APS
Palo Verde Nuclear Generating Station

Thomas N. Weber
Department Leader
Regulatory Affairs

Tel. 623-393-5764
Fax 623-393-5442

Mail Station 7636
PO Box 52034
Phoenix, Arizona 85072-2034

ID#: 102-06229-TNW/RAS/CJS
July 21, 2010

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

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2, and 3
Docket Nos. STN 50-528/529/530
Technical Specifications Bases Revision 52 Update

Pursuant to PVNGS Technical Specification (TS) 5.5.14, "Technical Specifications Bases Control Program," Arizona Public Service Company (APS) is submitting changes to the TS Bases incorporated into Revision 52, implemented on July 16, 2010.

The revision insertion instructions and replacement pages are provided in the enclosure.

No commitments are being made to the NRC by this letter. Should you need further information regarding this submittal, please contact Russell A. Stroud, Licensing Section Leader, at (623) 393-5111.

Sincerely,

Thomas N. Weber

TNW/RAS/CJS/gat

Enclosure: PVNGS Technical Specification Bases Revision 52 Insertion Instructions and Replacement Pages

cc:	E. E. Collins Jr.	NRC Region IV Regional Administrator (enclosure)
	J. R. Hall	NRC NRR Senior Project Manager (enclosure)
	L. K. Gibson	NRC NRR Project Manager (enclosure)
	R. I. Treadway	NRC Senior Resident Inspector for PVNGS (enclosure)

A001
NRR

ENCLOSURE

PVNGS
Technical Specification Bases
Revision 52

Insertion Instructions and
Replacement Pages

Stephenson, Carl J(Z05778)

Digitally signed by Stephenson, Carl J(Z05778)
DN: cn=Stephenson, Carl J(Z05778)
Reason: This is an accurate copy of the original document.
Date: 2010.07.13 14:47:30 -07'00'

Insertion Instructions for the Technical Specifications Bases Revision 52

REMOVE PAGES	**INSERT PAGES**
Cover page	Cover page
List of Effective Pages 1/2 through 7/8	List of Effective Pages 1/2 through 7/8
B 3.1.5-1 / B 3.1.5-2 through B 3.1.5-11 / Blank	B 3.1.5-1 / B 3.1.5-2 through B 3.1.5-11 / B 3.1.5-12
B 3.1.8-1 / B 3.1.8-2 through B 3.1.8-5 / Blank	B 3.1.8-1 / B 3.1.8-2 through B 3.1.8-5 / Blank
B 3.2.2-1 / B 3.2.2-2 B 3.2.2-3 / B 3.2.2-4	B 3.2.2-1 / B 3.2.2-2 B 3.2.2-3 / B 3.2.2-4
B 3.2.3-1 / B 3.2.3-2 B 3.2.3-3 / B 3.2.3-4	B 3.2.3-1 / B 3.2.3-2 B 3.2.3-3 / B 3.2.3-4
B 3.2.4-1 / B 3.2.4-2 B 3.2.4-3 / B 3.2.4-4	B 3.2.4-1 / B 3.2.4-2 B 3.2.4-3 / B 3.2.4-4
B 3.2.5-1 / B 3.2.5-2 B 3.2.5-3 / B 3.2.5-4 B 3.2.5-5 / B 3.2.5-6	B 3.2.5-1 / B 3.2.5-2 B 3.2.5-3 / B 3.2.5-4 B 3.2.5-5 / B 3.2.5-6
B 3.3.1-21 / B 3.3.1-22 B 3.3.1-23 / B 3.3.1-24	B 3.3.1-21 / B 3.3.1-22 B 3.3.1-23 / B 3.3.1-24
B 3.3.5-23 / B 3.3.5-24	B 3.3.5-23 / B 3.3.5-24
B 3.4.3-1 / B 3.4.3-2 through B 3.4.3-7 / B 3.4.3-8	B 3.4.3-1 / B 3.4.3-2 through B 3.4.3-7 / B 3.4.3-8
B 3.4.6-3 / B 3.4.6-4 B 3.4.6-5 / Blank	B 3.4.6-3 / B 3.4.6-4 B 3.4.6-5 / Blank
B 3.4.7-3 / B 3.4.7-4 B 3.4.7-7 / Blank	B 3.4.7-3 / B 3.4.7-4 B 3.4.7-7 / Blank

B 3.4.11-3 / B 3.4.11-4	B 3.4.11-3 / B 3.4.11-4
B 3.4.13-3 / B 3.4.13-4 through B 3.4.13-7 / B 3.4.13-8 B 3.4.13-11 / Blank	B 3.4.13-3 / B 3.4.13-4 through B 3.4.13-7 / B 3.4.13-8 B 3.4.13-11 / Blank
B 3.7.12-3 / B 3.7.12-4	B 3.7.12-3 / B 3.7.12-4
B 3.7.17-1 / B 3.7.17-2 B 3.7.17-5 / B 3.7.17-6	B 3.7.17-1 / B 3.7.17-2 B 3.7.17-5 / B 3.7.17-6

PVNGS

Palo Verde Nuclear Generating Station
Units 1, 2, and 3

Technical Specification Bases

Revision 52
July 16, 2010

PALO VERDE

Stephenson, Carl J(Z05778)

Digitally signed by Stephenson, Carl J (Z05778)
DN: cn=Stephenson, Carl J(Z05778)
Reason: This is an accurate copy of the original document.
Date: 2010.07.13 14:20:37 -07'00'

TECHNICAL SPECIFICATION BASES
LIST OF EFFECTIVE PAGES

Page No.	Rev. No.	Page No.	Rev No.
B 2.1.1-1	0	B 3.1.4-2	31
B 2.1.1-2	0	B 3.1.4-3	0
B 2.1.1-3	37	B 3.1.4-4	0
B 2.1.1-4	21	B 3.1.4-5	0
B 2.1.1-5	23	B 3.1.5-1	0
B 2.1.2-1	0	B 3.1.5-2	52
B 2.1.2-2	31	B 3.1.5-3	52
B 2.1.2-3	0	B 3.1.5-4	52
B 2.1.2-4	23	B 3.1.5-5	52
B 2.1.2-5	0	B 3.1.5-6	52
B 3.0-1	49	B 3.1.5-7	52
B 3.0-2	0	B 3.1.5-8	52
B 3.0-3	0	B 3.1.5-9	52
B 3.0-4	0	B 3.1.5-10	52
B 3.0-5	42	B 3.1.5-11	52
B 3.0-6	48	B 3.1.5-12	52
B 3.0-7	48	B 3.1.6-1	0
B 3.0-8	42	B 3.1.6-2	46
B 3.0-9	42	B 3.1.6-3	42
B 3.0-10	42	B 3.1.6-4	42
B 3.0-11	42	B 3.1.6-5	46
B 3.0-12	42	B 3.1.6-6	46
B 3.0-13	42	B 3.1.7-1	0
B 3.0-14	49	B 3.1.7-2	0
B 3.0-15	50	B 3.1.7-3	50
B 3.0-16	50	B 3.1.7-4	48
B 3.0-17	50	B 3.1.7-5	25
B 3.0-18	49	B 3.1.7-6	0
B 3.0-19	49	B 3.1.7-7	0
B 3.0-20	49	B 3.1.7-8	51
B 3.0-21	49	B 3.1.7-9	0
B 3.0-22	49	B 3.1.8-1	52
B 3.1.1-1	28	B 3.1.8-2	52
B 3.1.1-2	0	B 3.1.8-3	52
B 3.1.1-3	43	B 3.1.8-4	52
B 3.1.1-4	43	B 3.1.8-5	52
B 3.1.1-5	27	B 3.1.9-1	0
B 3.1.1-6	31	B 3.1.9-2	0
B 3.1.2-1	28	B 3.1.9-3	0
B 3.1.2-2	0	B 3.1.9-4	0
B 3.1.2-3	43	B 3.1.9-5	47
B 3.1.2-4	28	B 3.1.9-6	1
B 3.1.2-5	0	B 3.1.10-1	0
B 3.1.2-6	43	B 3.1.10-2	28
B 3.1.2-7	12	B 3.1.10-3	0
B 3.1.2-8	47	B 3.1.10-4	37
B 3.1.2-9	0	B 3.1.10-5	37
B 3.1.3-1	0	B 3.1.10-6	0
B 3.1.3-2	0	B 3.1.11-1	0
B 3.1.3-3	0	B 3.1.11-2	28
B 3.1.3-4	0	B 3.1.11-3	0
B 3.1.3-5	0	B 3.1.11-4	34
B 3.1.3-6	0	B 3.1.11-5	0
B 3.1.4-1	0	B 3.2.1-1	50

Page No.	Rev. No.	Page No.	Rev No.
B 3.2.1-2	10	B 3.3.1-14	35
B 3.2.1-3	28	B 3.3.1-15	35
B 3.2.1-4	0	B 3.3.1-16	35
B 3.2.1-5	0	B 3.3.1-17	35
B 3.2.1-6	0	B 3.3.1-18	35
B 3.2.1-7	0	B 3.3.1-19	35
B 3.2.1-8	0	B 3.3.1-20	35
B 3.2.2-1	52	B 3.3.1-21	35
B 3.2.2-2	10	B 3.3.1-22	52
B 3.2.2-3	0	B 3.3.1-23	52
B 3.2.2-4	52	B 3.3.1-24	35
B 3.2.2-5	1	B 3.3.1-25	35
B 3.2.2-6	0	B 3.3.1-26	35
B 3.2.2-7	0	B 3.3.1-27	35
B 3.2.3-1	52	B 3.3.1-28	35 Corrected
B 3.2.3-2	10	B 3.3.1-29	35
B 3.2.3-3	0	B 3.3.1-30	35 Corrected
B 3.2.3-4	52	B 3.3.1-31	35 Corrected
B 3.2.3-5	0	B 3.3.1-32	35 Corrected
B 3.2.3-6	0	B 3.3.1-33	35
B 3.2.3-7	0	B 3.3.1-34	35
B 3.2.3-8	0	B 3.3.1-35	35
B 3.2.3-9	0	B 3.3.1-36	35
B 3.2.3-10	0	B 3.3.1-37	35
B 3.2.4-1	52	B 3.3.1-38	35
B 3.2.4-2	10	B 3.3.1-39	35
B 3.2.4-3	0	B 3.3.1-40	35
B 3.2.4-4	52	B 3.3.1-41	35
B 3.2.4-5	25	B 3.3.1-42	35
B 3.2.4-6	25	B 3.3.1-43	35
B 3.2.4-7	27	B 3.3.1-44	35
B 3.2.4-8	48	B 3.3.1-45	38
B 3.2.4-9	48	B 3.3.1-46	42
B 3.2.4-10	31	B 3.3.1-47	35
B 3.2.5-1	52	B 3.3.1-48	42
B 3.2.5-2	10	B 3.3.1-49	35
B 3.2.5-3	0	B 3.3.1-50	51
B 3.2.5-4	52	B 3.3.1-51	51
B 3.2.5-5	0	B 3.3.1-52	35
B 3.2.5-6	52	B 3.3.1-53	35
B 3.2.5-7	0	B 3.3.1-54	35
B 3.3.1-1	35	B 3.3.1-55	35
B 3.3.1-2	25	B 3.3.1-56	35
B 3.3.1-3	25	B 3.3.1-57	35
B 3.3.1-4	25	B 3.3.1-58	35
B 3.3.1-5	25	B 3.3.1-59	35
B 3.3.1-6	27	B 3.3.1-60	35
B 3.3.1-7	25	B 3.3.2-1	50
B 3.3.1-8	25	B 3.3.2-2	0
B 3.3.1-9	34	B 3.3.2-3	1
B 3.3.1-10	35	B 3.3.2-4	35
B 3.3.1-11	35	B 3.3.2-5	35
B 3.3.1-12	35	B 3.3.2-6	51
B 3.3.1-13	35	B 3.3.2-7	35

TECHNICAL SPECIFICATION BASES
LIST OF EFFECTIVE PAGES

Page No.	Rev. No.	Page No.	Rev No.
B 3.3.2-8	35	B 3.3.5-8	31
B 3.3.2-9	50	B 3.3.5-9	0
B 3.3.2-10	38	B 3.3.5-10	0
B 3.3.2-11	42	B 3.3.5-11	0
B 3.3.2-12	42	B 3.3.5-12	1
B 3.3.2-13	51	B 3.3.5-13	0
B 3.3.2-14	51	B 3.3.5-14	0
B 3.3.2-15	35	B 3.3.5-15	35
B 3.3.2-16	35	B 3.3.5-16	51
B 3.3.2-17	35	B 3.3.5-17	35
B 3.3.2-18	35	B 3.3.5-18	35
B 3.3.3-1	25	B 3.3.5-19	35
B 3.3.3-2	27	B 3.3.5-20	35
B 3.3.3-3	25	B 3.3.5-21	35
B 3.3.3-4	25	B 3.3.5-22	35
B 3.3.3-5	25	B 3.3.5-23	52
B 3.3.3-6	25	B 3.3.5-24	38
B 3.3.3-7	27	B 3.3.5-25	42
B 3.3.3-8	27	B 3.3.5-26	51
B 3.3.3-9	27	B 3.3.5-27	35
B 3.3.3-10	46	B 3.3.5-28	35
B 3.3.3-11	25	B 3.3.5-29	35
B 3.3.3-12	25	B 3.3.5-30	35
B 3.3.3-13	25	B 3.3.6-1	0
B 3.3.3-14	46	B 3.3.6-2	0
B 3.3.3-15	27	B 3.3.6-3	0
B 3.3.3-16	51	B 3.3.6-4	0
B 3.3.3-17	51	B 3.3.6-5	31
B 3.3.3-18	51	B 3.3.6-6	0
B 3.3.3-19	51	B 3.3.6-7	27
B 3.3.3-20	51	B 3.3.6-8	27
B 3.3.3-21	27	B 3.3.6-9	0
B 3.3.4-1	0	B 3.3.6-10	0
B 3.3.4-2	0	B 3.3.6-11	0
B 3.3.4-3	0	B 3.3.6-12	0
B 3.3.4-4	0	B 3.3.6-13	0
B 3.3.4-5	0	B 3.3.6-14	0
B 3.3.4-6	31	B 3.3.6-15	0
B 3.3.4-7	0	B 3.3.6-16	0
B 3.3.4-8	0	B 3.3.6-17	27
B 3.3.4-9	0	B 3.3.6-18	0
B 3.3.4-10	0	B 3.3.6-19	0
B 3.3.4-11	0	B 3.3.6-20	0
B 3.3.4-12	0	B 3.3.6-21	1
B 3.3.4-13	0	B 3.3.6-22	46
B 3.3.4-14	0	B 3.3.7-1	2
B 3.3.4-15	0	B 3.3.7-2	2
B 3.3.5-1	0	B 3.3.7-3	0
B 3.3.5-2	0	B 3.3.7-4	0
B 3.3.5-3	0	B 3.3.7-5	0
B 3.3.5-4	35	B 3.3.7-6	42
B 3.3.5-5	0	B 3.3.7-7	0
B 3.3.5-6	0	B 3.3.7-8	51
B 3.3.5-7	0	B 3.3.7-9	51

TECHNICAL SPECIFICATION BASES
LIST OF EFFECTIVE PAGES

Page No.	Rev. No.	Page No.	Rev No.
B 3.3.8-1	0	B 3.4.1-5	0
B 3.3.8-2	44	B 3.4.2-1	7
B 3.3.8-3	0	B 3.4.2-2	1
B 3.3.8-4	0	B 3.4.3-1	52
B 3.3.8-5	0	B 3.4.3-2	52
B 3.3.8-6	51	B 3.4.3-3	0
B 3.3.8-7	0	B 3.4.3-4	52
B 3.3.8-8	44	B 3.4.3-5	52
B 3.3.9-1	48	B 3.4.3-6	0
B 3.3.9-2	48	B 3.4.3-7	52
B 3.3.9-3	21	B 3.4.3-8	52
B 3.3.9-4	10	B 3.4.4-1	0
B 3.3.9-5	51	B 3.4.4-2	50
B 3.3.9-6	0	B 3.4.4-3	7
B 3.3.9-7	0	B 3.4.4-4	0
B 3.3.10-1	0	B 3.4.5-1	0
B 3.3.10-2	0	B 3.4.5-2	38
B 3.3.10-3	0	B 3.4.5-3	38
B 3.3.10-4	0	B 3.4.5-4	0
B 3.3.10-5	18	B 3.4.5-5	6
B 3.3.10-6	0	B 3.4.6-1	0
B 3.3.10-7	0	B 3.4.6-2	6
B 3.3.10-8	14	B 3.4.6-3	52
B 3.3.10-9	14	B 3.4.6-4	6
B 3.3.10-10	51	B 3.4.6-5	52
B 3.3.10-11	50	B 3.4.7-1	0
B 3.3.10-12	50	B 3.4.7-2	6
B 3.3.10-13	50	B 3.4.7-3	52
B 3.3.10-14	50	B 3.4.7-4	38
B 3.3.10-15	50	B 3.4.7-5	0
B 3.3.10-16	50	B 3.4.7-6	0
B 3.3.10-17	50	B 3.4.7-7	52
B 3.3.10-18	50	B 3.4.8-1	0
B 3.3.10-19	51	B 3.4.8-2	6
B 3.3.10-20	50	B 3.4.8-3	6
B 3.3.10-21	50	B 3.4.9-1	41
B 3.3.10-22	32	B 3.4.9-2	31
B 3.3.11-1	0	B 3.4.9-3	41
B 3.3.11-2	2	B 3.4.9-4	41
B 3.3.11-3	2	B 3.4.9-5	0
B 3.3.11-4	42	B 3.4.9-6	0
B 3.3.11-5	42	B 3.4.10-1	50
B 3.3.11-6	51	B 3.4.10-2	7
B 3.3.11-7	50	B 3.4.10-3	0
B 3.3.12-1	15	B 3.4.10-4	0
B 3.3.12-2	50	B 3.4.11-1	0
B 3.3.12-3	37	B 3.4.11-2	7
B 3.3.12-4	37	B 3.4.11-3	0
B 3.3.12-5	51	B 3.4.11-4	52
B 3.3.12-6	6	B 3.4.11-5	0
B 3.4.1-1	10	B 3.4.11-6	0
B 3.4.1-2	28	B 3.4.12-1	1
B 3.4.1-3	0	B 3.4.12-2	34
B 3.4.1-4	0	B 3.4.12-3	48

TECHNICAL SPECIFICATION BASES
LIST OF EFFECTIVE PAGES

Page No.	Rev. No.	Page No.	Rev No.
B 3.4.12-4	0	B 3.5.1-7	1
B 3.4.12-5	31	B 3.5.1-8	1
B 3.4.13-1	0	B 3.5.1-9	0
B 3.4.13-2	0	B 3.5.1-10	35
B 3.4.13-3	1	B 3.5.2-1	0
B 3.4.13-4	52	B 3.5.2-2	35
B 3.4.13-5	52	B 3.5.2-3	0
B 3.4.13-6	0	B 3.5.2-4	0
B 3.4.13-7	52	B 3.5.2-5	0
B 3.4.13-8	52	B 3.5.2-6	0
B 3.4.13-9	42	B 3.5.2-7	1
B 3.4.13-10	42	B 3.5.2-8	22
B 3.4.13-11	52	B 3.5.2-9	1
B 3.4.14-1	0	B 3.5.2-10	35
B 3.4.14-2	34	B 3.5.3-1	0
B 3.4.14-3	34	B 3.5.3-2	48
B 3.4.14-4	38	B 3.5.3-3	0
B 3.4.14-5	38	B 3.5.3-4	0
B 3.4.14-6	38	B 3.5.3-5	0
B 3.4.14-7	38	B 3.5.3-6	2
B.3.4.14-8	38	B 3.5.3-7	2
B 3.4.15-1	0	B 3.5.3-8	1
B 3.4.15-2	48	B 3.5.3-9	0
B 3.4.15-3	0	B 3.5.3-10	2
B 3.4.15-4	0	B 3.5.4-1	15
B 3.4.15-5	0	B 3.5.4-2	0
B 3.4.15-6	35	B 3.5.4-3	42
B 3.4.15-7	35	B 3.5.5-1	51
B 3.4.16-1	2	B 3.5.5-2	51
B 3.4.16-2	10	B 3.5.5-3	51
B 3.4.16-3	0	B 3.5.5-4	51
B 3.4.16-4	42	B 3.5.5-5	51
B 3.4.16-5	0	B 3.5.5-6	51
B 3.4.16-6	0	B 3.5.5-7	51
B 3.4.17-1	0	B 3.5.5-8	51
B 3.4.17-2	27	B 3.5.5-9	51
B 3.4.17-3	42	B 3.5.5-10	48
B 3.4.17-4	42	B 3.5.5-11	48
B 3.4.17-5	0	B 3.5.5-12	48
B 3.4.17-6	0	B 3.5.5-13	48
B 3.4.18-1	38	B 3.5.5-14	48
B 3.4.18-2	40	B 3.5.5-15	48
B 3.4.18-3	38	B 3.5.5-16	48
B 3.4.18-4	38	B 3.5.6-1	0
B 3.4.18-5	38	B 3.5.6-2	1
B 3.4.18-6	38	B 3.5.6-3	0
B 3.4.18-7	38	B 3.5.6-4	24
B 3.4.18-8	38	B 3.5.6-5	50
B 3.5.1-1	0	B 3.6.1-1	0
B 3.5.1-2	48	B 3.6.1-2	49 Corrected
B 3.5.1-3	7	B 3.6.1-3	0
B 3.5.1-4	0	B 3.6.1-4	29
B 3.5.1-5	0	B 3.6.1-5	29
B 3.5.1-6	0	B 3.6.2-1	45

Page No.	Rev. No.	Page No.	Rev No.
B 3.6.2-2	49	B 3.7.2-7	40
B 3.6.2-3	0	B 3.7.2-8	40
B 3.6.2-4	0	B 3.7.2-9	40
B 3.6.2-5	0	B 3.7.3-1	1
B 3.6.2-6	0	B 3.7.3-2	1
B 3.6.2-7	0	B 3.7.3-3	37
B 3.6.2-8	0	B 3.7.3-4	0
B 3.6.3-1	36	B 3.7.3-5	0
B 3.6.3-2	43	B 3.7.4-1	50
B 3.6.3-3	49	B 3.7.4-2	50
B 3.6.3-4	43	B 3.7.4-3	50
B 3.6.3-5	43	B 3.7.4-4	50
B 3.6.3-6	43	B 3.7.4-5	50
B 3.6.3-7	43	B 3.7.5-1	0
B 3.6.3-8	43	B 3.7.5-2	0
B 3.6.3-9	43	B 3.7.5-3	40
B 3.6.3-10	43	B 3.7.5-4	27
B 3.6.3-11	43	B 3.7.5-5	42
B 3.6.3-12	43	B 3.7.5-6	42
B 3.6.3-13	43	B 3.7.5-7	9
B 3.6.3-14	43	B 3.7.5-8	9
B 3.6.3-15	43	B 3.7.5-9	9
B 3.6.3-16	43	B 3.7.5-10	9
B 3.6.3-17	27	B 3.7.5-11	9
B 3.6.3-18	43	B 3.7.6-1	0
B 3.6.3-19	43	B 3.7.6-2	28
B 3.6.4-1	35	B 3.7.6-3	28
B 3.6.4-2	38	B 3.7.6-4	0
B 3.6.4-3	1	B 3.7.7-1	0
B 3.6.5-1	0	B 3.7.7-2	1
B 3.6.5-2	1	B 3.7.7-3	1
B 3.6.5-3	48	B 3.7.7-4	1
B 3.6.5-4	0	B 3.7.7-5	1
B 3.6.6-1	0	B 3.7.8-1	1
B 3.6.6-2	0	B 3.7.8-2	1
B 3.6.6-3	38	B 3.7.8-3	1
B 3.6.6-4	7	B 3.7.8-4	1
B 3.6.6-5	1	B 3.7.9-1	0
B 3.6.6-6	0	B 3.7.9-2	44
B 3.6.6-7	1	B 3.7.9-3	44
B 3.6.6-8	48	B 3.7.10-1	10
B 3.6.6-9	0	B 3.7.10-2	1
B 3.7.1-1	50	B 3.7.10-3	1
B 3.7.1-2	50	B 3.7.10-4	1
B 3.7.1-3	34	B 3.7.11-1	50
B 3.7.1-4	34	B 3.7.11-2	50
B 3.7.1-5	34	B 3.7.11-3	51
B 3.7.1-6	28 Corrected	B 3.7.11-4	50
B 3.7.2-1	40	B 3.7.11-5	50
B 3.7.2-2	42	B 3.7.11-6	50
B 3.7.2-3	40	B 3.7.11-7	50
B 3.7.2-4	40	B 3.7.11-8	50
B 3.7.2-5	40	B 3.7.11-9	50
B 3.7.2-6	40	B 3.7.12-1	1

TECHNICAL SPECIFICATION BASES
LIST OF EFFECTIVE PAGES

Page No.	Rev. No.	Page No.	Rev No.
B 3.7.12-2	21	B 3.8.1-32	45
B 3.7.12-3	52	B 3.8.1-33	48
B 3.7.12-4	10	B 3.8.1-34	45
B 3.7.13-1	0	B 3.8.1-35	50
B 3.7.13-2	0	B 3.8.1-36	50
B 3.7.13-3	0	B 3.8.1-37	45
B 3.7.13-4	0	B 3.8.1-38	45
B 3.7.13-5	0	B 3.8.1-39	45
B 3.7.14-1	0	B 3.8.1-40	48
B 3.7.14-2	21	B 3.8.1-41	50
B 3.7.14-3	21	B 3.8.1-42	45
B 3.7.15-1	3	B 3.8.1-43	45
B 3.7.15-2	3	B 3.8.1-44	45
B 3.7.16-1	7	B 3.8.1-45	45
B 3.7.16-2	0	B.3.8.1-46	50
B 3.7.16-3	0	B.3.8.1-47	45
B 3.7.16-4	0	B.3.8.1-48	45
B 3.7.17-1	52	B 3.8.2-1	0
B 3.7.17-2	3	B 3.8.2-2	0
B 3.7.17-3	3	B 3.8.2-3	0
B 3.7.17-4	3	B 3.8.2-4	21
B 3.7.17-5	3	B 3.8.2-5	21
B 3.7.17-6	52	B 3.8.2-6	0
B 3.8.1-1	35	B 3.8.3-1	0
B 3.8.1-2	2	B 3.8.3-2	0
B 3.8.1-3	34	B 3.8.3-3	50
B 3.8.1-4	34	B 3.8.3-4	0
B 3.8.1-5	20	B 3.8.3-5	51
B 3.8.1-6	27	B 3.8.3-6	51
B 3.8.1-7	42	B 3.8.3-7	41
B 3.8.1-8	50	B 3.8.3-8	41
B 3.8.1-9	42	B 3.8.3-9	41
B 3.8.1-10	43	B 3.8.3-10	48
B 3.8.1-11	50	B 3.8.4-1	0
B 3.8.1-12	48	B 3.8.4-2	37
B 3.8.1-13	48	B 3.8.4-3	0
B 3.8.1-14	48	B 3.8.4-4	2
B 3.8.1-15	48	B 3.8.4-5	2
B 3.8.1-16	41	B 3.8.4-6	2
B 3.8.1-17	41	B 3.8.4-7	35
B 3.8.1-18	41	B 3.8.4-8	35
B 3.8.1-19	41	B 3.8.4-9	35
B 3.8.1-20	41	B 3.8.4-10	37
B 3.8.1-21	41	B 3.8.4-11	48
B 3.8.1-22	41	B 3.8.5-1	1
B 3.8.1-23	50	B 3.8.5-2	1
B 3.8.1-24	50	B 3.8.5-3	21
B 3.8.1-25	50	B 3.8.5-4	21
B 3.8.1-26	50	B 3.8.5-5	2
B 3.8.1-27	50	B 3.8.5-6	2
B 3.8.1-28	41	B 3.8.6-1	0
B 3.8.1-29	41 Corrected	B 3.8.6-2	0
B 3.8.1-30	50	B 3.8.6-3	0
B 3.8.1-31	50	B 3.8.6-4	6

Page No.	Rev. No.	Page No.	Rev No.
B 3.8.6-5	37	B 3.9.7-2	0
B 3.8.6-6	37	B 3.9.7-3	0
B 3.8.6-7	48		
B 3.8.7-1	48		
B 3.8.7-2	48		
B 3.8.7-3	48		
B 3.8.7-4	0		
B 3.8.8-1	1		
B 3.8.8-2	1		
B 3.8.8-3	21		
B 3.8.8-4	21		
B 3.8.8-5	1		
B 3.8.9-1	51		
B 3.8.9-2	0		
B 3.8.9-3	51		
B 3.8.9-4	0		
B 3.8.9-5	0		
B 3.8.9-6	0		
B 3.8.9-7	0		
B 3.8.9-8	0		
B 3.8.9-9	0		
B 3.8.9-10	0		
B 3.8.9-11	51		
B 3.8.10-1	0		
B 3.8.10-2	21		
B 3.8.10-3	48		
B 3.8.10-4	0		
B 3.9.1-1	34 Corrected		
B 3.9.1-2	0		
B 3.9.1-3	0		
B 3.9.1-4	0		
B 3.9.2-1	48		
B 3.9.2-2	15		
B 3.9.2-3	15		
B 3.9.2-4	15		
B 3.9.3-1	18		
B 3.9.3-2	19		
B 3.9.3-3	27		
B 3.9.3-4	19		
B 3.9.3-5	19		
B 3.9.3-6	19		
B 3.9.4-1	0		
B 3.9.4-2	1		
B 3.9.4-3	0		
B 3.9.4-4	0		
B 3.9.5-1	0		
B 3.9.5-2	16		
B 3.9.5-3	27		
B 3.9.5-4	16		
B 3.9.5-5	16		
B 3.9.6-1	0		
B 3.9.6-2	0		
B 3.9.6-3	0		
B 3.9.7-1	0		

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Control Element Assembly (CEA) Alignment

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the shutdown and regulating CEAs is an initial assumption in all safety analyses that assume CEA insertion upon reactor trip. Maximum CEA misalignment is an initial assumption in the safety analyses that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10 and GDC 26 (Ref. 1) and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a CEA to become inoperable or to become misaligned from its group. CEA inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available CEA worth for reactor shutdown. Therefore, CEA alignment and operability are related to core operation in design power peaking limits and the core design requirement of a minimum SDM. If a CEA(s) is discovered to be immovable but remains trippable and aligned, the CEA is considered to be OPERABLE. At anytime, if a CEA(s) is immovable, a determination of the trippability (OPERABILITY) of that CEA(s) must be made, and appropriate action taken.

Limits on CEA alignment and operability have been established, and all CEA positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

CEAs are moved by their control element drive mechanisms (CEDMs). Each CEDM moves its CEA one step (approximately ¾ inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Control Element Drive Mechanism Control System (CEDMCS).

(continued)

BASES

BACKGROUND (continued)

The CEAs are arranged into groups that are radially symmetric. Therefore, movement of the CEAs does not introduce radial asymmetries in the core power distribution. The shutdown and regulating CEAs provide the required reactivity worth for immediate reactor shutdown upon a reactor trip. The regulating CEAs also provide reactivity (power level) control during normal operation and transients. Their movement may be automatically controlled by the Reactor Regulating System. Part strength CEAs are not credited in the safety analyses for shutting down the reactor, as are the regulating and shutdown groups. The part strength CEAs are used solely for ASI control.

The axial position of shutdown and regulating CEAs is indicated by two separate and independent systems, which are the Pulse Counting CEA Position Indication System (described in Ref. 4) and the Reed Switch CEA Position Indication System (described in Ref. 5).

The Pulse Counting CEA Position Indicating System indicates CEA position to the actual step, if each CEA moves one step for each command signal. However, if each CEA does not follow the commands, the system will incorrectly reflect the position of the affected CEA(s). This condition may affect the operability of COLSS (refer to Section 3.2, Power Distribution Limits for the applicable actions) and should be detected by the Reed Switch Position Indication System through surveillance or alarm. Although the Reed Switch Position Indication System is less precise than the Pulse Counting CEA Position Indicating System, it is not subject to the same error mechanisms.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

CEA misalignment accidents are analyzed in the safety analysis (Ref. 3). The accident analysis defines CEA misoperation as any event, with the exception of sequential group withdrawals, which could result from a single malfunction in the reactivity control systems. For example, CEA misalignment may be caused by a malfunction of the CEDM, CEDMCS, or by operator error. A stuck CEA may be caused by mechanical jamming of the CEA fingers or of the gripper.

Inadvertent withdrawal of a single CEA may be caused by an electrical failure in the CEA coil power programmers. A dropped CEA could be caused by an opening of the electrical circuit of the CEDM holding coil for a full strength, or part strength CEA.

The acceptance criteria for addressing CEA inoperability or misalignment are that:

There shall be no violations of:

1. specified acceptable fuel design limits, or

2. Reactor Coolant System (RCS) pressure boundary integrity.

To ensure that these acceptance criteria are met, the CEAs shall be capable of inserting the required negative reactivity and in the time period assumed in the accident analysis upon a reactor trip.

Three types of misalignment are distinguished. They are misalignment within deadband (< 6.6 inches), misalignment in excess of deadband, and CEA/subgroup drop. During movement of a group, one CEA may stop moving while the other CEAs in the group continue. This condition may cause excessive power peaking. This misalignment can be within or exceed the deadband. The last type of misalignment occurs when one CEA or subgroup drops partially or fully into the reactor core. This event causes an initial power reduction followed by a return towards the original power due to positive reactivity feedback from the negative moderator temperature coefficient. Increased peaking during the power increase may result in erosion of DNB margin.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Misalignments within deadband are evaluated to ensure specified acceptable fuel design limits (SAFDLs) are not exceeded. Misalignments in excess of deadband considers the case of a single CEA withdrawn approximately 10 inches from a bank inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio (DNBR) bounds the situation when a CEA is misaligned from its group by 6.6 inches.

The effect of any misoperated CEA on the core power distribution will be assessed by the CEA calculators, and an appropriately augmented power distribution penalty factor will be supplied as input to the core protection calculators (CPCs). As the reactor core responds to the reactivity changes caused by the misoperated CEA and the ensuing reactor coolant and Doppler feedback effects, the CPCs will initiate a low DNBR or high local power density trip signal if SAFDLs are approached.

The accident analysis analyzed a single four finger full and part strength CEA drop, a twelve finger drop, and a subgroup drop.

The twelve finger and subgroup drops cause larger distortions than the four finger drops. With CEACS In Service (IS), the subgroup and twelve finger rod drops will result in a penalty factor such that a CPC trip will occur if SAFDLs are approached. The four finger CEA drop is protected by the thermal margin reserved in COLSS or CPC DNBR limit lines (COLR figures 3.2.4-2 for CEACs IS and 3.2.4-3 for CEACs OOS) when COLSS is Out of Service (OOS). With CEACs OOS, CPCs will not penalize DNBR nor LPD when CEAs are misaligned; therefore, additional thermal margin is required to be preserved due to the larger radial power distortion associated with twelve finger and subgroup drops. The most rapid approach to the DNBR SAFDL or the fuel centerline melt SAFDL is caused by a single full strength CEA drop with CEACS IS and either a twelve finger or subgroup drop with CEACS OOS.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

In the case of the full strength CEA drop, a prompt decrease in core average power and a distortion in radial power are initially produced, which when conservatively coupled result in local power and heat flux increases, and a decrease in DNBR. A part strength CEA drop would cause a similar reactivity response although with less of a magnitude due to the full strength CEAs having a more significant reactivity worth.

With CEACS OOS, a twelve finger and subgroup drop will result in greater radial power distortion. To accommodate the greater distortion without a reactor trip, increased thermal margin is required to be preserved.

With CEACS IS, as the twelve finger drop is detected, core power and an appropriately augmented power distribution penalty factor are supplied to the CPCs. CPCs will trip if required to prevent SAFDLs from being exceeded. For plant operation within the DNBR and local power density (LPD) LCOs, DNBR and LPD trips can normally be avoided on a dropped 4-finger CEA since CEACs do not penalize DNBR or LPD for a four finger CEA drop.

With CEACS IS and a subgroup drop, a distortion in power distribution, and a decrease in core power are produced. As the position of the dropped CEA subgroup is detected, an appropriate power distribution penalty factor is supplied to the CPCs, and a reactor trip signal on low DNBR is generated.

CEA alignment satisfies Criteria 2 and 3 of 10 CFR 50.3(c)(2)(ii).

LCO

The limits on part strength, shutdown, and regulating CEA alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the CEAs will be available and will be inserted to provide enough negative reactivity to shut down the reactor. The OPERABILITY requirements also ensure that the CEA banks maintain the correct power distribution and CEA alignment.

(continued)

BASES (continued)

LCO (continued	The requirement is to maintain the CEA alignment to within 6.6 inches between any CEA and all other CEAs in its group. Failure to meet the requirements of this LCO may produce unacceptable power peaking factors, DNBR, and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.
APPLICABILITY	The requirements on CEA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (e.g., trippability) and alignment of CEAs have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down and not producing fission power. In the shutdown modes, the OPERABILITY of the shutdown and regulating CEAs has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.2, "SHUTDOWN MARGIN (SDM) - Reactor Trip Breakers Closed," for SDM in MODES 3, 4, and 5, and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

BASES (continued)

ACTIONS

A.1 and A.2

A CEA may become misaligned, yet remain trippable. In this condition, the CEA can still perform its required function of adding negative reactivity should a reactor trip be necessary.

If one or more CEAs (regulating, shutdown, or part strength) are misaligned by 6.6 inches and ≤ 9.9 inches but trippable, or one CEA misaligned by > 9.9 inches but trippable, continued operation in MODES 1 and 2 may continue, provided, within 1 hour, the power is reduced in accordance with the limits in the COLR, and within 2 hours CEA alignment is restored. Regulating and part strength CEA alignment can be restored by either aligning the misaligned CEA(s) to within 6.6 inches of its group or aligning the misaligned CEA's group to within 6.6 inches of the misaligned CEA(s). Shutdown CEA alignment can be restored by aligning the misaligned CEA(s) to within 6.6 inches of its group.

Xenon redistribution in the core starts to occur as soon as a CEA becomes misaligned. Reducing THERMAL POWER in accordance with the limits in the COLR ensures acceptable power distributions are maintained (Ref. 3). For small misalignments (< 9.9 inches) of the CEAs, there is:

a. A small effect on the time dependent long term power distributions relative to those used in generating LCOs and limiting safety system settings (LSSS) setpoints;

b. A negligible effect on the available SDM; and

c. A small effect on the ejected CEA worth used in the accident analysis.

With a large CEA misalignment (≥ 9.9 inches), however, this misalignment would cause distortion of the core power distribution. This distortion may, in turn, have a significant effect on the time dependent, long term power distributions relative to those used in generating LCOs and LSSS setpoints. The effect on the available SDM and the ejected CEA worth used in the accident analysis remain small.

Therefore, this condition is limited to the single CEA misalignment, while still allowing 2 hours for recovery.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

In both cases, a 2 hour time period is sufficient to:

a. Identify cause of a misaligned CEA;

b. Take appropriate corrective action to realign the CEAs; and

c. Minimize the effects of xenon redistribution.

The CEA must be returned to OPERABLE status within 2 hours. If a CEA misalignment results in the COLSS programs being declared INOPERABLE, refer to Section 3.2 Power Distribution Limits for applicable actions.

B.1 and B.2

At least two of the following three CEA position indicator channels shall be OPERABLE for each CEA:

a. CEA Reed Switch Position Transmitter (RSPT 1) with the capability of determining the absolute CEA positions within 5.2 inches,

b. CEA Reed Switch Position Transmitter (RSPT 2) with the capability of determining the absolute CEA positions within 5.2 inches, and

c. The CEA pulse counting position indicator channel.

If only one CEA position indicator channel is OPERABLE for one CEA per CEA Group, continued operation in MODES 1 and 2 may continue, provided, within 6 hours, at least two position indicator channels are returned to OPERABLE status; or within 6 hours and once per 12 hours, verify that the CEA group with the inoperable position indicators are either fully withdrawn or fully inserted while maintaining the insertion limits of LCO 3.1.6, LCO 3.1.7 and LCO 3.1.8. CEAs are fully withdrawn when the requirements of LCO 3.1.6 and 3.1.7 are met.

Additionally, the Upper Electrical Limit (UEL) CEA reed switches provide an acceptable indication of CEA position for a fully withdrawn condition.

(continued)

BASES

ACTIONS

C.1

If a Required Action or associated Completion Time of Condition A or Condition B is not met, or if one or more regulating or shutdown CEAs are untrippable (immovable as a result of excessive friction or mechanical interference or known to be untrippable), the unit is required to be brought to MODE 3. By being brought to MODE 3, the unit is brought outside its MODE of applicability.

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

If a full strength CEA is untrippable, it is not available for reactivity insertion during a reactor trip. With an untrippable CEA, meeting the insertion limits of LCO 3.1.6, "Shutdown Control Element Assembly (CEA) Insertion Limits," and LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits," does not ensure that adequate SDM exists. Therefore, the plant must be shut down in order to evaluate the SDM required boron concentration and power level for critical operation. Continued operation is allowed with untrippable part strength CEAs if the alignment and insertion limits are met.

Continued operation is not allowed with one or more full length CEAs untrippable. This is because these cases are indicative of a loss of SDM and power distribution, and a loss of safety function, respectively.

D.1

Continued operation is not allowed in the case of more than one CEA misaligned from any other CEA in its group by > 9.9 inches. For example, two CEAs in a group misaligned from any other CEA in that group by > 9.9 inches, or more than one CEA group that has at least one CEA misaligned from any other CEA in that group by > 9.9 inches. This is indicative of a loss of power distribution and a loss of safety function, respectively. Multiple CEA misalignments should result in automatic protective action. Therefore, with two or more CEAs misaligned more than 9.9 inches, this

(continued)

BASES

ACTIONS

D.1 (continued)

could result in a situation outside the design basis and immediate action would be required to prevent any potential fuel damage. Immediately opening the reactor trip breakers minimizes these effects.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that individual CEA positions are within 6.6 inches (indicated reed switch positions) of all other CEAs in the group at a 12 hour Frequency allows the operator to detect a CEA that is beginning to deviate from its expected position. The specified Frequency takes into account other CEA position information that is continuously available to the operator in the control room, so that during actual CEA motion, deviations can immediately be detected.

SR 3.1.5.2

OPERABILITY of at least two CEA position indicator channels is required to determine CEA positions, and thereby ensure compliance with the CEA alignment and insertion limits. The CEA full in and full out limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions.

SR 3.1.5.3

Verifying each full strength CEA is trippable would require that each CEA be tripped. In MODES 1 and 2 tripping each full strength CEA would result in radial or axial power tilts, or oscillations. Therefore individual full strength CEAs are exercised every 92 days to provide increased confidence that all full strength CEAs continue to be trippable, even if they are not regularly tripped. A movement of 5 inches is adequate to demonstrate motion without exceeding the alignment limit when only one full strength CEA is being moved. The 92 day Frequency takes into consideration other information available to the operator in the control room and other surveillances being performed more frequently, which add to the determination of OPERABILITY of the CEAs (Ref. 3). Between required

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.5.3 (continued)

performances of SR 3.1.5.3, if a CEA(s) is discovered to be immovable but remains trippable and aligned, the CEA is considered to be OPERABLE. At anytime, if a CEA(s) is immovable, a determination of the trippability (OPERABILITY) of that CEA(s) must be made, and appropriate action taken.

SR 3.1.5.4

Performance of a CHANNEL FUNCTIONAL TEST of each reed switch position transmitter channel ensures the channel is OPERABLE and capable of indicating CEA position. Since this test must be performed when the reactor is shut down, an 18 month Frequency to be coincident with refueling outage was selected. Operating experience has shown that these components usually pass this Surveillance when performed at a Frequency of once every 18 months. Furthermore, the Frequency takes into account other factors, which determine the OPERABILITY of the CEA Reed Switch Indication System. These factors include:

a. Other, more frequently performed surveillances that help to verify OPERABILITY;

b. On-line diagnostics performed automatically by the CPCs, CEACs, and the Plant Computer which include CEA position comparisons and sensor validation; and

c. The CHANNEL CALIBRATIONs for the CPCs (SR 3.3.1.9) and CEACs (SR 3.3.3.4) input channels that are performed at 18 month intervals and is an overlapping test.

SR 3.1.5.5

Verification of full strength CEA drop times determines that the maximum CEA drop time permitted is consistent with the assumed drop time used in the safety analysis (Ref. 3). Measuring drop times prior to reactor criticality, after reactor vessel head removal, ensures the reactor internals and CEDM will not interfere with CEA motion or drop time, and that no degradation in these systems has occurred that would adversely affect CEA motion or drop time. Individual CEAs whose drop times are greater than safety analysis assumptions are not OPERABLE. This SR is performed prior to criticality due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.5.5 (continued)

The 4 second CEA drop time is the maximum time it takes for a fully withdrawn individual full strength CEA to reach its 90% insertion position when electrical power is interrupted to the CEA drive mechanism with RCS T_{cold} greater than or equal to 550°F and all reactor coolant pumps operating.

The CEA drop time of full strength CEAs shall also be demonstrated through measurement prior to reactor criticality for specifically affected individual CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. UFSAR, Section 15.4.
4. UFSAR, Section 7.7.1.3.2.3.
5. UFSAR, Section 7.5.1.1.4.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Part Strength Control Element Assembly (CEA) Insertion Limits

BASES

BACKGROUND

The insertion limits of the part strength CEAs are initial assumptions in the safety analyses for CEA misoperation events. The insertion limits directly affect core power distributions. The applicable criteria for these power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Plants" (Ref. 2). Limits on part strength CEA insertion have been established, and all CEA positions are monitored and controlled during power operation to ensure that the power distribution defined by the design power peaking limits is preserved.

The part strength CEAs are used for axial power shape control of the reactor. The positions of the part strength CEAs are manually controlled. They are capable of changing reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that preserve the criteria specified in 10 CFR 50.46 (Ref. 2). Together, LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits"; LCO 3.1.8; LCO 3.2.4, "Departure From Nucleate Boiling Ratio (DNBR)"; and LCO 3.2.5, "AXIAL SHAPE INDEX (ASI)," provide limits on control component operation and on monitored process variables to ensure the core operates within the linear heat rate (LHR) (LCO 3.2.1, "Linear Heat Rate (LHR)"); planar peaking factor (F_{xy}) (LCO 3.2.2, "Planar Radial Peaking Factors (F_{xy})"); and LCO 3.2.4 limits in the COLR.

Operation within the limits given in the COLR prevents power peaks that would exceed the loss of coolant accident (LOCA) limits derived by the Emergency Core Cooling Systems analysis. Operation within the F_{xy} and departure from nucleate boiling (DNB) limits given in the COLR prevents DNB during a loss of forced reactor coolant flow accident.

(continued)

BASES

BACKGROUND (continued)

The establishment of limiting safety system settings and LCOs requires that the expected long and short term behavior of the radial peaking factors be determined. The long term behavior relates to the variation of the steady state radial peaking factors with core burnup; it is affected by the amount of CEA insertion assumed, the portion of a burnup cycle over which such insertion is assumed, and the expected power level variation throughout the cycle. The short term behavior relates to transient perturbations to the steady state radial peaks due to radial xenon redistribution. The magnitudes of such perturbations depend upon the expected use of the CEAs during anticipated power reductions and load maneuvering. Analyses are performed, based on the expected mode of operation of the Nuclear Steam Supply System (base loaded, maneuvering, etc.). From these analyses, CEA insertions are determined, and a consistent set of radial peaking factors are defined. The long term (steady state) and short term insertion limits are determined, based upon the assumed mode of operation used in the analyses; they provide a means of preserving the assumptions on CEA insertions used. The long and short term insertion limits of LCO 3.1.8 are specified for the plant, which has been designed primarily for base loaded operation, but has the ability to accommodate a limited amount of load maneuvering.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition I) and anticipated operational occurrences (Condition II). The regulating CEA insertion, part strength CEA insertion, ASI, and T_q LCOs preclude core power distributions from occurring that would violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed 2200°F (Ref. 2);

b. During CEA misoperation events, there must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 3); and

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

d. The CEAs must be capable of shutting down the reactor with a minimum required SDM, with the highest worth CEA stuck fully withdrawn, GDC 26 (Ref. 1).

Regulating CEA position, part strength CEA position, ASI, and T_q are process variables that together characterize and control the three dimensional power distribution of the reactor core.

Fuel cladding damage does not occur when the core is operated outside these LCOs during normal operation. However, fuel cladding damage could result, should an accident occur with simultaneous violation of one or more of these LCOs. Changes in the power distribution can cause increased power peaking and corresponding increased local LHRs.

The part strength CEA insertion limits satisfy Criterion 2 of 10 CFR 50.36 (c)(2)(ii). The part strength CEAs are required due to the potential peaking factor violations that could occur if part strength CEAs exceed insertion limits.

LCO

The limits on part strength CEA insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution.

APPLICABILITY

The part strength insertion limits shall be maintained with the reactor in MODES 1 and 2. These limits must be maintained, since they preserve the assumed power distribution. Applicability in lower MODES is not required, since the power distribution assumptions would not be exceeded in these MODES.

(continued)

BASES (continued)

ACTIONS

A.1, A.2 and B.1

If the part strength CEA groups are inserted beyond the following limits, flux patterns begin to develop that are outside the range assumed for long term fuel burnup:

1) Transient insertion limits;

2) Between the long term (steady-state) insertion limit and the transient insertion limit for:

   a) 7 or more effective full power days (EFPD) out of any 30 EFPD period;

   b) 14 EFPD or more out of any 365 EFPD period.

If allowed to continue beyond this limit, the peaking factors assumed as initial conditions in the accident analysis may be invalidated (Ref. 4). Restoring the CEAs to within limits or reducing THERMAL POWER to that fraction of RTP that is allowed by CEA group position, using the limits specified in the COLR, ensures that acceptable peaking factors are maintained.

Since these effects are cumulative, actions are provided to limit the total time the part strength CEAs can be out of limits in any 30 EFPD or 365 EFPD period. Since the cumulative out of limit times are in days, an additional Completion Time of 2 hours is reasonable for restoring the part strength CEAs to within the allowed limits.

C.1

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should commence. A Completion Time of 6 hours is reasonable, based on operating experience, for reaching Mode 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verification of each part strength CEA group position every 12 hours is sufficient to detect CEA positions that may approach the limits, and provide the operator with time to undertake the Required Action(s), should insertion limits be found to be exceeded. The 12 hour frequency also takes into account the indication provided by the power dependent insertion limit alarm circuit and other information about CEA group positions available to the operator in the control room.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. Regulatory Guide 1.77, Rev. 0, May 1974.
4. UFSAR, Section 15.4.

This page intentionally blank

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Planar Radial Peaking Factors (F_{xy})

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a Loss Of Coolant Accident (LOCA), loss of flow accident, ejected Control Element Assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protective System (RPS) trip function. This LCO limits damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using full strength or part strength CEAs to alter the axial power distribution;

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and

c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. Limiting safety system settings and this LCO are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of Anticipated Operational Occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes xenon distribution skewing, which is a significant factor in controlling axial power distribution. Power distribution is a product of multiple parameters, various combinations of

(continued)

BACKGROUND (continued)

which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on Linear Heat Rate (LHR) and Departure from Nucleate Boiling (DNB).

Proximity to the DNB condition is expressed by the Departure from Nucleate Boiling Ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and AOOs is the DNBR Safety Limit as calculated by the CE-1 Correlation (Ref. 3) and corrected for such factors as rod bow and grid spacers, and it is accepted as an appropriate margin to DNB for all operating conditions.

There are two systems that monitor core power distribution online: the Core Operating Limit Supervisory System (COLSS) and the Core Protection Calculators (CPCs). The COLSS and CPCs that monitor the core power distribution are capable of verifying that the LHR and the DNBR do not exceed their limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating core power operating limits corresponding to the allowable peak LHR and DNBR values. The CPCs perform this function by continuously calculating actual values of DNBR and Local Power Density (LPD) for comparison with the respective trip setpoints.

DNBR penalty factors are included in both the COLSS and CPC DNBR calculations to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher than average burnup experience greater rod bow. Conversely, fuel assemblies that receive lower than average burnup experience less rod bow. In design calculations for a reload core, each batch of fuel is assigned a penalty applied to the maximum integrated planar radial power peak of the batch. This penalty is correlated with the amount of rod bow determined from the maximum average assembly burnup of the batch. A single net penalty for the COLSS and CPCs is then determined from the penalties associated with each batch that comprises a core reload, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

The COLSS indicates continuously to the operator how far the core is to the operating limits and provides an audible

(continued)

BACKGROUND (continued)

alarm if an operating limit is exceeded. Such a condition signifies a reduction in the capability of the plant to withstand an anticipated transient, but does not necessarily imply an immediate violation of fuel design limits. If the margin to fuel design limits continues to decrease, the RPS ensures that the specified acceptable fuel design limits are not exceeded for AOOs by initiating a reactor trip.

The COLSS continually generates an assessment of the calculated margin for LHR and DNBR specified limits. The data required for these assessments include measured incore neutron flux, CEA positions, and Reactor Coolant System (RCS) inlet temperature, pressure, and flow.

In addition to monitoring performed by the COLSS, the RPS (via the CPCs) continually infers the core power distribution and thermal margins by processing reactor coolant data, signals from excore neutron flux detectors, and input from redundant reed switch assemblies that indicates CEA position. In this case, the CPCs assume a minimum core power of 20% RTP. This threshold is set at 20% RTP because the power range excore neutron flux detecting system is inaccurate below this power level. If power distribution or other parameters are perturbed as a result of an AOO, the high LPD or low DNBR trips in the RPS initiate a reactor trip prior to exceeding fuel design limits.

The limits on ASI, F_{xy}, and T_q represent limits within which the LHR and DNBR algorithms are valid. These limits are obtained directly from the initial core or reload analysis.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation or AOOs (Ref. 4). The power distribution and CEA insertion and alignment LCOs prevent core power distributions from reaching levels that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 5);

(continued)

APPLICABLE SAFETY ANALYSES (continued)

b. During CEA misoperation events or a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 4);

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 6); and

d. The control rods (excluding part strength rods) must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 7).

The power density at any point in the core must be limited to maintain the fuel design criteria (Refs. 4 and 5). This result is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations between measured quantities, the power distribution, and the uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum Linear Heat Generation Rate (LHGR) so that the peak cladding temperature does not exceed 2200°F (Ref. 5). Peak cladding temperatures exceeding 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, CEAs, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the ranges used in the accident analyses (Ref. 1).

Fuel cladding damage does not occur because of conditions outside the limits of these LCOs for ASI, F_{xy}, and T_q during normal operation. However, fuel cladding damage results if an accident occurs from initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AZIMUTHAL POWER TILT (T_q)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a Loss Of Coolant Accident (LOCA), loss of flow accident, ejected Control Element Assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protective System (RPS) trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using full strength or part strength CEAs to alter the axial power distribution;

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and

c. Correcting off optimum conditions, (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings and this LCO are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of Anticipated Operational Occurrences (AOOs) and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes xenon distribution skewing, which is a significant factor in controlling axial power distribution.

(continued)

BACKGROUND (continued)

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on the Linear Heat Rate (LHR) and the Departure from Nucleate Boiling (DNB).

Proximity to the DNB condition is expressed by the Departure from Nucleate Boiling Ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and AOOs is the DNBR Safety Limit as calculated by the CE-1 Correlation (Ref. 3) and corrected for such factors as rod bow and grid spacers, and it is accepted as an appropriate margin to DNB for all operating conditions.

There are two systems that monitor core power distribution online: the Core Operating Limit Supervisory System (COLSS) and the Core Protection Calculators (CPCs). The COLSS and CPCs that monitor the core power distribution are capable of verifying that the LHR and the DNBR do not exceed their limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating core power operating limits corresponding to the allowable peak LHR and DNBR. The CPCs perform this function by continuously calculating actual values of DNBR and Local Power Density (LPD) for comparison with the respective trip setpoints.

A DNBR penalty factor is included in the COLSS and CPC DNBR calculation to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by the assembly. Fuel assemblies that incur higher than average burnup experience greater magnitude of rod bow. Conversely, fuel assemblies that receive lower than average burnup experience less rod bow. In design calculations for a reload core, each batch of fuel is assigned a penalty applied to the maximum integrated planar radial power peak of the batch. This penalty is correlated with the amount of rod bow that is determined from the maximum average assembly burnup of the batch. A single net penalty for the COLSS and CPCs is then determined from the penalties associated with each batch that comprises a core reload, accounting for the offsetting margins caused by the lower radial power peaks in the higher burnup batches.

(continued)

BACKGROUND (continued)

The COLSS indicates continuously to the operator how far the core is from the operating limits and provides an audible alarm if an operating limit is exceeded. Such a condition signifies a reduction in the capability of the plant to withstand an anticipated transient, but does not necessarily imply an immediate violation of fuel design limits. If the margin to fuel design limits continues to decrease, the RPS ensures that the specified acceptable fuel design limits are not exceeded for AOOs by initiating a reactor trip.

The COLSS continually generates an assessment of the calculated margin for LHR and DNBR specified limits. The data required for these assessments include measured incore neutron flux data, CEA positions, and Reactor Coolant System (RCS) inlet temperature, pressure, and flow.

In addition to the monitoring performed by the COLSS, the RPS (via the CPCs) continually infers the core power distribution and thermal margins by processing reactor coolant data, signals from excore neutron flux detectors, and input from redundant reed switch assemblies that indicates CEA position. In this case, the CPCs assume a minimum core power of 20% RTP. This threshold is set at 20% RTP because the power range excore neutron flux detection system is inaccurate below this power level. If power distribution or other parameters are perturbed as a result of an AOO, the high local power density or low DNBR trips in the RPS initiate a reactor trip prior to exceeding fuel design limits.

The limits on the ASI, F_{xy}, and T_q represent limits within which the LHR and DNBR algorithms are valid. These limits are obtained directly from the initial core or reload analysis.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of operation and AOOs (Ref. 4). The power distribution and CEA insertion and alignment LCOs preclude core power distributions that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 5);

(continued)

APPLICABLE SAFETY ANALYSES (continued)

b. During CEA misoperation events or a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 4);

c. During a CEA ejection accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 6); and

d. The control rods (excluding part strength rods) must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 7).

The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. 1). This result is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analysis (Ref. 2) with due regard for the correlations between measured quantities, the power distribution, and uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum Linear Heat Generation Rate (LHGR) so that the peak cladding temperature does not exceed 2200°F (Ref. 1). Peak cladding temperatures exceeding 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, CEAs, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits of these variables ensures that their actual values are within the range used in the accident analyses (Ref. 1).

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 Departure from Nucleate Boiling Ratio (DNBR)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial value assumed in the accident analyses. Specifically, operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a Loss Of Coolant Accident (LOCA), loss of flow accident, ejected Control Element Assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protective System (RPS) trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using full strength or part strength CEAs to alter the axial power distribution;

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and

c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings and this LCO are based on the accident analysis (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of Anticipated Operational Occurrences (AOOs) and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling axial power distribution.

(continued)

BASES

BACKGROUND (continued)

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on the Linear Heat Rate (LHR) and the Departure from nucleate boiling (DNB).

Proximity to the DNB condition is expressed by the DNBR, defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and AOOs is the DNBR Safety Limit as calculated by the CE-1 Correlation (Ref. 3) and corrected for such factors as rod bows and grid spacers and it is accepted as an appropriate margin to DNB for all operating conditions.

There are two systems that monitor core power distribution online: the Core Operating Limits Supervisory System (COLSS) and the Core Protection Calculators (CPCs). The COLSS and CPCs that monitor the core power distribution are capable of verifying that the LHR and DNBR do not exceed their limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating core power operating limits corresponding to the allowable peak LHR and DNBR. The CPCs perform this function by continuously calculating an actual value of DNBR and LPD for comparison with the respective trip setpoints.

A DNBR penalty factor is included in both the COLSS and CPC DNBR calculation to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher than average burnup experience a greater magnitude of rod bow. Conversely, fuel assemblies that receive lower than average burnup experience less rod bow. In design calculations for a reload core, each batch of fuel is assigned a penalty that is applied to the maximum integrated planar radial power peak of the batch. This penalty is correlated with the amount of rod bow that is determined from the maximum average assembly burnup of the batch. A single net penalty for the COLSS and CPCs is then determined from the penalties associated with each batch that comprises a core reload, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

(continued)

BASES

BACKGROUND (continued)

The COLSS indicates continuously to the operator how far the core is from the operating limits and provides an audible alarm when an operating limit is exceeded. Such a condition signifies a reduction in the capability of the plant to withstand an anticipated transient, but does not necessarily imply an immediate violation of fuel design limits. If the margin to fuel design limits continues to decrease, the RPS ensures that the specified acceptable fuel design limits are not exceeded during AOOs by initiating a reactor trip.

The COLSS continually generates an assessment of the calculated margin for LHR and DNBR specified limits. The data required for these assessments include measured incore neutron flux, CEA positions, and Reactor Coolant System (RCS) inlet temperature, pressure, and flow.

In addition to the monitoring performed by the COLSS, the RPS (via the CPCs) continually infers the core power distribution and thermal margins by processing reactor coolant data, signals from excore neutron flux detectors, and input from redundant reed switch assemblies that indicates CEA position. In this case, the CPCs assume a minimum core power of 20% RTP because the power range excore neutron flux detecting system is inaccurate below this power level. If power distribution or other parameters are perturbed as a result of an AOO, the high local power density or low DNBR trips in the RPS initiate a reactor trip prior to exceeding fuel design limits.

The limits on ASI, F_{xy}, and T_q represent limits within which the LHR and DNBR algorithms are valid. These limits are obtained directly from the initial core or reload analysis.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation or AOOs (Ref. 4). The power distribution and CEA insertion and alignment LCOs prevent core power distributions from reaching levels that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 5);

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. During CEA misoperation events or a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 3);

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 6); and

d. The control rods (excluding part strength rods) must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 7).

The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. 4). This is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations between measured quantities, the power distribution, and uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum Linear Heat Generation Rate (LHGR) so that the peak cladding temperature does not exceed 2200°F (Ref. 4). Peak cladding temperatures exceeding 2200°F may cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, CEAs, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the range used in the accident analyses (Ref. 1).

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.5 AXIAL SHAPE INDEX (ASI)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analysis. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a Loss Of Coolant Accident (LOCA), loss of flow accident, ejected Control Element Assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protective System (RPS) trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable conditions at the onset of a transient.

Methods of controlling the axial power distribution include:

a. Using full strength or part strength CEAs to alter the axial power distribution;

b. Decreasing CEA insertion by boration, thereby improving the axial power distribution; and

c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of Anticipated Operational Occurrences (AOOs) and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes xenon distribution skewing, which is a significant factor in controlling axial power distribution.

(continued)

BACKGROUND (continued)

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on the Linear Heat Rate (LHR) and the Departure from Nucleate Boiling (DNB).

Proximity to the DNB condition is expressed by the Departure from Nucleate Boiling Ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and AOOs is the DNBR Safety Limit as calculated by the CE-1 Correlation (Ref. 3), and corrected for such factors as rod bow and grid spacers, and it is accepted as an appropriate margin to DNB for all operating conditions.

There are two systems that monitor core power distribution online: the Core Operating Limit Supervisory System (COLSS) or the Core Protection Calculators (CPCs). The COLSS and CPCs monitor the core power distribution and are capable of verifying that the LHR and DNBR do not exceed their limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating core power operating limits corresponding to the allowable peak LHR and DNBR. The CPCs perform this function by continuously calculating actual values of DNBR and local power density (LPD) for comparison with the respective trip setpoints.

A DNBR penalty factor is included in both the COLSS and CPC DNBR calculations to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher than average burnup experience greater rod bow. Conversely, fuel assemblies that receive lower than average burnup experience less rod bow. In design calculations for a reload core, each batch of fuel is assigned a penalty that is applied to the maximum integrated planar radial power peak of the batch. This penalty is correlated with the amount of rod bow that is determined from the maximum average assembly burnup of the batch. A single net penalty for the COLSS and CPC is then determined from the penalties associated with each batch that comprises a core reload, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

(continued)

BASES

BACKGROUND (continued)

The COLSS indicates continuously to the operator how far the core is from the operating limits and provides an audible alarm if an operating limit is exceeded. Such a condition signifies a reduction in the capability of the plant to withstand an anticipated transient, but does not necessarily imply an immediate violation of fuel design limits. If the margin to fuel design limits continues to decrease, the RPS ensures that the specified acceptable fuel design limits are not exceeded for AOOs by initiating a reactor trip.

The COLSS continually generates an assessment of the calculated margin for LHR and DNBR specified limits. The data required for these assessments include measured incore neutron flux, CEA positions, and Reactor Coolant System (RCS) inlet temperature, pressure, and flow.

In addition to the monitoring performed by the COLSS, the RPS (via the CPCs) continually infers the core power distribution and thermal margins by processing reactor coolant data, signals from excore neutron flux detectors, and input from redundant reed switch assemblies that indicates CEA position. In this case, the CPCs assume a minimum core power of 20% RTP because the power range excore neutron flux detecting system is inaccurate below this power level. If power distribution or other parameters are perturbed as a result of an AOO, the high local power density or low DNBR trips in the RPS initiate a reactor trip prior to exceeding fuel design limits.

The limits on ASI, F_{xy}, and T_q represent limits within which the LHR and DNBR algorithms are valid. These limits are obtained directly from the initial core or reload analysis.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of operation or AOOs (Ref. 4). The power distribution and CEA insertion and alignment LCOs prevent core power distributions from reaching levels that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 5);

(continued)

APPLICABLE SAFETY ANALYSES (continued)

b. During CEA misoperation events or a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 4);

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 6);

d. The control rods (excluding part strength rods) must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 7).

The power density at any point in the core must be limited to maintain the fuel design criteria (Refs. 4 and 5). This is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations among measured quantities, the power distribution, and uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum Linear Heat Generation Rate (LHGR) so that the peak cladding temperature does not exceed 2200°F (Ref. 5). Peak cladding temperatures exceeding 2200°F may cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the range used in the accident analysis (Ref. 1).

Fuel cladding damage does not occur from conditions outside these LCOs during normal operation. However, fuel cladding damage results when an accident occurs due to initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and correspondingly increased local LHRs.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The ASI satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to LHR and DNBR operating limits. The power distribution LCO limits are provided in the COLR. The COLR provides separate limits that are based on different combinations of COLSS and CEACs being in and out of service.

The limitation on ASI ensures that the actual ASI value is maintained within the range of values used in the accident analyses. The ASI limits ensure that with T_q at its maximum upper limit, the DNBR does not drop below the DNBR Safety Limit for AOOs.

APPLICABILITY

Power distribution is a concern any time the reactor is critical. The power distribution LCOs, however, are only applicable in MODE 1 above 20% RTP. The reasons these LCOs are not applicable below 20% RTP are:

a. The incore neutron detectors that provide input to the COLSS, which then calculates the operating limits, are inaccurate due to the poor signal to noise ratio that they experience at relatively low core power levels.

b. As a result of this inaccuracy, the CPCs assume a minimum core power of 20% RTP when generating the LPD and DNBR trip signals. When the core power is below this level, the core is operating well below the thermal limits and the resultant CPC calculated LPD and DNBR trips are strongly conservative.

(continued)

BASES

ACTIONS

A.1

The ASI limits specified in the COLR ensure that the LOCA and loss of flow accident criteria assumed in the accident analyses remain valid. If the ASI exceeds its limit, a Completion Time of 2 hours is allowed to restore the ASI to within its specified limit. This duration gives the operator sufficient time to reposition the regulating or part strength CEAs to reduce the axial power imbalance. The magnitude of any potential xenon oscillation is significantly reduced if the condition is not allowed to persist for more than 2 hours.

B.1

If the ASI is not restored to within its specified limits within the required Completion Time, the reactor continues to operate with an axial power distribution mismatch. Continued operation in this configuration induces an axial xenon oscillation, and results in increased LHGRs when the xenon redistributes. Reducing thermal power to ≤ 20% RTP reduces the maximum LHR to a value that does not exceed the fuel design limits if a design basis event occurs. The allowed Completion Time of 4 hours is reasonable, based on operating experience, to reduce power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.5.1

The ASI can be monitored by both the incore (COLSS) and excore (CPC) neutron detector systems. The COLSS provides the operator with an alarm if an ASI limit is approached.

Verification of the ASI every 12 hours ensures that the operator is aware of changes in the ASI as they develop. A 12 hour Frequency for this Surveillance is acceptable because the mechanisms that affect the ASI, such as xenon redistribution or CEA drive mechanism malfunctions, cause slow changes in the ASI, which can be discovered before the limits are exceeded.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

Design Basis Definition (continued)

6, 7. Steam Generator Pressure - Low

The Steam Generator #1 Pressure - Low and Steam Generator #2 Pressure - Low trips provide protection against an excessive rate of heat extraction from the steam generators and resulting rapid, uncontrolled cooldown of the RCS. This trip is needed to shut down the reactor and assist the ESF System in the event of an MSLB or main feedwater line break accident. A main steam isolation signal (MSIS) is initiated simultaneously.

8, 9. Steam Generator Level - Low

The Steam Generator #1 Level - Low and Steam Generator #2 Level - Low trips ensure that a reactor trip signal is generated for the following events to help prevent exceeding the design pressure of the RCS due to the loss of the heat sink:

- Inadvertent Opening of a Steam Generator Atmospheric Dump Valve (AOO);
- Loss of Condenser Vacuum (AOO);
- Loss of Normal Feedwater Event (AOO);
- Feedwater System Pipe Break (Accident); and
- Single RCP Rotor Seizure (AOO).

10, 11. Steam Generator Level - High

The Steam Generator #1 Level - High and Steam Generator #2 Level - High trips are provided to protect the turbine from excessive moisture carryover in case of a steam generator overfill event. A Main Steam Isolation Signal (MSIS) is initiated simultaneously.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

Design Basis Definition (continued)

12, 13. Reactor Coolant Flow – Low

The Reactor Coolant Flow Steam Generator #1-Low and Reactor Coolant Flow Steam Generator #2-Low trips provide protection against an RCP Sheared Shaft Event. A trip is initiated when the pressure differential across the primary side of either steam generator decreases below a variable setpoint. This variable setpoint stays below the pressure differential by a preset value called the step function, unless limited by a preset maximum decreasing rate determined by the Ramp Function, or a set minimum value determined by the Floor Function. The setpoints ensure that a reactor trip occurs to limit fuel failure and ensure offsite doses are within 10 CFR 100 guidelines.

14. Local Power Density – High

The CPCs perform the calculations required to derive the DNBR and LPD parameters and their associated RPS trips. The DNBR – Low and LPD – High trips provide plant protection during the following AOOs and assist the ESF systems in the mitigation of the following accidents.

The LPD – High trip provides protection against fuel centerline melting due to the occurrence of excessive local power density peaks during the following AOOs:

- Decrease in Feedwater Temperature;
- Increase in Feedwater Flow;
- Increased Main Steam Flow (not due to the steam line rupture) Without Turbine Trip;
- Uncontrolled CEA Withdrawal From Low Power;
- Uncontrolled CEA Withdrawal at Power; and
- CEA Misoperation.

For the events listed above (except CEA Misoperation where the DNBR and LPD trips will occur near simultaneously), DNBR – Low will trip the reactor first, since DNB would occur before fuel centerline melting would occur.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

Design Basis Definition (continued)

15. Departure from Nucleate Boiling Ratio (DNBR) – Low

The CPCs perform the calculations required to derive the DNBR and LPD parameters and their associated RPS trips. The DNBR – Low and LPD – High trips provide plant protection during the following AOOs and assist the ESF systems in the mitigation of the following accidents.

The DNBR – Low trip provides protection against core damage due to the occurrence of locally saturated conditions in the limiting (hot) channel during the following events and is the primary reactor trip (trips the reactor first) for these events:

- Decrease in Feedwater Temperature;
- Increase in Feedwater Flow;
- Increased Main Steam Flow (not due to steam line rupture) Without Turbine Trip;
- Increased Main Steam Flow (not due to steam line rupture) With a Concurrent Single Failure of an Active Component;
- Steam Line Break With Concurrent Loss of Offsite AC Power;
- Loss of Normal AC Power;
- Partial Loss of Forced Reactor Coolant Flow;
- Total Loss of Forced Reactor Coolant Flow;
- Single Reactor Coolant Pump (RCP) Shaft Seizure;
- Uncontrolled CEA Withdrawal From Low Power;
- Uncontrolled CEA Withdrawal at Power;
- CEA Misoperation;
- Primary Sample or Instrument Line Break; and
- Steam Generator Tube Rupture.

In the above list, only the steam line break, the steam generator tube rupture, the RCP shaft seizure, and the sample or instrument line break are accidents. The rest are AOOs.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

15. Departure from Nucleate Boiling Ratio (DNBR)-Low (continued)

In the safety analyses for transients involving reactivity and power distribution anomalies, credit may be taken for the CPC VOPT auxiliary trip algorithm in lieu of the RPS VOPT trip function. The exact trip credited (CPC or RPS) is documented in chapter 15 of the UFSAR under the individual event sections. The CPC VOPT auxiliary trip acts through the CPC DNBR-Low and LPD-High trip contacts to provide over power protection. When credit is taken for the CPC VOPT algorithm, the CPC VOPT setpoints installed in the plant are based on the safety analyses and may differ from the RPS VOPT allowable values and nominal setpoints. The setpoints associated with the CPC VOPT are controlled via Addressable Constants (TS Section 5.4.1) and Reload Data Block Constants (Ref. 8 and 13). The CPC VOPT auxiliary trip algorithm may provide protection against core damage during the following events:

- Uncontrolled CEA Withdrawal From Low Power (AOO);
- Uncontrolled CEA Withdrawal at Power (AOO);
- Single CEA Withdrawal within Deadband (AOO);
- Steam Bypass Control System Misoperation (AOO);
- CEA Ejection (Accident); and
- Main Steam Line Break (Accident).

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

The Completion Time of 1 hour allotted to restore, bypass, or trip the channel is sufficient to allow the operator to take all appropriate actions for the failed channel and still ensures that the risk involved in operating with the failed channel is acceptable.

The failed channel must be restored to OPERABLE status prior to entering MODE 2 following the next MODE 5 entry. With a channel bypassed, the coincidence logic is now in a two-out-of-three configuration. The Completion Time of prior to entering MODE 2 following the next MODE 5 entry is based on adequate channel to channel independence, which allows a two-out-of-three channel operation, since no single failure will cause or prevent an ESF actuation.

The intent of this requirement is that should a failure occur that cannot be repaired during power operation, then continued operation is allowed without requiring a plant shutdown. However, the failure needs to be repaired during the next MODE 5 outage. Allowing the unit to exit MODE 5 is acceptable, as the appropriate retest may not be possible until normal operating pressures and temperatures are achieved. If the failure occurs while in MODE 5, then the problem needs to be resolved during that shutdown, and OPERABILITY restored prior to the subsequent MODE 2 entry.

B.1

Condition B applies to the failure of two channels of one or more input parameters in the following ESFAS automatic trip Functions:

1. Safety Injection Actuation Signal
   Containment Pressure — High
   Pressurizer Pressure — Low

2. Containment Spray Actuation Signal
   Containment Pressure — High High

(continued)

BASES

ACTIONS

B.1 (continued)

3. Containment Isolation Actuation Signal
   Containment Pressure – High
   Pressurizer Pressure – Low

4. Main Steam Isolation Signal
   Steam Generator #1 Pressure – Low
   Steam Generator #2 Pressure – Low
   Steam Generator #1 Level-High
   Steam Generator #2 Level-High
   Containment Pressure-High

5. Recirculation Actuation Signal
   Refueling Water Storage Tank Level – Low

6. Auxiliary Feedwater Actuation Signal SG #1 (AFAS-1)
   Steam Generator #1 Level – Low
   SG Pressure Difference (SG #2 > SG #1) – High

7. Auxiliary Feedwater Actuation Signal SG #2 (AFAS-2)
   Steam Generator #2 Level – Low
   SG Pressure Difference (SG #1 > SG #2) – High

With two inoperable channels, power operation may continue, provided one inoperable channel is placed in bypass and the other channel is placed in trip within 1 hour. With one channel of protective instrumentation bypassed, the ESFAS Function is in two-out-of-three logic in the bypassed input parameter, but with another channel failed, the ESFAS may be operating with a two-out-of-two logic. This is outside the assumptions made in the analyses and should be corrected. To correct the problem, the second channel is placed in trip. This places the ESFAS Function in a one-out-of-two logic. If any of the other OPERABLE channels receives a trip signal, ESFAS actuation will occur.

One of the two inoperable channels will need to be restored to OPERABLE status prior to the next required CHANNEL FUNCTIONAL TEST because channel surveillance testing on an OPERABLE channel requires that the OPERABLE channel be placed in bypass. However, it is not possible to bypass more than one ESFAS channel, and placing a second channel in trip will result in an ESFAS actuation. Therefore, if one ESFAS channel is in trip and a second channel is in bypass, a third inoperable channel would place the unit in LCO 3.0.3.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The Pressure and Temperature Limits Report (PTLR) contains P/T limit curves for heatup, cooldown, and inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature (Ref. 1).

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the Reactor Coolant Pressure Boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the ASME Code, Section III, Appendix G (Ref. 3).

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 4) and Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 3.

(continued)

BASES

BACKGROUND (continued)

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limit includes the Reference 2 requirement that the limit be no less than 40°F above the heatup curve or the cooldown curve and not less than the minimum permissible temperature for inservice leak and hydrostatic (ISLH) testing. However, the criticality limit is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 6), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) Analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

The RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The two elements of this LCO are:

a. The limit curves for heatup, cooldown, and ISLH testing; and

b. Limits on the rate of change of temperature.

The LCO limits apply to all components of the RCS, except the pressurizer.

These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure.

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and ISLH testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follows:

a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature;

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

(continued)

BASES

LCO (continued)

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T limits Specification provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 3). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, their Applicability is at all times, except when reactor vessel head is fully detensioned such that the RCS cannot be pressurized, in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer.

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; LCO 3.4.2, "RCS Minimum Temperature for Criticality"; and Safety Limit 2.1, "Safety Limits," also provide operational restrictions for pressure and temperature and maximum pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

The actions of this LCO consider the premise that a violation of the limits occurred during normal plant maneuvering. Severe violations caused by abnormal transients, at times accompanied by equipment failures, may also require additional actions from emergency operating procedures.

(continued)

BASES

ACTIONS

A.1 and A.2

Operation outside the P/T limits must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation to within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

Condition A is modified by a Note requiring Required Action A.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because:

a. The RCS remained in an unacceptable P/T region for an extended period of increased stress; or

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

b. A sufficiently severe event caused entry into an unacceptable region.

Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. With reduced pressure and temperature conditions, the possibility of propagation of undetected flaws is decreased.

Pressure and temperature are reduced by placing the plant in MODE 3 within 6 hours and in MODE 5 with RCS pressure < 500 psia within 36 hours.

The 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.

C.1 and C.2

The actions of this LCO, anytime other than in MODE 1, 2, 3, or 4, consider the premise that a violation of the limits occurred during normal plant maneuvering. Severe violations caused by abnormal transients, at times accompanied by equipment failures, may also require additional actions from emergency operating procedures. Operation outside the P/T limits must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The Completion Time of "immediately" reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in a short period of time in a controlled manner.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

Besides restoring operation to within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The Completion Time of prior to entering MODE 4 forces the evaluation prior to entering a MODE where temperature and pressure can be significantly increased. The evaluation for a mild violation is possible within several days, but more severe violations may require special, event specific stress analyses or inspections.

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the PTLR limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1 (continued)

This SR is modified by a Note that requires this SR be performed only during RCS system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

REFERENCES

1. TRM Appendix TA, Reactor Coolant System Pressure and Temperature Limits Report (PTLR); (limits determined using methods described in Topical Report CE NPSD-683-A, Revision 6, Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications, April 2001).

2. 10 CFR 50, Appendix G.

3. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.

4. ASTM E 185-82, July 1982.

5. 10 CFR 50, Appendix H.

6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

BASES

LCO (continued)

Note 2 requires secondary side water temperature in each SG is < 100°F above each of the RCS cold leg temperatures before an RCP may be started with any RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR.

Satisfying the above condition will preclude a large pressure surge in the RCS when the RCP is started.

Note 3 restricts RCP operation to no more than 2 RCPs with RCS cold leg temperature ≤ 200°F, and no more than 3 RCPs with RCS cold leg temperature >200°F but ≤ 500°F. Satisfying these conditions will maintain the analysis assumptions of the flow induced pressure correction factors due to RCP operation (Ref. 1)

An OPERABLE RCS loop consists of at least one OPERABLE RCP and an SG that is OPERABLE and has the minimum water level specified in SR 3.4.6.2.

Similarly, for the SDC System, an OPERABLE SDC train is composed of an OPERABLE SDC pump (LPSI) capable of providing flow to the SDC heat exchanger for heat removal. RCPs and SDC pumps are OPERABLE if they are capable of being powered and are able to provide flow, if required.

APPLICABILITY

In MODE 4, this LCO applies because it is possible to remove core decay heat and to provide proper boron mixing with either the RCS loops and SGs or the SDC System.

Operation in other MODES is covered by:

LCO 3.4.4 "RCS Loops-MODES 1 and 2";
LCO 3.4.5, "RCS Loops – MODE 3";

(continued)

BASES

APPLICABILITY (continued)

LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled";
LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation – High Water Level" (MODE 6); and
LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation – Low Water Level" (MODE 6).

ACTIONS

A.1

If only one required RCS loop is OPERABLE and in operation, redundancy for heat removal is lost. Action must be initiated immediately to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for decay heat removal.

B.1

If only one required SDC train is OPERABLE and in operation, redundancy for heat removal is lost. The plant must be placed in MODE 5 within the next 24 hours. Placing the plant in MODE 5 is a conservative action with regard to decay heat removal. With only one SDC train OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining SDC train, it would be safer to initiate that loss from MODE 5 (≤ 210°F) rather than MODE 4 (210°F to 350°F). The Completion Time of 24 hours is reasonable, based on operating experience, to reach MODE 5 from MODE 4, with only one SDC train operating, in an orderly manner and without challenging plant systems.

C.1 and C.2

If no RCS loops or SDC trains are OPERABLE, or in operation, all operations involving reduction of RCS boron concentration must be suspended and action to restore one RCS loop or SDC train to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and the margin to criticality must not be reduced in this type of operation. The immediate Completion Times reflect the importance of decay heat removal. The action to restore must continue until one loop or train is restored to operation.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This SR requires verification every 12 hours that one required loop or train is in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm. This ensures forced flow is providing heat removal. Verification includes flow rate, temperature, or pump status monitoring. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess RCS loop status. In addition, control room indication and alarms will normally indicate loop status.

SR 3.4.6.2

This SR requires verification every 12 hours of secondary side water level in the required SG(s) ≥ 25% wide range. An adequate SG water level is required in order to have a heat sink for removal of the core decay heat from the reactor coolant. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions.

SR 3.4.6.3

Verification that the required pump is OPERABLE ensures that an additional RCS loop or SDC train can be placed in operation, if needed to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pumps. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. PVNGS Operating License Amendments 52, 38 and 24 for Units 1, 2 and 3, respectively, and associated NRC Safety Evaluation dated July 25, 1990.

2. Not used.

3. PVNGS Calculation 13-JC-SH-0200, Section 2.9.

This page intentionally blank

BASES

LCO (continued)

in order to use the provisions of the Note allowing the pumps to be de-energized. In this MODE, the SG(s) can be used as the backup for SDC heat removal. To ensure their availability, the RCS loop flow path is to be maintained with subcooled liquid.

In MODE 5, it is sometimes necessary to stop all RCP or SDC forced circulation. This is permitted to change operation from one SDC train to the other, perform surveillance or startup testing, perform the transition to and from the SDC, or to avoid operation below the RCP minimum net positive suction head limit. The time period is acceptable because natural circulation is acceptable for decay heat removal the reactor coolant temperature can be maintained subcooled, and boron stratification affecting reactivity control is not expected.

Note 2 allows one SDC train to be inoperable for a period of up to 2 hours provided that the other SDC train is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable train during the only time when such testing is safe and possible.

Note 3 requires that secondary side water temperature in each SG is < 100°F above each of the RCS cold leg temperatures before an RCP may be started with any RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR.

Satisfying the above condition will preclude a low temperature overpressure event due to a thermal transient when the RCP is started.

Note 4 restricts RCP operation to no more than 2 RCPs with RCS cold leg temperature ≤ 200°F, and no more than 3 RCPs with RCS cold leg temperature > 200°F but ≤ 500°F. Satisfying these conditions will maintain the analysis assumptions of the flow induced pressure correction factors due to RCP operation (Ref. 3).

(continued)

BASES

LCO (continued)

Note 5 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of SDC trains from operation when at least one RCP is in operation. This Note provides for the transition to MODE 4 where an RCP is permitted to be in operation and replaces the RCS circulation function provided by the SDC trains.

An OPERABLE SDC train is composed of an OPERABLE SDC pump (CS or LPSI) capable of providing flow to the SDC heat exchanger for heat removal.

SDC pumps are OPERABLE if they are capable of being powered and are able to provide flow (current Section XI), if required. A SG can perform as a heat sink when it is OPERABLE and has the minimum water level specified in SR 3.4.7.2.

The RCS loops may not be considered filled until two conditions needed for operation of the steam generators are met. First, the RCS must be intact. This means that all removable portions of the primary pressure boundary (e.g., manways, safety valves) are securely fastened. Nozzle dams are removed. All manual drain and vent valves are closed, and any open system penetrations (e.g., letdown, reactor head vents) are capable of remote closure from the control room. An intact primary allows the system to be pressurized as needed to achieve the subcooling margin necessary to establish natural circulation cooling. When the RCS is not intact as described, a loss of SDC flow results in blowdown of coolant through boundary openings that also could prevent adequate natural circulation between the core and steam generators. Secondly, the concentration of dissolved or otherwise entrained gases in the coolant must be limited or other controls established so that gases coming out of solution in the SG U-tubes will not adversely affect natural circulation. With these conditions met, the SGs are a functional method of RCS heat removal upon loss of the operating SDC train. The ability to feed and steam SGs at all times is not required when RCS temperature is less than 210°F because significant loss of SG inventory through boiling will not occur during time anticipated to take corrective action. The required SG level provides sufficient time to either restore the SDC train or implement a method for feeding and steaming the SGs (using non-class components if necessary).

(continued)

BASES (continued)

REFERENCES

1. Not Used
2. CE NPSD-770 Analysis for Lower Mode Functional Recovery Guidelines.
3. PVNGS Operating License Amendments 52, 38, and 24 for Units 1, 2 and 3, respectively, and associated NRC Safety Evaluation dated July 25, 1990.
4. Not used.
5. PVNGS Calculation 13-JC-SH-0200, Section 2.9.

This page intentionally blank

BASES (continued)

LCO

One pressurizer safety valve is required to be OPERABLE in MODE 4 with no Shutdown Cooling System suction line relief valves in service. The four pressurizer safety valves are set to open 25 psia less than RCS design pressure (2475 psia) and within the ASME specified tolerance to avoid exceeding the maximum RCS design pressure SL to maintain accident analysis assumptions, and to comply with ASME Code requirements. The limit protected by this specification is the Reactor Coolant Pressure Boundary (RCPB) SL of 110% of design pressure. Inoperability of all 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 MODE 4 above the LTOP System temperatures OPERABILITY of one valve is required. MODE 4 is conservatively included, although the listed accidents may not require a safety valve for protection.

The requirements for overpressure protection in other MODES and in MODE 4 at or below the LTOP System temperatures are covered by LCOs 3.4.10, "Pressurizer Safety Valves - MODES 1, 2 and 3," and LCO 3.4.13, LTOP System.

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 72 hour exception is based on 18 hour outage time for each of the four valves. The 18 hour period is derived from operating experience that hot testing can be performed within this timeframe.

(continued)

BASES (continued)

ACTIONS

A.1, A.2, and A.3

If all pressurizer safety valves are inoperable, the plant must be brought to a condition where overpressure protection is provided, then to a MODE in which the requirement does not apply. To achieve this status, one Shutdown Cooling System suction line relief must be placed in service immediately, then the plant must be brought to at least MODE 4 with any RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR within 8 hours, so that LCO 3.4.13 (LTOP System) would apply. It is reasonable to pursue the ACTION to place a shutdown cooling system suction relief valve in service immediately (without delay) because the plant is already within the shutdown cooling system entry temperature of less than 350°F. The Completion Time of immediately requires that the required action be pursued without delay and in a controlled manner, and reflects the importance of maintaining the RCS overprotection system. The 8 hours allowed to be in MODE 4 with any RCS temperature less than or equal to the LTOP enable temperature specified in the PTLR is reasonable, based on operating experience, to reach this condition without challenging plant systems.

For the Shutdown Cooling System suction line relief valve that is required to be in service in accordance with Required Action A.1, SR 3.4.11.2 and SR 3.4.11.3 must be performed or verified performed within 12 hours. This ensures that the required Shutdown Cooling System suction line relief valve is OPERABLE. A Shutdown Cooling System suction line relief valve is OPERABLE when its isolation valves are open, its lift setpoint is set at 467 psig or less, and testing has proven its ability to open at that setpoint.

If the Required Actions and associated Completion Times are not met, overpressurization is possible.

The 8 hours Completion Time to be in MODE 4 with any RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR places the unit in a condition where the LCO does not apply.

(continued)

BACKGROUND (continued)

Shutdown Cooling System Suction Line Relief Valve Requirements (continued)

When a Shutdown Cooling System suction line relief valve lifts due to an increasing pressure transient, the release of coolant causes the pressure increase to slow and reverse. As the Shutdown Cooling System suction line relief valve releases coolant, the system pressure decreases until valve reseat pressure is reached and the Shutdown Cooling system suction line relief valve closes.

At low temperatures with the Shutdown Cooling System suction line relief valves aligned to the RCS, it is necessary to restrict heatup and cooldown rates to assure that P-T limits are not exceeded. These P-T limits are usually applicable to a finite time period such as one cycle, 5 EFPY, etc. and are based upon irradiation damage prediction by the end of the period. Accordingly, each time P-T limits change, the LTOP System needs to be reanalyzed and modified, if necessary, to continue its function.

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at containment ambient pressure in an RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting LTOP mass or heat input transient and maintaining pressure below the P/T limits. The required vent capacity may be provided by one or more vent paths.

For an RCS vent to meet the specified flow capacity, it requires removing all pressurizer safety valves, or similarly establishing a vent by opening the pressurizer manway (Ref. 11). The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

(continued)

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 3) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits during shutdown. In MODES 1, 2, and 3, and in MODE 4 with any RCS cold leg temperature greater than the LTOP enable temperature specified in the PTLR, the pressurizer safety valves prevent RCS pressure from exceeding the Reference 1 limits. At the LTOP enable temperature specified in the PTLR and below, overpressure prevention falls to the OPERABLE Shutdown Cooling System suction line relief valves or to a depressurized RCS and a sufficient sized RCS vent. 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 limit curves are revised, the LTOP System will be re-evaluated to ensure its functional requirements can still be satisfied using the Shutdown Cooling System suction line relief valve method or the depressurized and vented RCS condition.

Reference 3 contains the acceptance limits that satisfy the LTOP requirements. Any change to the RCS must be evaluated against these 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:

<u>Mass Input Type Transients</u>

a. Inadvertent safety injection; or

b. Charging/letdown flow mismatch.

<u>Heat Input Type Transients</u>

a. Inadvertent actuation of pressurizer heaters;

b. Loss of shutdown cooling (SDC); or

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

(continued)

APPLICABLE SAFETY ANALYSES (continued)

References 3, 7, 8 and 9 analyses demonstrate that either one Shutdown Cooling System suction line relief valve or the RCS vent can maintain RCS pressure below limits for the two most limiting analyzed events:

a. The start of an idle RCP with secondary water temperature of the SG ≤ 100°F above RCS cold leg temperatures.

b. An inadvertent SIAS with two HPSI pumps injecting into a water solid RCS, three charging pumps injecting, and letdown isolated.

Fracture mechanics analyses established the temperature of LTOP Applicability at less than or equal to the LTOP enable temperature specified in the PTLR. Above these temperatures, the pressurizer safety valves provide the reactor vessel pressure protection. The vessel materials were assumed to have a neutron irradiation accumulation equal to the effective full power years of operation specified in the PTLR.

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).

The fracture mechanics analyses show that the vessel is protected when the Shutdown Cooling System suction line relief valves are set to open at or below 467 psig. The setpoint is derived by modeling the performance of the LTOP System, assuming the limiting allowed LTOP transient. The Shutdown Cooling System suction line relief valves setpoints at or below the derived limit ensure the Reference 1 limits will be met.

The Shutdown Cooling System suction line relief valves setpoints will be re-evaluated for compliance 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 embrittlement caused by neutron irradiation. Revised P/T 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.

(continued)

APPLICABLE SAFETY ANALYSES (continued)

The Shutdown Cooling System suction line relief valves are considered active components. Thus, the failure of one Shutdown Cooling System suction line relief valve represents the worst case, single active failure.

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of 16 square inches is capable of mitigating the limiting allowed LTOP overpressure transient. In that event, this size vent maintains RCS pressure less than the maximum RCS pressure on the P/T limit curve.

The RCS vent size will also 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.

LTOP System satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

This LCO is required to ensure that the LTOP System is OPERABLE. The LTOP System is OPERABLE when the 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.

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

a. Two OPERABLE Shutdown Cooling System suction line relief valves; or

b. The depressurized RCS and an RCS vent.

A Shutdown Cooling System suction line relief valve is OPERABLE for LTOP when its isolation valves are open, its lift setpoint is set at 467 psig or less and testing has proven its ability to open at that setpoint.

An RCS vent is OPERABLE when open with an area ≥ 16 square inches. For an RCS vent to meet the specified flow capacity, it requires removing all pressurizer safety valves, or similarly establishing a vent by opening the pressurizer manway (Ref. 11). The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

(continued)

LCO (continued)

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

The Note requires that, before an RCP may be started, the secondary side water temperature (saturation temperature corresponding to SG pressure) in each SG is ≤ 100°F above each of the RCS cold leg temperatures. Satisfying this condition will preclude a large pressure surge in the RCS when the RCP is started.

APPLICABILITY

This LCO is applicable in MODE 4 when the temperature of any RCS cold leg is less than or equal to the LTOP enable temperature specified in the PTLR, 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 the LTOP enable temperature. The requirements for overpressure protection in MODES 1, 2 and 3, and in MODE 4 above the LTOP System temperatures are covered by LCO 3.4.10, "Pressurizer Safety Valves - MODES 1, 2, and 3," and LCO 3.4.11, "Pressurizer Safety Valves - MODE 4." When the reactor vessel head is off overpressurization cannot occur.

LCO 3.4.3 provides the operational P/T limits for all MODES.

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 operator action to mitigate the event.

(continued)

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 the risk assessment addressing inoperable the systems and components, should not be applied in this circumstance.

A.1

In MODE 4 when any RCS cold leg temperature is less than or equal to the LTOP enable temperature specified in the PTLR with one Shutdown Cooling System suction line relief valve inoperable, two Shutdown Cooling System suction line relief valves must be restored to OPERABLE status within a Completion Time of 7 days. Two valves are required to meet the LCO requirement and to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

The Completion Time is based on the facts that only one Shutdown Cooling System suction line relief valve 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.

B.1

The consequences of operational events that will overpressure the RCS are more severe at lower temperature (Ref. 6). Thus, one required Shutdown Cooling System suction line relief valve inoperable in MODE 5 or in MODE 6 with the head on, the Completion Time to restore inoperable valve to OPERABLE status is 24 hours.

The 24 hour Completion Time to restore two Shutdown Cooling System suction line relief valves OPERABLE in MODE 5 or in MODE 6 when the vessel head is on is a reasonable amount of time to investigate and repair several types of Shutdown

(continued)

REFERENCES (continued)

8. Pressure Transient Analyses

   a. V-PSAC-009 (3876 MWt w/Original Steam Generators)

   b. MN725-00118 (Unit 2, 4070 MWt w/Replacement Steam Generators)

   c. MN725-00562 (Units 31, 4070 MWt w/Replacement Steam Generators)

9. Mass Input Pressure Transient in Water Solid RCS

   a. V-PSAC-010 (3876 MWt w/Original Steam Generators)

   b. MN725-00117 (Unit 2, 4070 MWt w/Replacement Steam Generators)

   c. MN725-01495 (Units 31, 4070 MWt w/Replacement Steam Generators)

10. ASME, Boiler and Pressure Vessel Code, Section XI.

11. 13-C00-93-016, Sensitivity Study on Pressurizer Vent Paths vs. Days Post Shutdown.

This page intentionally blank

BASES (continued)

ACTIONS (continued)

B.1 and B.2

In MODE 1, 2, 3, or 4, when Required Action A.1 cannot be completed within the required Completion Time, the unit must be placed in a MODE that minimizes the accident risk. 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 operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1

In MODE 5 or 6, if Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE CREATCS train must be placed in operation immediately. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure will be readily detected.

D.1 and D.2

During movement of irradiated fuel assemblies, if Required Action A.1 cannot be completed within the Required Completion Time, the OPERABLE CREATCS train must be placed in operation immediately or movement of irradiated fuel assemblies must be suspended immediately. The first action ensures that the remaining train is OPERABLE, that no undetected failures preventing system operation will occur, and that any active failure will be readily detected. If the system is not immediately placed in operation, this action requires suspension of the movement of irradiated fuel assemblies in order to minimize the risk of a release of radioactivity that might require isolation of the control room. This does not preclude the movement of fuel to a safe position.

E.1 and E.2

In MODE 5 or 6, or during movement of irradiated fuel assemblies with two CREATCS trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

(continued)

BASES (continued)

ACTIONS (continued)	F.1 If both CREATCS trains are inoperable in MODE 1, 2, 3, or 4, the CREATCS may not be capable of performing the intended function and the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately
SURVEILLANCE REQUIREMENTS	SR 3.7.12.1 This SR verifies that the heat removal capability of the system is sufficient to meet design requirements. This SR consists of a combination of testing and calculations. An 18 month Frequency is appropriate, since significant degradation of the CREATCS is slow and is not expected over this time period.
REFERENCES	1. UFSAR, Section 9.4.

B 3.7 PLANT SYSTEMS

B 3.7.17 Spent Fuel Assembly Storage

BASES

BACKGROUND

The spent fuel storage is designed to store either new (nonirradiated) nuclear fuel assemblies, or burned (irradiated) fuel assemblies in a vertical configuration underwater. The storage pool was originally designed to store up to 1329 fuel assemblies in a borated fuel storage mode. The current storage configuration, which allows credit to be taken for boron concentration, burnup, and decay time, and does not require neutron absorbing (boraflex) storage cans, provides for a maximum storage of 1209 fuel assemblies in a four-region configuration. The design basis of the spent fuel cooling system, however, is to provide adequate cooling to the spent fuel during all operating conditions (including full core offload) for only 1205 fuel assemblies (UFSAR section 9.1.3). Therefore, an additional four spaces are mechanically blocked to limit the maximum number of fuel assemblies that may be stored in the spent fuel storage pool to 1205.

Region 1 is comprised of two 9x8 storage racks and one 12x8 storage rack. Cell blocking devices are placed in every other storage cell location in Region 1 to maintain a two-out-of-four checkerboard configuration. These cell blocking devices prevent inadvertent insertion of a fuel assembly into a cell that is not allowed to contain a fuel assembly.

Region 3 is comprised of three 9x8 storage racks and one 9x9 storage rack in Units 2 and 3. Region 3 is comprised of five 9x8 storage racks and one 9x8 storage rack in Unit 1. Since fuel assemblies may be stored in every Region 3 cell location, no cell blocking devices are installed in Region 3.

Regions 2 and 4 are mixed and are comprised of seven 9x8 storage racks and three 12x8 storage racks in Units 2 and 3. Regions 2 and 4 are mixed and are comprised of five 9x8 storage racks and three 12x8 storage racks in Unit 1. Regions 2 and 4 are mixed in a repeating 3x4 storage pattern in which two-out-of-twelve cell locations are designated Region 2 and ten-out-of-twelve cell locations are designated Region 4 (see UFSAR Figures 9.1-7 and 9.1-7A). Since fuel assemblies may be stored in every Region 2 and Region 4 cell location, no cell blocking devices are installed in Region 2 and Region 4.

(continued)

BASES

BACKGROUND (continued)	The spent fuel storage cells are installed in parallel rows with a nominal center-to-center spacing of 9.5 inches. This spacing, a minimum soluble boron concentration of 900 ppm, and the storage of fuel in the appropriate region based on assembly burnup in accordance with TS Figures 3.7.17-1, 3.7.17-2, and 3.7.17-3 is sufficient to maintain a k_{eff} of ≤ 0.95 for fuel of original maximum radially averaged enrichment of up to 4.80%.
APPLICABLE SAFETY ANALYSES	The spent fuel storage pool is designed for non-criticality by use of adequate spacing, credit for boron concentration, and the storage of fuel in the appropriate region based on assembly burnup in accordance with TS Figures 3.7.17-1, 3.7.17-2, and 3.7.17-3. The design requirements related to criticality (TS 4.3.1.1) are $k_{eff} < 1.0$ assuming no credit for boron and $k_{eff} \leq 0.95$ taking credit for soluble boron. The burnup versus enrichment requirements (TS Figures 3.7.17-1, 3.7.17-2, and 3.7.17-3) are developed assuming $k_{eff} < 1.0$ with no credit taken for soluble boron, and that $k_{eff} \leq 0.95$ assuming a soluble boron concentration of 900 ppm and the most limiting single fuel mishandling accident. The analysis of the reactivity effects of fuel storage in the spent fuel storage racks was performed by ABB-Combustion Engineering (CE) using the three-dimensional Monte Carlo code KENO-VA with the updated 44 group ENDF/B-5 neutron cross section library. The KENO code has been previously used by CE for the analysis of fuel rack reactivity and have been benchmarked against results from numerous critical experiments. These experiments simulate the PVNGS fuel storage racks as realistically as possible with respect to parameters important to reactivity such as enrichment and assembly spacing. The modeling of Regions 2, 3, and 4 included several conservative assumptions. These assumptions neglected the reactivity effects of poison shims in the assemblies and structural grids. These assumptions tend to increase the calculated effective multiplication factor (k_{eff}) of the racks. The stored fuel assemblies were modeled as CE 16x16 assemblies with a nominal pitch of 0.5065 inches between fuel rods, a fuel pellet diameter of 0.3255 inches, and a UO(2) density of 10.31 g/cc.

(continued)

BASES

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the spent fuel pool.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with Figures 3.7.17-1, 3.7.17-2, and 3.7.17-3, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figures 3.7.17-1, 3.7.17-2, and 3.7.17-3.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figures 3.7.17-1, 3.7.17-2, and 3.7.17-3 in the accompanying LCO and Specification 4.3.1.1.

To manually determine the allowed SFP region for a fuel assembly, the actual burnup is compared to the burnup requirement for the given initial enrichment and appropriate decay time from Figure 3.7.17-1, 3.7.17-2, or 3.7.17-3. If the actual burnup is greater than or equal to the burnup requirement, then the fuel assembly is eligible to be stored in the corresponding region. If the actual burnup is less than the burnup requirement, then the comparison needs to be repeated using another curve for a lower numbered region. Note the following:

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

- that a fuel assembly that does not meet the burnup requirement for Region 2 must be stored in Region 1.
- that any fuel assembly may be stored in Region 1.
- that any fuel assembly may be stored in a lower numbered region than the region for which it qualifies because burnup requirements decrease as region numbers decrease (refer also to Tech Spec 4.3.1.1).
- and that comparing actual burnup to the burnup requirement for zero decay time will always be correct or conservative.

REFERENCES

1. UFSAR, Sections 9.1.2 and 9.1.3.
2. PVNGS Operating License Amendments 82, 69, and 54 for Units 1, 2, and 3 respectively, and associated NRC Safety Evaluation, dated September 30, 1994.
3. Letter to T. E. Collins, U.S. NRC to T. Greene, WOG, "Acceptance for Referencing of Licensing Topical Report WCAP-14416-P, Westinghouse Spent Fuel Rack Methodology (TAC NO. M93254)", October 25, 1996.
4. 13-N-001-1900-1221-1, "Palo Verde Spent Fuel Pool Criticality Analysis," ABB calculation A-PV-FE-0106, revision 03, dated January 15, 1999.
5. Westinghouse letter NF-APS-10-19, "Criticality Safety Evaluation of the Spent Fuel Pool Map with a Proposed Region 3 Increase," dated February 25, 2010.