FD001C	AUX	54-55/DD-EE	554+0	10/22	DID	C
0BRX-FD001D	AUX	54-55/DD-EE	554+0	10/22	DID	C
0BRX-FD001E	AUX	54-55/DD-EE	554+0	10/22	DID	C
0BRX-FD001F	AUX	54-55/DD-EE	554+0	10/22	DID	C
0BRX-FD001G	AUX	54-55/DD-EE	554+0	10/22	DID	C
0BRX-FD001H	AUX	54-55/DD-EE	554+0	10/22	DID	C
0BRX-FD002	AUX	54-55/DD-EE	554+0	10/22	DID	C
0BRX-FD009	AUX	57/AA-BB	554+0	9/10	DID	C
0BRX-FD010	AUX	57/AA-BB	554+0	9/10	DID	C
0BRX-FD011	AUX	57/BB-CC	554+0	9/10	DID	C
0BRX-FD012	AUX	57/CC-DD	554+0	9/10	DID	C
0BRX-FD013	AUX	57/CC-DD	554+0	9/10	DID	C
0BRX-FD014	AUX	57/DD-EE	554+0	9/10	DID	C
0BRX-FD021	AUX	60/DD-EE	554+0	9/22	DID	C
0BRX-FD022A	AUX	60/DD-EE	554+0	9/22	DID	C
0BRX-FD022B	AUX	60/DD-EE	554+0	9/22	DID	C
0BRX-FD022C	AUX	60/DD-EE	554+0	9/22	DID	C
0BRX-FD022D	AUX	60/DD-EE	554+0	9/22	DID	C
0BRX-FD022E	AUX	60/DD-EE	554+0	9/22	DID	C
0BRX-FD022F	AUX	60/DD-EE	554+0	9/22	DID	C
0BRX-FD022G	AUX	60/DD-EE	554+0	9/22	DID	C
0BRX-FD022H	AUX	60/DD-EE	554+0	9/22	DID	C
0BRX-FD023	AUX	57/BB-CC	554+0	9/10	DID	C
1CRA-FD005A	AUX	54-55/DD-EE	594+0	21/22	HSS	A
1CRA-FD005B	AUX	54-55/DD-EE	594+0	21/22	HSS	A
1CRA-FD008	AUX	54/AA	594+0	21/35	HSS	A
1CRA-FD009	AUX	53-54/CC-DD	594+0	22/STAIR	DID	C
1CRA-FD010	AUX	53-54/CC	594+0	21/STAIR	HSS	A
1CRA-FD011	AUX	53/AA-BB	594+0	20/35	DID	C
1CRA-FD012	AUX	53/BB-CC	594+0	20/21	HSS	A

(continued)

Table 16.9-5-2

REQUIRED FIRE DAMPERS

DAMPER NUMBER	BLDG	LOCATION	ELEVATION	FIRE AREA INTERFACE	RISK CRITERIA	REMEDIAL ACTION CONDITION
1CRA-FD013	AUX	52/CC-DD	594+0	20/22	LSS	B
1CRA-FD016	AUX	54-55/DD-EE	574+0	22/45	LSS	B
1CRA-FD017	AUX	54-55/DD	574+0	17/45	HSS	A
1CRA-FD018	AUX	54-55/DD	574+0	17/45	HSS	A
1CRA-FD019	AUX	54/AA-BB	574+0	17/45	HSS	A
1CRA-FD020	AUX	57/CC-DD	574+0	16/17	HSS	A
1CRA-FD021	AUX	53-54/DD-EE	574+0	22/45	LSS	B
1CRA-FD022	AUX	55-56/DD	574+0	17/45	HSS	A
1CRA-FD023	AUX	56-57/DD	574+0	17/45	HSS	A
1CRA-FD024A	AUX	57/DD-EE	574+0	45/46	DID	C
1CRA-FD024B	AUX	57/DD-EE	574+0	45/46	DID	C
1CRA-FD025A	AUX	54-55/DD-EE	574+0	22/45	LSS	B
1CRA-FD025B	AUX	54-55/DD-EE	574+0	22/45	LSS	B
1CRA-FD026	AUX	54-55/EE	577+0	18/22	DID	C
1CRA-FD028	AUX	53-54/EE	577+0	18/22	DID	C
1CRA-FD029	AUX	54-55/EE	568+0	11/22	DID	C
1CRA-FD030	AUX	54-55/EE	568+0	11/22	DID	C
1CRA-FD039	AUX	57/EE-FF	577+0	18/18 (KC PUMPS)	DID	C
1CR-FD001	AUX	55-56/DD-EE	594+0	21/22	HSS	A
1CR-FD002	AUX	55-56/DD-EE	594+0	21/22	HSS	A
1CR-FD003	AUX	54/AA-BB	594+0	21/35	HSS	A
1CR-FD004	AUX	53-54/BB	594+0	21/35	HSS	A
1CR-FD005	AUX	53-54/BB	594+0	21/35	HSS	A
1CR-FD007	AUX	51/CC-DD	594+0	13/20	HSS	A
2CRA-FD005A	AUX	59-60/DD-EE	594+0	21/22	HSS	A
2CRA-FD005B	AUX	59-60/DD-EE	594+0	21/22	HSS	A
2CRA-FD008	AUX	60/AA-BB	594+0	19/21	HSS	A
2CRA-FD009	AUX	60-61/CC	594+0	19/22	LSS	B
2CRA-FD012	AUX	61/CC-DD	594+0	19/22	LSS	B
2CRA-FD015	AUX	59-60/DD-EE	574+0	22/46	LSS	B
2CRA-FD016	AUX	59-60/DD	574+0	16/46	HSS	A
2CRA-FD017	AUX	59-60/DD	574+0	16/46	HSS	A
2CRA-FD018	AUX	60/AA-BB	574+0	16/46	HSS	A
2CRA-FD019	AUX	58-59/DD	574+0	16/46	HSS	A
2CRA-FD020	AUX	57-58/DD	574+0	16/46	HSS	A
2CRA-FD021	AUX	60-61/DD-EE	574+0	22/46	LSS	B
2CRA-FD022A	AUX	59-60/DD-EE	574+0	22/46	LSS	B

(continued)

Table 16.9-5-2

REQUIRED FIRE DAMPERS

DAMPER NUMBER	BLDG	LOCATION	ELEVATION	FIRE AREA INTERFACE	RISK CRITERIA	REMEDIAL ACTION CONDITION
2CRA-FD022B	AUX	59-60/DD-EE	574+0	22/46	LSS	B
2CRA-FD023	AUX	59-60/EE	577+0	18/22	DID	C
2CRA-FD025	AUX	60-61/EE	577+0	18/22	DID	C
2CRA-FD026	AUX	59-60/EE	568+0	11/22	DID	C
2CRA-FD027	AUX	59-60/EE	568+0	11/22	DID	C
2CR-FD001	AUX	58-59/DD-EE	594+0	21/22	HSS	A
2CR-FD002	AUX	58-59/DD	594+0	21/22	HSS	A
2CR-FD003	AUX	63-64/CC	594+0	12/19	HSS	A
1VF-FD001A	AUX	51, NN-PP	605+10	22/24	LSS	B
1VF-FD001B	AUX	51, NN-PP	605+10	22/24	LSS	B
1VF-FD002A	AUX	50-51/NN-PP	631+6	24/38	DID	C
1VF-FD002B	AUX	50-51/NN-PP	631+6	24/38	DID	C
1VF-FD004	AUX	49/PP-QQ	631+6	24/38	DID	C
1VF-FD005	AUX	49-50/PP-QQ	631+6	24/38	DID	C
1VF-FD006	AUX	50-51	631+6	24/38	DID	C
1VF-FD007	AUX	50-51/KK-LL	605+10	22/24	LSS	B
1VF-FD010	AUX	50-51/KK	605+10	22/24	LSS	B
1VF-FD011	AUX	50-51/JJ-KK	631+6	22/38	LSS	B
1VF-FD013	AUX	50-51/JJ-KK	616+10	22/24	LSS	B
1VF-FD014	AUX	50-51/JJ-KK	616+10	22/24	LSS	B
2VF-FD001A	AUX	63, NN-PP	605+10	22/23	LSS	B
2VF-FD001B	AUX	63, NN-PP	605+10	22/23	LSS	B
2VF-FD002A	AUX	63-64/NN-PP	631+6	23/47	DID	C
2VF-FD002B	AUX	63-64/NN-PP	631+6	23/47	DID	C
2VF-FD004	AUX	65/PP-QQ	631+6	23/47	DID	C
2VF-FD005	AUX	64-65/PP-QQ	631+6	23/47	DID	C
2VF-FD006	AUX	63-64/PP-QQ	631+6	23/47	DID	C
2VF-FD007	AUX	63-64/KK-LL	605+10	22/23	LSS	B
2VF-FD010	AUX	63-64/KK	605+10	22/23	LSS	B
2VF-FD011	AUX	64-64/JJ-KK	631+6	22/47	LSS	B
2VF-FD013	AUX	63-64/JJ-KK	616+10	22/23	LSS	B
2VF-FD014	AUX	63-64/JJ-KK	616+10	22/23	LSS	B
1TB-FD001	TB1	18-19/V	594+0	TB1/SRV	DID	C
1TB-FD002	TB1	18-19/V	594+0	TB1/SRV	DID	C
1TB-FD003	TB1	18-19/V	594+0	TB1/SRV	DID	C
1TB-FD004	TB1	18-19/V	594+0	TB1/SRV	DID	C
1TB-FD005	TB1	18-19/V	594+0	TB1/SRV	DID	C
1TB-FD006	TB1	18-19/V	594+0	TB1/SRV	DID	C
1TB-FD007	TB1	21-22/V	594+0	TB1/SRV	DID	C
1TB-FD008	TB1	21-22/V	594+0	TB1/SRV	DID	C
1TB-FD009	TB1	21-22/V	594+0	TB1/SRV	DID	C
1TB-FD010	TB1	21-22/V	594+0	TB1/SRV	DID	C
1TB-FD011	TB1	21-22/V	594+0	TB1/SRV	DID	C
1TB-FD012	TB1	21-22/V	594+0	TB1/SRV	DID	C
1TB-FD032	TB1	18-19/V	594+0	TB1/SRV	DID	C

(continued)

Table 16.9-5-2

REQUIRED FIRE DAMPERS

DAMPER NUMBER	BLDG	LOCATION	ELEVATION	FIRE AREA INTERFACE	RISK CRITERIA	REMEDIAL ACTION CONDITION
1TB-FD038	TB1	16-17/V	594+0	TB1/SRV	DID	C
1TB-FD039	TB1	16-17/V	594+0	TB1/SRV	DID	C
1TB-FD040	TB1	16/V	594+0	TB1/SRV	DID	C
1TB-FD043	TB1	30-31/1J-1K	568+0	TB1/OTT	DID	C
1TB-FD044	TB1	32/1J-1K	594+0	TB1/MTOT	DID	C
1TB-FD045	TB1	30/1J-1K	594+0	TB1/MTOT	DID	C
1TB-FD046	TB1	32/1K-1L	568+0	TB1/OTT	DID	C
2TB-FD013	TB2	21-22/P	594+0	TB2/SRV	DID	C
2TB-FD014	TB2	21-22/P	594+0	TB2/SRV	DID	C
2TB-FD015	TB2	21-22/P	594+0	TB2/SRV	DID	C
2TB-FD016	TB2	21-22/P	594+0	TB2/SRV	DID	C
2TB-FD017	TB2	21-22/P	594+0	TB2/SRV	DID	C
2TB-FD018	TB2	21-22/P	594+0	TB2/SRV	DID	C
2TB-FD019	TB2	18-19/P	594+0	TB2/SRV	DID	C
2TB-FD020	TB2	18-19/P	594+0	TB2/SRV	DID	C
2TB-FD021	TB2	18-19/P	594+0	TB2/SRV	DID	C
2TB-FD022	TB2	18-19/P	594+0	TB2/SRV	DID	C
2TB-FD023	TB2	18-19/P	594+0	TB2/SRV	DID	C
2TB-FD024	TB2	18-19/P	594+0	TB2/SRV	DID	C
2TB-FD031	TB2	32/2K-2L	568+0	TB2/OTT	DID	C
2TB-FD032	TB2	18/P	594+0	TB2/SRV	DID	C
2TB-FD036	TB2	16-17/P	594+0	TB2/SRV	DID	C
2TB-FD038	TB2	17-18/P	594+0	TB2/SRV	DID	C
2TB-FD039	TB2	32/2J-2K	594+0	TB2/MTOT	DID	C
2TB-FD040	TB2	30/2J-2K	594+0	TB2/MTOT	DID	C
2TB-FD041	TB2	30-31/2J/2K	568+0	TB2/OTT	DID	C

Table 16.9-5-3

HIGH SAFETY SIGNIFICANT (HSS) FIRE AREAS*

FIRE AREA	BLDG	ELEVATION	DESCRIPTION
6	AUX	560+0	Unit 1 Electrical Pen Room El 560
12	AUX	577+0	Unit 2 Electrical Pen Room El 577
13	AUX	577+0	Unit 1 Electrical Pen Room El 577
14	AUX	577+0	Unit 2 4160V Essential Swgr Room (2ETA)
15	AUX	577+0	Unit 1 4160V Essential Swgr Room (1ETA)
16	AUX	574+0	Unit 2 Cable Room El 574
17	AUX	574+0	Unit 1 Cable Room El 574
21	AUX	594+0	Main Control Room El 594

*High Safety Significant (HSS) Fire Areas are defined as the areas with HSS fire barrier features in accordance with the Catawba NFPA 805 Monitoring Program.

Table 16.9-5-4

FIRE AREAS AND SHUTDOWN TRAIN / METHOD

FIRE AREA	FIRE AREA DESCRIPTIONS	ASSURED SHUTDOWN TRAIN / METHOD
1	ND & NS Pump Room El 522 (Common)	SSS
2	Unit 2 CA Pump Room El 543	SSS
3	Unit 1 CA Pump Room El 543	SSS
4	Aux Bldg. Gen Area & NV Pump Room El 543 (Common)	SSS
5	Unit 2 Electrical Pen Room El 560	A
6	Unit 1 Electrical Pen Room El 560	A
7	Unit 2 4160V Essential SWGR Room El 560	A
8	Unit 1 4160V Essential SWGR Room El 560	A
9	Unit 2 Battery Room El 554	SSS
10	Unit 1 Battery Room El 554	SSS
11	Aux Bldg. Gen Area & U1 KC Pump Room El 560 (Common)	SSS
12	Unit 2 Electrical Pen Room El 577	B
13	Unit 1 Electrical Pen Room El 577	B
14	Unit 2 4160V Essential SWGR Room El 577	B
15	Unit 1 4160V Essential SWGR Room El 577	B
16	Unit 2 Cable Room El 574	SSS
17	Unit 1 Cable Room El 574	SSS
18	Aux Bldg. Gen Area & U2 KC Pump Room El 577 (Common)	SSS
19	Unit 2 Electrical Pen Room El 594	A
20	Unit 1 Electrical Pen Room El 594	A
21	Control Room El 594 (Common)	SSS
22	Aux Bldg. Gen Area El 594 (Common)	SSS
23	Unit 2 Fuel Storage Area El 605	A
24	Unit 1 Fuel Storage Area El 605	A
25	Diesel Generator Bldg. 1A El 556	B
25A	Diesel Generator Bldg. 1A Stairwell	B
26	Diesel Generator Bldg. 1B El 556	A
26B	Diesel Generator Bldg. 1B Stairwell	A
27	Diesel Generator Bldg. 2A El 556	B
27A	Diesel Generator Bldg. 2A Stairwell	B
28	Diesel Generator Bldg. 2B El 556	A
28B	Diesel Generator Bldg. 2B Stairwell	A
29	Train A RN Pump Structure El 600 (Common)	B
30	Train B RN Pump Structure El 600 (Common)	A
31	Unit 2 Train A Aux Shutdown Panel El 543	B
32	Unit 1 Train A Aux Shutdown Panel El 543	B
33	Unit 2 Train B Aux Shutdown Panel El 543	A
34	Unit 1 Train B Aux Shutdown Panel El 543	A
35	Control Room Tagout Area El 594	A or B
36	Unit 2 Turbine Driven CA Pump Control Panel Room El 543	B
37	Unit 1 Turbine Driven CA Pump Control Panel Room El 543	B
38	Unit 1 Fuel Storage Area HVAC Room El 631	A or B

(continued)

Table 16.9-5-4

FIRE AREAS AND SHUTDOWN TRAIN / METHOD

FIRE AREA	FIRE AREA DESCRIPTIONS	ASSURED SHUTDOWN TRAIN / METHOD
39	Unit 2 Turbine Driven CA Pump Pit El 543	B
40	Unit 1 Turbine Driven CA Pump Pit El 543	B
41	DG1A Sequencer Tunnel El 556	B
42	DG1B Sequencer Tunnel El 556	A
43	DG2A Sequencer Tunnel El 556	B
44	DG2B Sequencer Tunnel El 556	A
45	Unit 1 Cable Room Corridor El 574	B
46	Unit 2 Cable Room Corridor El 574	B
47	Unit 2 Fuel Storage Area HVAC Room El 631	A or B
48	Unit 2 Interior Doghouse	A and B
49	Unit 1 Interior Doghouse	A and B
50	Unit 2 Exterior Doghouse	A and B
51	Unit 1 Exterior Doghouse	A and B
ASB	Auxiliary Service Building	A or B
RB1	Unit 1 Reactor Building	A and B
RB2	Unit 2 Reactor Building	A and B
SRV	Service Building	B
SSF	Standby Shutdown Facility	A or B
STAIR*	Stairway	See Note
TB1	Unit 1 Turbine Building	A or B
TB2	Unit 2 Turbine Building	A or B
YRD**	Yard Area	A or B

* IF the barrier in a stairway is adjacent to a HSS Fire Area (see Table 16.9-5-3), enter CONDITION A; otherwise enter CONDITION C.
** Exterior walls that interface with the YRD do not require entry into a CONDITION statement and therefore do not have a REQUIRED ACTION.

A = A TRAIN
B = B TRAIN
SSS = STANDBY SHUTDOWN SYSTEM

BASES

The functional integrity of the Fire Rated Assemblies and associated sealing devices ensures that fires will be confined or adequately retarded so as not to spread between fire areas/compartments.

The fire barriers and associated penetration seals are passive elements in the facility fire protection program and are subject to periodic inspections.

Risk-informed insights from the Fire PRA process can apply to compensatory actions. The safety significance of the fire area can provide relief for required compensatory actions. In addition, the presence of functional fire detection can reduce the required compensatory actions. Functional fire detection in the area provides early warning of a fire for fire brigade response. Fire detection can provide a compensatory action equivalent to or better than fire watch.

Fire barrier penetration seals, including cable/pipe penetration seals, fire doors, and fire dampers, are considered FUNCTIONAL when the visually observed condition indicates no abnormal change in appearance or abnormal degradation. An evaluation is performed to determine the cause of any identified fire barrier penetration seal abnormal change in appearance or abnormal degradation and the effect of this change on the ability of the fire barrier penetration seal to perform its function. Based on this evaluation additional inspections may be performed.

During periods of time when a barrier is not FUNCTIONAL, either:

(1) Perform the recommended fire watch in accordance with the criteria in the remedial actions, or

(2) a licensee may choose to implement a different required action or a combination of actions (e.g., additional administrative controls, operator briefings, temporary procedures, interim shutdown strategies, operator manual actions, temporary fire barriers, temporary detection or suppression systems). Such a change must be made to the approved Fire Protection Plan (FPP). However, the licensee must complete a documented evaluation of the impact of the proposed required action to the FPP. The evaluation must demonstrate that the required actions would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. Any change to the FPP must maintain compliance with the General Design Criteria and 10 CFR 50.48(a).

The evaluation of the required action should incorporate risk insights regarding the location, quantity, and type of combustible material in the fire area; the presence of ignition sources and their likelihood of occurrence; the automatic fire suppression and the fire detection capability in the fire area; the manual fire suppression capability in the fire area; and the human error probability where applicable.

BASES (continued)

The expectation is to promptly complete the corrective action at the first available opportunity and eliminate the reliance on the required action.

This SLC is part of the Catawba Fire Protection Program and therefore subject to the provisions of Section 2.C.(5) of the Catawba Renewed Facility Operating Licenses.

REFERENCES

1. Catawba UFSAR, Section 9.5.1.
2. Catawba Nuclear Station 10 CFR 50.48(c) Fire Protection Safety Evaluation (SE).
3. Catawba Plant Design Basis Specification for Fire Protection, CNS-1465.00-00-0006, as revised.
4. Catawba UFSAR, Section 18.2.8.
5. Catawba License Renewal Commitments, CNS-1274.00-00-0016, Section 4.12.2.
6. NRC Regulatory Issue Summary 2005-07, Compensatory Measures to Satisfy the Fire Protection Program Requirements, April 19, 2005.
7. Catawba Renewed Facility Operating License Conditions 2.C.(5).
8. CNC-1435.00-00-0084, Catawba NFPA 805 Monitoring Program.
9. CNC-1435.00-00-0044, Fire Protection Nuclear Safety Capability Assessment.
10. CN-1209.10 series drawings.
11. CN-1105 series drawings.

16.9 AUXILIARY SYSTEMS

16.9-6 Fire Detection Instrumentation

COMMITMENT The Fire Detection Instrumentation for each fire detection zone shown in Table 16.9-6-1 shall be FUNCTIONAL.

APPLICABILITY: At all times.

---NOTE---

See Table 16.9-6-1 for Risk Criteria

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. All fire detection instruments in all fire zones non-functional due to fire detection system failure.	A.1 Establish roving fire watch in accordance with fire detection system operating procedure.	1 hour
	<u>AND</u>	
	A.2 Restore the non-functional fire detection system to FUNCTIONAL status.	72 hours
	<u>OR</u>	
	A.3 If the non-functional fire detection system cannot be restored to FUNCTIONAL status within 72 hours, enter Conditions B through E as applicable.	72 hours
B. One or more of the fire detection instruments in a HSS* EFA zone is non-functional.	B.1 Establish a continuous fire watch.	1 hour
	<u>OR</u>	
		(continued)

REMEDIAL ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2.1 Verify affected area has FUNCTIONAL full area automatic fire suppression.	1 hour
	AND	
	B.2.2 Establish an hourly fire watch patrol.	1 hour
	OR	
	B.3 Complete an evaluation as permitted by NRC RIS 2005-07 to implement required action(s).	Prior to terminating Required Action B.1 or B.2
C. One or more fire detection instruments in a LSS** EFA zone is non-functional.	C.1 Establish an hourly fire watch.	1 hour
	OR	
	C.2.1 Verify affected area has FUNCTIONAL full area automatic fire suppression.	1 hour
	AND	
	C.2.2 Establish a once per shift fire watch patrol.	1 hour
	OR	
	C.3 Complete an evaluation as permitted by NRC RIS 2005-07 to implement required action(s).	Prior to terminating Required Action C.1 or C.2

(continued)

REMEDIAL ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more fire detection instruments in a DID*** EFA zone is non-functional.	D.1 Establish a once per shift fire watch.	1 hour
	<u>OR</u>	
	D.2 Verify affected area has FUNCTIONAL full area automatic fire suppression.	1 hour
	<u>OR</u>	
	D.3 Complete an evaluation as permitted by NRC RIS 2005-07 to institute required action(s).	Prior to terminating Required Action D.1 or D.2
E. One or more fire detection instruments in a CONTAINMENT**** EFA zone is non-functional.	E.1 IF Containment is accessible, establish a once per 8 hour fire watch.	1 hour
	<u>OR</u>	(continued)

REMEDIAL ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. (continued)	E.2	Monitor containment air temperature at least once per hour at the locations listed in Technical Specification Surveillance Requirements 3.6.5.1 or 3.6.5.2 as applicable. [Note: Containment air temperature may be monitored continuously by verifying the Operator Aid Computer is available with computer points C1(2)P1500 (upper containment average air temperature) and C1(2)P1501 (lower containment average air temperature) in service.]	1 hour
	OR		
	E.3	Complete an evaluation as permitted by NRC RIS 2005-07 to implement required action(s).	Prior to terminating Required Action E.1 or E.2

*High Safety Significant (HSS) EFA Zone Detection is defined by the Catawba NFPA 805 Monitoring Program.

**Low Safety Significant (LSS) EFA Zone Detection is defined as that detection required by Risk as part of the Catawba NFPA 805 fire protection program.

***Defense-in-Depth (DID) EFA Zone Detection is defined as detection devices required by engineering evaluation or previous licensing approval as part of the Catawba NFPA 805 fire protection program; or detection protecting equipment important to safety added to the fire protection program as 'best practice'.

****CONTAINMENT EFA Zone Detection is defined as detection devices located inside the Reactor Building (excluding the Annulus area).

TESTING REQUIREMENTS

TEST		FREQUENCY
TR 16.9-6-1	Verify the NFPA Standard 72D supervised circuits supervision associated with the detector alarms that are accessible during plant operation is FUNCTIONAL.	12 months
TR 16.9-6-2	Perform a visual inspection of each flame detection instrument.	12 months
TR 16.9-6-3	Perform a visual inspection of each smoke detection instrument accessible during plant operation.	12 months
TR 16.9-6-4	Perform a functional test of each restorable line type heat detection instrument by applying a heat source.	12 months
TR 16.9-6-5	Perform a functional test of each flame detection instrument.	12 months
TR 16.9-6-6	Perform a functional test of each smoke detection instrument accessible during plant operation.	12 months
TR 16.9-6-7	Perform a non-destructive functional test of each non-restorable line type heat detection instrument accessible during plant operation.	12 months
TR 16.9-6-8	Perform a functional test of each smoke detection instrument not accessible during plant operation.	18 months
TR 16.9-6-9	Perform a non-destructive functional test of each non-restorable line type heat detection instrument not accessible during plant operation.	18 months
TR 16.9-6-10	Verify the NFPA Standard 72D supervised circuits supervision associated with the detector alarms that are not accessible during plant operation is FUNCTIONAL.	18 months

Table 16.9-6-1

Fire Detection Instruments

EFA ZONE	DESCRIPTION	LOCATION		REQUIRED INSTRUMENTS(1)			RISK CRITERIA
		COLUMN OR AZIMUTH	ELEVATION	SMOKE	FLAME	HEAT	
1	ND Pump 1B	GG-53	522 + 0	1	0	0	DID
2	ND Pump 1A	FF-53	522 + 0	1	0	0	DID
3	NS Pump 1B	GG-54	522 + 0	3	0	0	DID
4	NS Pump 1A	GG-55	522 + 0	2	0	0	DID
5	ND Pump 2B	GG-61	522 + 0	1	0	0	DID
6	ND Pump 2A	FF-61	522 + 0	1	0	0	DID
7	NS Pump 2B	GG-60	522 + 0	3	0	0	DID
8	NS Pump 2A	GG-59	522 + 0	2	0	0	DID
9	CA Pumps (Unit 1)	BB-51	543 + 0	14	0	0	LSS
10	Mech Pen Room	JJ-52	543 + 0	3	0	0	LSS
11	Corridor/Cables	NN-51	543 + 0	6	0	0	DID
12	NV Pump #1 (Unit 1)	JJ-53	543 + 0	1	0	0	DID
13	NI Pump 1B	HH-53	543 + 0	1	0	0	DID
14	NI Pump 1A	GG-53	543 + 0	1	0	0	DID
15	NV Pump 1B	JJ-54	543 + 0	2	0	0	DID
16	NV Pump 1A	JJ-55	543 + 0	2	0	0	DID
17	Aisles/Cables	KK-56	543 + 0	18	0	0	DID
18	Aisles/Cables	EE-55	543 + 0	6	0	0	LSS
19	CA Pumps (Unit 2)	BB-63	543 + 0	14	0	0	LSS
20	Mech Pen Room	JJ-62	543 + 0	3	0	0	LSS
21	Aisles/Cables	NN-61	543 + 0	6	0	0	DID
22	NV Pump #2 (Unit 2)	JJ-60	543 + 0	1	0	0	DID
23	NI Pump 2B	HH-60	543 + 0	1	0	0	DID
24	NI Pump 2A	GG-60	543 + 0	1	0	0	DID
25	NV Pump 2B	JJ-59	543 + 0	2	0	0	DID
26	NV Pump 2A	JJ-58	543 + 0	2	0	0	DID
27	Aisles/Cables	KK-59	543 + 0	20	0	0	DID
28	Aisles/Cables	EE-58	543 + 0	6	0	0	LSS
29	Swgr Equip Room	AA-50	560 + 0	7	0	0	HSS
30	Elect Pen Room	CC-50	560+ 0	8	0	0	LSS
31	Corridor/Cables	EE-53	560 + 0	5	0	0	DID
32	Corridor/Cables	KK-52	560 + 0	8	0	0	DID

Table 16.9-6-1

Fire Detection Instruments

EFA ZONE	DESCRIPTION	LOCATION		REQUIRED INSTRUMENTS[1]			RISK CRITERIA
		COLUMN OR AZIMUTH	ELEVATION	SMOKE	FLAME	HEAT	
33	Corridor/Cables	NN-54	560 + 0	9[3]	0	0	DID
34	Aisles/Cables	JJ-56	560 + 0	14	0	0	DID
35	Motor Control Center	GG-56	560 + 0	2	0	0	DID
36	Cable Tray Access	FF-56	568 + 0	2	0	0	DID
37	Equip Batteries	DD-55	554 + 0	5	0	0	LSS
38	Equip Batteries	CC-55	544 + 0	5	0	0	LSS
39	Battery Room	CC-56	554 + 0	17	0	0	LSS
41	Swgr Equip Room	AA-64	560 + 0	7	0	0	HSS
42	Elect Pen Room	CC-65	560 + 0	8	0	0	LSS
43	Corridor/Cables	FF-61	560 + 0	5	0	0	DID
44	Aisles/Cables	KK-63	560 + 0	8	0	0	DID
45	Aisles/Cables	NN-60	560 + 0	12[3]	0	0	DID
46	Aisles/Cables	HH-59	560 + 0	13	0	0	DID
47	Motor Control Center	GG-58	560 + 0	2	0	0	DID
48	Cable Tray Access	FF-58	560 + 0	2	0	0	DID
49	Equip Batteries	DD-60	560 + 0	5	0	0	LSS
50	Equip Batteries	CC-60	560 + 0	5	0	0	LSS
51	Battery Room	CC-59	560 + 0	17	0	0	LSS
53	Swgr Equip Room	AA-49	577 + 0	7	0	0	HSS
54	Elect Pen Room	CC-50	577 + 0	10	0	0	LSS
55	Aisles/Cables	NN-52	577 + 0	8	0	0	DID
56	Aisles/Cables	PP-55	577 + 0	13[3]	0	0	DID
57	Aisles/Cables	LL-55	577 + 0	11	0	0	DID
58	Aisles/Cables	HH-55	577 + 0	17[3]	0	0	DID
59	Motor Control Center	EE-54	577 + 0	2	0	0	DID
60	Cable Room	CC-56	574 + 0	18	0	0	LSS
62	Swgr Equip Room	AA-64	577 + 0	7	0	0	HSS
63	Elect Pen Room	CC-64	577 + 0	10	0	0	LSS
64	Aisles/Cables	PP-62	577 + 0	8	0	0	DID
65	Aisles/Cables	PP-59	577 + 0	11[3]	0	0	DID
66	Aisles/Cables	LL-59	577 + 0	11	0	0	DID
67	Aisles/Cables	HH-59	577 + 0	17[3]	0	0	DID

Table 16.9-6-1

Fire Detection Instruments

EFA ZONE	DESCRIPTION	LOCATION		REQUIRED INSTRUMENTS(1)			RISK CRITERIA
		COLUMN OR AZIMUTH	ELEVATION	SMOKE	FLAME	HEAT	
68	Motor Control Center	FF-60	577 + 0	2	0	0	DID
69	Cable Room	CC-59	577 + 0	18	0	0	LSS
71	Elect Pen Room	CC-51	594 + 0	10	0	0	LSS
72	Control Room	CC-56	594 + 0	25	0	0	LSS
73	Vent Equip Room	FF-56	594 + 0	9	0	0	DID
74	Aisles/Cables	LL-56	594 + 0	25	0	0	DID
76	Aisles/Cables	PP-54	594 + 0	15	0	0	DID
79	Elect Pen Room	BB-63	594 + 0	11	0	0	LSS
80	Control Room	BB-59	594 + 0	22	0	0	LSS
81	Vent Equip Room	FF-58	594 + 0	12	0	0	DID
82	Aisles/Cables	KK-58	594 + 0	27	0	0	DID
84	Aisles/Cables	NN-58	594 + 0	17	0	0	DID
89	Fuel Pool Area (Unit 1)	PP-50	605 + 10	19	7	0	DID
90	Fuel Pool Area (Unit 2)	PP-64	605 + 10	19	7	0	DID
129	Fuel Pool Purge Room (Unit 1)	NN-50	631 + 6	6	0	0	LSS
131	Reactor Bldg (Unit 1)	0^0-45^0	Below 565 + 3	4	0	0	CONTAINMENT
132	Reactor Bldg (Unit 1)	45^0-90^0	Below 565 + 3	3	0	0	CONTAINMENT
133	Reactor Bldg (Unit 1)	90^0-135^0	Below 565 + 3	4	0	0	CONTAINMENT
134	Reactor Bldg (Unit 1)	135^0-180^0	Below 565 + 3	5	0	0	CONTAINMENT
135	Reactor Bldg (Unit 1)	180^0-225^0	Below 565 + 3	4	0	0	CONTAINMENT
136	Reactor Bldg (Unit 1)	270^0-315^0	Below 565 + 3	3	0	0	CONTAINMENT
137	Reactor Bldg (Unit 1)	315^0-0^0	Below 565 + 3	8	0	0	CONTAINMENT
138	Reactor Bldg (Unit 1)	0^0-45^0	Below 586 + 3	6	0	0	CONTAINMENT
139	Reactor Bldg (Unit 1)	45^0-90^0	Below 586 + 3	4	0	0	CONTAINMENT
140	Reactor Bldg (Unit 1)	90^0-135^0	Below 565 + 3	3	0	0	CONTAINMENT
141	Reactor Bldg (Unit 1)	135^0-180^0	Below 586 + 3	8	0	0	CONTAINMENT
142	Reactor Bldg (Unit 1)	180^0-225^0	Below 586 + 3	5	0	0	CONTAINMENT
143	Reactor Bldg (Unit 1)	315^0-0^0	Below 586 + 3	5	0	0	CONTAINMENT
144	Reactor Bldg (Unit 1)	0^0-45^0	Below 593 + 2 1/2	14	0	0	CONTAINMENT
145	Reactor Bldg (Unit 1)	45^0-90^0	Below 593 + 2 1/2	16	0	0	CONTAINMENT

Table 16.9-6-1

Fire Detection Instruments

EFA ZONE	DESCRIPTION	LOCATION		REQUIRED INSTRUMENTS[1]			RISK CRITERIA
		COLUMN OR AZIMUTH	ELEVATION	SMOKE	FLAME	HEAT	
146	Reactor Bldg (Unit 1)	90^0-135^0	Below 593 + 2 1/2	11	0	0	CONTAINMENT
147	Reactor Bldg (Unit 1)	135^0-180^0	Below 593 + 2 1/2	10	0	0	CONTAINMENT
148	Reactor Bldg (Unit 1)	180^0-225^0	Below 593 + 2 1/2	2	0	0	CONTAINMENT
149	Reactor Bldg (Unit 1)	315^0-0^0	Below 593 + 2 1/2	7	0	0	CONTAINMENT
150	Reactor Bldg (Unit 2)	0^0-45^0	Below 565 + 3	4	0	0	CONTAINMENT
151	Reactor Bldg (Unit 2)	45^0-90^0	Below 565 + 3	3	0	0	CONTAINMENT
152	Reactor Bldg (Unit 2)	90^0-135^0	Below 565 + 3	4	0	0	CONTAINMENT
153	Reactor Bldg (Unit 2)	135^0-180^0	Below 565 + 3	5	0	0	CONTAINMENT
154	Reactor Bldg (Unit 2)	180^0-225^0	Below 565 + 3	2	0	0	CONTAINMENT
155	Reactor Bldg (Unit 2)	270^0-315^0	Below 565 + 3	4	0	0	CONTAINMENT
156	Reactor Bldg (Unit 2)	315^0-0^0	Below 565 + 3	6	0	0	CONTAINMENT
157	Reactor Bldg (Unit 2)	0^0-45^0	Below 586 + 6	6	0	0	CONTAINMENT
158	Reactor Bldg (Unit 2)	45^0-90^0	Below 586 + 6	4	0	0	CONTAINMENT
159	Reactor Bldg (Unit 2)	90^0-135^0	Below 586 + 6	3	0	0	CONTAINMENT
160	Reactor Bldg (Unit 2)	135^0-180^0	Below 586 + 6	8	0	0	CONTAINMENT
161	Reactor Bldg (Unit 2)	180^0-225^0	Below 586 + 6	5	0	0	CONTAINMENT
162	Reactor Bldg (Unit 2)	315^0-0^0	Below 586 + 6	5	0	0	CONTAINMENT
163	Reactor Bldg (Unit 2)	0^0-45^0	Below 593 + 2 1/2	13	0	0	CONTAINMENT
164	Reactor Bldg (Unit 2)	45^0-90^0	Below 593 + 2 1/2	17	0	0	CONTAINMENT
165	Reactor Bldg (Unit 2)	90^0-135^0	Below 593 + 2 1/2	13	0	0	CONTAINMENT
166	Reactor Bldg (Unit 2)	135^0-180^0	Below 593 + 2 1/2	10	0	0	CONTAINMENT
167	Reactor Bldg (Unit 2)	180^0-225^0	Below 593 + 2 1/2	2	0	0	CONTAINMENT
168	Reactor Bldg (Unit 2)	315^0-0^0	Below 593 + 2 1/2	7	0	0	CONTAINMENT
169	NC Pump 1A	45^0	593 + 2 1/2	0	0	1[2]	CONTAINMENT
170	NC Pump 1B	135^0	593 + 2 1/2	0	0	1[2]	CONTAINMENT
171	NC Pump 1C	225^0	593 + 2 1/2	0	0	1[2]	CONTAINMENT
172	NC Pump 1D	315^0	593 + 2 1/2	0	0	1[2]	CONTAINMENT
173	NC Pump 2A	45^0	Below 593 + 2 1/2	0	0	1[2]	CONTAINMENT
174	NC Pump 2B	135^0	Below 593 + 2 1/2	0	0	1[2]	CONTAINMENT
175	NC Pump 2C	225^0	Below 593 + 2 1/2	0	0	1[2]	CONTAINMENT
176	NC Pump 2D	315^0	Below 593 + 2 1/2	0	0	1[2]	CONTAINMENT
177	Filter Bed Unit 1A	0^0	Below 565 + 3	2	0	0	CONTAINMENT

Table 16.9-6-1

Fire Detection Instruments

EFA ZONE	DESCRIPTION	LOCATION		REQUIRED INSTRUMENTS[1]			RISK CRITERIA
		COLUMN OR AZIMUTH	ELEVATION	SMOKE	FLAME	HEAT	
178	Filter Bed Unit 1B	180^0	Below 565 + 3	2	0	0	CONTAINMENT
179	Filter Bed Unit 2A	0^0	565 + 3	2	0	0	CONTAINMENT
180	Filter Bed Unit 2B	180^0	565 + 3	2	0	0	CONTAINMENT
181a	Annulus (Unit 1)	0^0-350^0	561 + 0	0	0	1[2]	DID
181b	Annulus (Unit 1)	10^0-360^0	583 + 0	0	0	1[2]	DID
181c	Annulus (Unit 1)	0^0-350^0	604 + 0	0	0	1[2]	DID
181d	Annulus (Unit 1)	10^0-360^0	629 + 5	0	0	1[2]	DID
181e	Annulus (Unit 1)	0^0-350^0	649 + 5	0	0	1[2]	DID
181f	Annulus (Unit 1)	10^0-360^0	664 + 0	0	0	1[2]	DID
182a	Annulus (Unit 2)	0^0-350^0	561 + 0	0	0	1[2]	DID
182b	Annulus (Unit 2)	10^0-360^0	583 + 0	0	0	1[2]	DID
182c	Annulus (Unit 2)	0^0-350^0	604 + 0	0	0	1[2]	DID
182d	Annulus (Unit 2)	10^0-360^0	629 + 5	0	0	1[2]	DID
182e	Annulus (Unit 2)	0^0-350^0	649 + 5	0	0	1[2]	DID
182f	Annulus (Unit 2)	10^0-360^0	664 + 0	0	0	1[2]	DID
183	Fuel Pool Purge Room (Unit 2)	NN-64	631 + 6	6	0	0	LSS
184	HVAC Duct for Rooms 331 & 332	FF-53	543 + 0	1(Duct)	0	0	DID
185	HVAC Duct for Rooms 203, 205, 205A, 206A, 206B, 207, & 209A	MM-60	543 + 0	1(Duct)	0	0	DID
186	HVAC Duct for Rooms 301, 302, 305, & 307	NN-60	560 + 0	1(Duct)	0	0	DID
212	Aisles/Cables	GG-57	522 + 0	2	0	0	DID
213	Aux Battery Room	AA-55	544 + 0	4	0	0	LSS
214	Aux Control Power Batteries	AA-59	560 + 0	4	0	0	LSS
215	D/G Corridor/ Battery/Stairwell	BB-45	556 + 0	5	0	0	LSS
216	D/G Corridor/ Battery/Stairwell	AA-45	556 + 0	4	0	0	LSS

Table 16.9-6-1

Fire Detection Instruments

EFA ZONE	DESCRIPTION	LOCATION		REQUIRED INSTRUMENTS(1)			RISK CRITERIA
		COLUMN OR AZIMUTH	ELEVATION	SMOKE	FLAME	HEAT	
217	D/G Corridor/ Battery/Stairwell	CC-71	560 + 0	5	0	0	LSS
218	D/G Corridor/ Battery/Stairwell	BB-71	560 + 0	4	0	0	LSS
219	Mech Pen Room	HH-52	577 + 0	6	0	0	DID
220	Mech Pen Room	JJ-62	577 + 0	6	0	0	DID
222	Airlock Access (Unit 1)	JJ-51	605 + 10	1	0	0	DID
224	Airlock Access (Unit 2)	JJ-63	605 + 10	1	0	0	DID
225	RN Pump Structure (B Train)	West Section	600 + 0	8	0	0	DID
226	RN Pump Structure (A Train)	East Section	600 + 0	8	0	0	DID
231	Reactor Bldg (Unit 1)	260^0-303^0	Below 668 + 10	10	0	0	CONTAINMENT
232	Reactor Bldg (Unit 2)	260^0-303^0	Below 668 + 10	10	0	0	CONTAINMENT

(1) The fire detection instruments located within the containment are not required to be FUNCTIONAL during the performance of Type A containment leakage rate tests.

(2) Line type detector.

(3) See Bases section for explanation of the number of required detectors for this zone.

BASES

FUNCTIONALITY of the detection instrumentation ensures that adequate warning capability is available for prompt detection of fires. Prompt detection and suppression of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility Fire Protection Program.

Definitions:

Conventional Detection — A detector where the signal received does not identify the individual device operated.

Addressable Detection — A detector with discrete identification that can have its status individually identified.

An acceptable method to satisfy the functional test requirement for smoke detectors is by pre-testing (actuates at the required setpoint within the required accuracy) the detection instruments, replacing all detection instruments on a signaling circuit, and then actuating one detection instrument on the signaling circuit to verify FUNCTIONALITY of alarm.

The following detectors are not included in the Table 16.9-6-1 EFA Zones as "REQUIRED INSTRUMENTS" because they are located in high radiation areas with minimal combustible loading, or because they are located in non-safety related areas. As a result, they are also exempt from the TESTING REQUIREMENTS given in this SLC:

EFA ZONE	DETECTOR NUMBER (NOTE 1)	ROOM NUMBER	LOCATION	REASON CODE
33	FD033D04	315	560, Valve Gallery, LL/53	1
45	FD045D04	306	560, Valve Gallery, LL/61	1
56	FD056D14	401	577, Elevator Corridor, RR-SS/55-58	2
56	FD056D15	401	577, Elevator Corridor, RR-SS/55-58	2
56	FD056D16	401	577, Elevator Corridor, RR-SS/55-58	2
58	FD058D01	476	577, U1 Letdown Hx Room, KK/53	1
58	FD058D02	477	577, U1 Moderating Hx Room, JJ/53	1
58	FD058D03	474	577, U1 Letdown Reheat Hx Room	1
58	FD058D04	475	577, U1 Letdown Chiller Hx Room	1
65	FD065D01	405	577, Filter Bunker Room, RR-SS/58-60	2
65	FD065D02	405	577, Filter Bunker Room, RR-SS/58-60	2
65	FD065D04	404B	577, Radwaste Feed Skid and Dewatering Pump Room, PP-QQ/59-60	1
65	FD065D12	403	577, Container and Drum Stg. Room, NN/57-58	1
67	FD067D15	465	577, U2 Letdown Reheat Hx Room	1
67	FD067D16	466	577, U2 Letdown Chiller Hx Room	1
67	FD067D17	467	577, U2 Letdown Hx Room, KK/61	1
67	FD067D18	468	577, U2 Moderating Hx Room, JJ/61	1

BASES (continued)

REASON CODES:
1. High radiation area
2. Non-safety related area

For the individual detector designation, see the EFA System Detector Series drawings (CNM-1376.00 series).

Excluding the detectors in the above listed rooms from the "REQUIRED INSTRUMENTS" is acceptable for the following reasons:

- All of the above listed detectors, with the exception of detector FD065D12, are located in rooms that consist of very little to no combustible loading. These rooms contain primarily piping and mechanical equipment such as tanks, demineralizers, or heat exchangers. This type of equipment is not significantly prone to fire damage or fire propagation. In the case of detector FD065D12, the room is separated from adjacent areas with fire rated three-foot thick concrete walls and has additional detectors in the room that will remain in the scope of this SLC.
- All of the above listed rooms are adjacent to areas that also contain smoke detectors that alarm to the control room. Thus, in the unlikely event that a significant fire should occur in one of these rooms, the smoke detectors in the adjacent areas would provide an alarm to the control room.
- These rooms are either located in non-safety related areas, or are located in safety related areas, but contain components that do not perform a safe shutdown function.
- The areas adjacent to these rooms rely on the same assured train of post fire safe shutdown. Therefore, a fire in any of these rooms or in an adjacent area would have no adverse impact on post fire safe shutdown, even in the unlikely event of fire extension to the adjacent areas.

This SLC is part of the Catawba Fire Protection Program and therefore subject to the provisions of the Catawba Renewed Facility Operating Licenses Conditions

REFERENCES

1. Catawba UFSAR, Section 9.5.1.
2. Catawba Nuclear Station 10 CFR 50.48(c) Fire Protection Safety Evaluation (SE).
3. Catawba Plant Design Basis Specification for Fire Protection, CNS-1465.00-00-0006, as revised.
4. Catawba Renewed Facility Operating License Conditions 2.C.(5).
5. RIS 2005-07, Compensatory Measures to Satisfy the Fire Protection Program Requirements.

16.9 AUXILIARY SYSTEMS

16.9-26 Commitments Associated With Movement of Non-Recently Irradiated Fuel Assemblies

COMMITMENT

The following requirements governing containment closure and fuel building closure shall be maintained to ensure defense-in-depth:

---NOTE---
If containment closure or fuel building closure, as applicable, is hampered by an outage activity or by a loss of power, suitable contingency actions providing an equivalent level of protection shall be developed.
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a. The equipment necessary to implement containment closure shall be appropriately staged prior to maintaining any containment penetration open, including containment air lock doors. For the purposes of this SLC, the containment equipment hatch is not considered a penetration that may be maintained open; therefore, it shall be maintained closed.

b. Hoses and cables running through an open containment penetration or containment air lock shall be configured to facilitate rapid removal in the event that containment closure is required.

c. The containment air lock may be open provided the following conditions exist:

   1. One door in each air lock shall be capable of being closed.
   2. Hoses and cables running through the air lock shall employ a means to allow safe, quick disconnection or severance.
   3. The air lock door shall not be blocked in such a way that it cannot be immediately closed. Protective covers used to protect the air lock seals/doors or devices used to keep the air lock door open/supported do not violate this provision.
   4. Personnel shall be available with the responsibility for immediate closure of at least one air lock door or for closure of an appropriate temporary door following containment evacuation.

d. Other containment penetrations may be open provided the following conditions exist:

(continued)

COMMITMENT (continued)

1. One valve in each open penetration shall be capable of being closed, or other methods to close the penetration(s) (i.e., restrict air flow out of containment), such as a closure cover, shall be fabricated and available, along with the necessary installation tools.
2. Personnel shall be available with the responsibility for immediate closure of the penetration opening(s) following a fuel handling accident inside containment.

e. Fuel Building Closure and Ventilation Requirements

1. The Fuel Handling Ventilation Exhaust System (FHVES) shall be AVAILABLE to provide a filtered release flow path for a postulated Fuel Handling Accident (FHA) whenever non-recently irradiated fuel handling activities are in progress.

2. The requirements in SLC 16.7-10, Radiation Monitoring for Plant Operations, for the FHVES radiation monitor EMF-42 are applicable.

3. When a Fuel Building door is maintained OPEN, one FHVES train shall be AVAILABLE AND in operation. If the FHVES flow is not aligned through the filter train, radiation monitor EMF-42 is required to be functional.

4. Designated personnel shall be available with the responsibility for promptly closing any open fuel building doors (personnel access or fuel receiving/railroad bay doors) should a FHA occur. If closure would be hampered by an outage/maintenance activity, suitable contingency actions will be developed.

5. When the Fuel Building doors are maintained CLOSED, except for normal entry and exit, one FHVES train is required to be AVAILABLE. The FHVES is not *required* to be in operation provided radiation monitor EMF-42 is functional.

APPLICABILITY: During movement of non-recently irradiated fuel assemblies in containment or in the fuel building, as applicable.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. COMMITMENT not met.	A.1 Suspend movement of non-recently irradiated fuel assemblies in containment or in the fuel building, as applicable.	Immediately
	AND	
	A.2 Restore compliance with COMMITMENT.	Prior to resuming movement of non-recently irradiated fuel assemblies in containment or in the fuel building, as applicable

TESTING REQUIREMENTS None

BASES

This SLC formally documents and implements a portion of the regulatory commitments made in Reference 1, approved by the NRC via Reference 2, and subsequently modified via Reference 3 and Reference 4. These commitments were made as a means of providing additional defense-in-depth during movement of non-recently irradiated fuel assemblies in containment or in the fuel building. The commitments concerning non-recently irradiated fuel movement with an open containment equipment hatch have not been implemented in this SLC. Until such time that a methodology is developed for complying with the commitments concerning non-recently irradiated fuel movement with an open containment equipment hatch, the hatch shall be maintained closed.

In the revised analysis utilizing the Alternative Source Term for a fuel handling accident in containment, no credit is taken for containment integrity with respect to containment equipment hatch closure, containment air lock closure, containment penetration closure, or filtration by the CPES prior to release. For a postulated fuel handling accident within the fuel building, both the fuel handling accident and the weir gate drop accident analyses were revised. In the revised analyses, no credit is taken for filtration of the release by the FHVES.

The revised analyses confirmed that sufficient radioactive decay has occurred after 72 hours that certain existing controls are no longer necessary to ensure that the consequences of a postulated fuel handling accident remain within regulatory limits. The analysis demonstrated that a decay period of 19.5 days was sufficient to ensure that the consequences of a weir gate drop accident remain within regulatory limits.

Although not required by the accident analysis, these administrative controls were committed to be put in place during the movement of non-recently irradiated fuel. If movement of irradiated fuel assemblies were to occur prior to the 72-hour decay period, the currently existing Technical Specification controls would apply.

Administrative controls exist such that the weir gate will not be moved over spent fuel assemblies decayed less than 19.5 days.

The FHVES will be Available during fuel handling activities involving movement of non-recently irradiated fuel assemblies or loads over the spent fuel pool to provide a filtered release flow path if a fuel handling accident occurs. Fuel handling operations when the FHVES is Unavailable and fuel building openings to the outside are closed, has not been implemented in this SLC.

For the purpose of this SLC, the definition of availability for the FHVES is being capable of performing its design basis functions as defined by the Technical Specification 3.7.13, "Fuel Handling Ventilation Exhaust System (FHVES)" surveillance tests.

For the purpose of this SLC, the definition of an open containment penetration is a penetration that provides direct access from the containment atmosphere to the outside environment.

REFERENCES

1. Letters from G.R. Peterson, Duke, to NRC, "Proposed Amendment for Partial Scope Implementation of the Alternate Source Term and Proposed Amendment to Technical Specifications (TS) 3.7.10, Control Room Area Ventilation System, TS 3.7.11, Control Room Area Chilled Water System, TS 3.7.13, Fuel Handling Ventilation Exhaust System, and TS 3.9.3, Containment Penetrations", dated December 20, 2001, February 14, 2002, and March 26, 2002.

2. Letter from NRC to G.R. Peterson, Duke, License Amendments 198 and 191 for Units 1 and 2, respectively, dated April 23, 2002.

3. Letter from J.R. Morris, Duke, to NRC, "Request to Amend Technical Specification (TS) 3.7.10: Control Room Area Ventilation System", dated September 3, 2009.

4. Letter from NRC to J.R. Morris, Duke, License Amendments 260 and 255 for Units 1 and 2, respectively, dated August 9, 2010.

16.13 CONDUCT OF OPERATIONS

16.13-4 Minimum Station Staffing Requirements

COMMITMENT Minimum station staffing shall be as indicated in Table 16.13-4-1.

APPLICABILITY: According to Table 16.13-4-1.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Minimum station staffing requirements not met.	A.1 Initiate action to fill required positions. AND A.2 Restore minimum station staffing levels.	Immediately 2 hours

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.13-4-1 Verify station staffing levels.	12 hours

Table 16.13-4-1

Minimum Station Staffing Requirements

POSITION	BOTH UNITS IN MODES 1-4	ONE UNIT IN MODES 1-4	BOTH UNITS IN MODES 5, 6, OR NO MODE
Operations Shift Manager (OSM)	1	1	1
Shift Technical Advisor (STA)	1	1	1
Senior Reactor Operator (SRO)(1)(2)(3)	2	2	1
Reactor Operator (RO)(1)(4)	3	3	2
Non-Licensed Operator (NLO)(1)	5	5	4
Chemistry Technician	1	1	1
Radiation Protection Qualified Individual (Note 5)	2	2	2
Dose Assessment Qualified Individual	1	1	1
Mechanical Maintenance Technician	1	1	1
Instrumentation and Electrical Technician	2	2	2
Medical Emergency Response Team (MERT)	2	2	2
Security Personnel	Per Security Plan	Per Security Plan	Per Security Plan

(1) Either a SRO (active or inactive), RO, or other designated personnel (NLO) may be designated as the fire brigade leader. The totals for the appropriate position shall be increased by one, depending upon which position is being used to fulfill the role of fire brigade leader.

(2) In addition to these requirements, during CORE ALTERATIONS (including fuel loading or transfer), a SRO or SRO limited to fuel handling shall be present to directly supervise the activity. During this time, no other duties shall be assigned to this person.

(3) With any unit in MODES 1-4, a SRO shall be present in the control room at all times.

(4) For each fueled unit, a RO shall be present at the controls at all times.

(5) One RP Qualified Individual must meet or exceed RP Technician minimum ANSI qualifications per Technical Specification 5.2.2.d.

BASES

The requirements of this SLC consolidate Catawba station staffing requirements into one document. This SLC includes the unit staff requirements of the Catawba Facility Operating Licenses, Technical Specification (TS) 5.2.2, 10 CFR 50.54(m), applicable Operations Management Procedures (OMPs), Administrative Directive (AD) AD-TQ-ALL-0086, "Fire Brigade Training," the Catawba Fire Protection Program, the Catawba Emergency Plan, and SLC 16.13-1, "Fire Brigade." The total requirement for each position was obtained by summing the various individual requirements for that position. The bases for the numbers in the first column of Table 16.13-4-1 are as follows:

1 OSM (active SRO) – Required by 10 CFR 50.54(m)(2)(ii) and implemented via OMP.

1 STA (active or inactive SRO) – Required by TS 5.2.2g, which indicates that the individual fulfilling the STA position is the Shift Work Manager, and implemented via OMP. Note that old TS (pre-Improved TS) Table 6.2-1, which implemented the requirements of NUREG-0737, "Clarification of TMI Action Plan Requirements," did not require an STA on shift when both units were in MODE 5, 6, or defueled. Table 16.13-4-1 is more restrictive in that it requires an STA on shift at all times.

2 SROs (active SRO) – Required by 10 CFR 50.54(m)(2)(i). Per TS 5.2.2b and 10 CFR 50.54(m)(2)(iii), at least 1 SRO must be in the control room.

3 ROs – Required by 10 CFR 50.54(m)(2)(i).

3 NLOs – Required by TS 5.2.2a and Section B, Table B-1a of the Emergency Plan and implemented via OMP.

2 NLOs – Required by the Fire Protection Program and implemented via AD and OMP.

Fire Brigade Leader – Required by the Catawba Facility Operating Licenses and Fire Protection Program and implemented via AD and OMP. The individual fulfilling this position shall be a SRO, RO, or other designated personnel (NLO) who is qualified to be a fire brigade leader. This individual functions as the fire brigade leader and is not available for other activities when directing the fire brigade. No regulations explicitly specify that the fire brigade leader be a SRO or RO. However, the fire brigade leader shall have sufficient training in or knowledge of plant safety related systems to understand the effects of a fire and fire suppression systems on safe shutdown capability.

1 Chemistry Technician (ERO) – Required by Section B, Table B-1a of the Emergency Plan. Any technician who is qualified may be credited towards fulfilling the ERO requirement.

BASES (continued)

2 Radiation Protection Qualified Individuals (ERO) - Required by Section B, Table B-1a of the Emergency Plan. 1 Radiation Protection Technician is required by TS 5.2.2d and may be counted towards fulfilling the ERO requirement. Any RP Qualified Individual may be credited towards fulfilling the ERO requirement. In the event of a fire, the RP Qualified Individual will respond to the fire for radiological monitoring purposes until directed otherwise.

1 Dose Assessment Qualified Individual (ERO) - Required by Section B, Table B-1a of the Emergency Plan. This position is designated to perform Off-Site Dose Assessment during the first 75 minutes following Emergency Class Declaration with no collateral duties. Any Dose Assessment Qualified Individual may be credited towards fulfilling the ERO requirement.

1 Mechanical Maintenance Technician (ERO) – Required by Section B, Table B-1a of the Emergency Plan. Any technician who is fire brigade qualified may be credited towards fulfilling the ERO requirement and the fire brigade requirement. In the event of a fire, the technician will respond to the fire until directed otherwise.

2 Instrumentation and Electrical Technicians (ERO) – Required by Section B, Table B-1a of the Emergency Plan. Any technician who is fire brigade qualified may be credited towards fulfilling the ERO requirement and the fire brigade requirement. In the event of a fire, the technician will respond to the fire until directed otherwise.

2 MERT (ERO) – Required by Section B, Table B-1a of the Emergency Plan. Any technician who is qualified may be credited towards fulfilling the ERO requirement. In the event of a fire, the technician will respond to the fire for security purposes until directed otherwise.

Minimum station staffing totals for the SRO and RO positions in Table 16.13-4-1 are a function of the number of units in MODES 1-4. The totals for the remaining positions in Table 16.13-4-1 are not a function of the operational MODES of the units.

10 CFR 50.54(m)(2)(i) requires 2 SROs when both units are in MODES 1-4, 2 SROs when one unit is in MODES 1-4, and 1 SRO when no units are in MODES 1-4.

10 CFR 50.54(m)(2)(i) requires 3 ROs when both units are in MODES 1-4, 3 ROs when one unit is in MODES 1-4, and 2 ROs when no units are in MODES 1-4.

BASES (continued)

The primary purpose of the Fire Protection Program is to minimize both the probability and consequence of postulated fires. Despite designed active and passive fire protection systems installed throughout the plant, a properly trained and equipped fire brigade organization of at least 5 members is required to provide immediate response to fires that may occur at the site. The fire brigade requirement is met by using personnel from Operations and

SPOC. 3 personnel from Operations are required (including the fire brigade leader) and the other 2 personnel are from SPOC.

The 2-hour REMEDIAL ACTION for restoring minimum station staffing levels is consistent with TS 5.2.2c and 5.2.2d, which allow 2 hours to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

REFERENCES

1. Catawba Facility Operating Licenses for Units 1 and 2, NPF-35 and NPF-52.
2. Catawba TS 5.2.2.
3. 10 CFR 50.54(m).
4. OMP 1-10, "Shift Manning and Overtime Requirements."
5. AD-TQ-ALL-0086, "Fire Brigade Training."
6. CNS-1465.00-00-0006, "Plant Design Basis Specification for Fire Protection."
7. Catawba Emergency Plan.
8. SLC 16.13-1, "Fire Brigade."
9. Catawba Nuclear Station 10 CFR 50.48(c) Fire Protection Safety Evaluation (SE).
10. John Stang, Senior Project Manager Office of Nuclear Reactor Regulation, letter to Mr. Michael R. Glover dated July 29, 2011, Emergency Plan Safety Evaluation. (ML11158A209).