September 22, 2008

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

Gentlemen:

In the Matter of the	)	Docket No. 50-390
Tennessee Valley Authority	)	

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1 - CHANGES MADE TO THE WBN TECHNICAL SPECIFICATION BASES AND TECHNICAL REQUIREMENTS MANUAL

The purpose of this letter is to provide the NRC with copies of changes to the WBN Technical Specification Bases (TS Bases), through Revision 89, and WBN Technical Requirements Manual (TRM), through Revision 42, in accordance with WBN TS Section 5.6, "TS Bases Control Program," and WBN TRM Section 5.1, "Technical Requirements Control Program," respectively. These changes have been implemented at WBN during the period since WBN's last update (May 23, 2007) and meet the criteria described within the above control programs for which prior NRC approval is not required. Both control programs require such changes to be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). WBN's FSAR update in accordance with 10 CFR 50.71(e) will be provided via a separate letter.

During the period since the last update, WBN has been in the process of updating the format of the TS Bases and the TRM. This has resulted, for some updates, in the moving of text between pages. Therefore, in lieu of providing only the pages affected by a revision, the complete TS Bases section (i.e., Bases for LCO 3.1.8) or TRM section (i.e., TR and TR Bases for Section 3.0) is provided in the enclosures (listed below):

Enclosure 1 - WBN Technical Specification Bases - Table of Contents

Enclosure 2 - WBN Technical Specifications Bases - Changed Pages

Enclosure 3 - WBN Technical Requirements Manual - Table of Contents

Enclosure 4 - WBN Technical Requirements Manual - Changed Pages

There are no regulatory commitments in this submittal. If you should have any questions, please contact me at (423) 365-1824.

Sincerely,

Original signed by

M. K. Brandon
Manager, Site Licensing
and Industry Affairs

cc: see page 3

September 22, 2008

Enclosures
cc (Enclosures):

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Enclosure 1

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LIST OF ACRONYMS
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Acronym	Title
ABGTS	Auxiliary Building Gas Treatment System
ACRP	Auxiliary Control Room Panel
ASME	American Society of Mechanical Engineers
AFD	Axial Flux Difference
AFW	Auxiliary Feedwater System
ARO	All Rods Out
ARFS	Air Return Fan System
ADV	Atmospheric Dump Valve
BOC	Beginning of Cycle
CAOC	Constant Axial Offset Control
CCS	Component Cooling System
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CREVS	Control Room Emergency Ventilation System
CSS	Containment Spray System
CST	Condensate Storage Tank
DNB	Departure from Nucleate Boiling
ECCS	Emergency Core Cooling System
EFPD	Effective Full-Power Days
EGTS	Emergency Gas Treatment System
EOC	End of Cycle
ERCW	Essential Raw Cooling Water
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Features Actuation System
HEPA	High Efficiency Particulate Air
HVAC	Heating, Ventilating, and Air-Conditioning
LCO	Limiting Condition For Operation
MFIV	Main Feedwater Isolation Valve
MFRV	Main Feedwater Regulation Valve
MSIV	Main Steam Line Isolation Valve
MSSV	Main Steam Safety Valve
MTC	Moderator Temperature Coefficient
NMS	Neutron Monitoring System
ODCM	Offsite Dose Calculation Manual
PCP	Process Control Program
PIV	Pressure Isolation Valve
PORV	Power-Operated Relief Valve
PTLR	Pressure and Temperature Limits Report
QPTR	Quadrant Power Tilt Ratio
RAOC	Relaxed Axial Offset Control
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RTP	Rated Thermal Power

LIST OF ACRONYMS
(Page 2 of 2)

Acronym	Title
RTS	Reactor Trip System
RWST	Refueling Water Storage Tank
SG	Steam Generator
SI	Safety Injection
SL	Safety Limit
SR	Surveillance Requirement
UHS	Ultimate Heat Sink

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B 3.3-135	68	03-22-05
B 3.3-136	0	Initial
B 3.3-137	49	03-08-02
B 3.3-138	0	Initial
B 3.3-139	0	Initial
B 3.3-140	0	Initial
B 3.3-141	0	Initial
B 3.3-142	0	Initial
B 3.3-143	68	03-22-05
B 3.3-144	0	Initial
B 3.3-145	0	Initial
B 3.3-146	0	Initial
B 3.3-147	48	03-06-02
B 3.3-148	0	Initial
B 3.3-149	0	Initial
B 3.3-150	0	Initial
B 3.3-151	0	Initial
B 3.3-152	0	Initial
B 3.3-153	0	Initial
B 3.3-154	87	02-12-08
B 3.3-154A	87	02-12-08
B 3.3-155	9	04-29-97
B 3.3-156	87	02-12-08
B 3.3-157	0	Initial

TECHNICAL SPECIFICATIONS BASES

LIST OF EFFECTIVE PAGES

PAGE	REVISION	DATE
B 3.3-158	0	Initial
B 3.3-159	45	02-12-02
B 3.3-160	0	Initial
B 3.3-161	26	12-30-98
B 3.3-162	26	12-30-98
B 3.3-163	0	Initial
B 3.3-164	45	02-12-02
B 3.3-165	0	Initial
B 3.3-166	0	Initial
B 3.3-167	45	02-12-02
B 3.3-168	0	Initial
B 3.3-169	0	Initial
B 3.3-170	0	Initial
B 3.3-171	87	02-12-08
B 3.3-172	87	02-12-08
B 3.3-173	0	Initial
B 3.3-174	87	02-12-08
B 3.3-175	0	Initial
B 3.3-176	0	Initial
B 3.3-177	0	Initial
B 3.3-178	0	Initial
B 3.4-1	0	Initial
B 3.4-2	60	10-06-03
B 3.4-3	60	10-06-03
B 3.4-4	29	03-13-00
B 3.4-5	60	10-06-03
B 3.4-6	0	Initial
B 3.4-7	55	05-22-03
B 3.4-8	29	03-13-00
B 3.4-9	0	Initial
B 3.4-10	0	Initial
B 3.4-11	0	Initial
B 3.4-12	0	Initial
B 3.4-13	0	Initial
B 3.4-14	0	Initial
B 3.4-15	0	Initial
B 3.4-16	0	Initial
B 3.4-17	0	Initial
B 3.4-18	82	11-17-06
B 3.4-19	82	11-17-06
B 3.4-20	0	Initial
B 3.4-21	0	Initial
B 3.4-22	0	Initial
B 3.4-23	82	11-17-06
B 3.4-24	0	Initial
B 3.4-25	79	11-03-06

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PAGE	REVISION	DATE
B 3.4-26	29	03-13-00
B 3.4-27	0	Initial
B 3.4-28	0	Initial
B 3.4-29	82	11-17-06
B 3.4-30	0	Initial
B 3.4-31	0	Initial
B 3.4-32	79	11-03-06
B 3.4-33	79	11-03-06
B 3.4-34	79	11-03-06
B 3.4-35	82	11-17-06
B 3.4-36	79	11-03-06
B 3.4-37	29	03-13-00
B 3.4-38	0	Initial
B 3.4-39	68	03-22-05
B 3.4-40	0	Initial
B 3.4-41	0	Initial
B 3.4-42	0	Initial
B 3.4-43	0	Initial
B 3.4-44	29	03-13-00
B 3.4-45	29	03-13-00
B 3.4-46	0	Initial
B 3.4-47	0	Initial
B 3.4-48	0	Initial
B 3.4-49	89	05-01-08
B 3.4-50	89	05-01-08
B 3.4-51	0	Initial
B 3.4-52	42	03-07-01
B 3.4-53	68	03-22-05
B 3.4-54	42	03-07-01
B 3.4-55	42	03-07-01
B 3.4-56	42	03-07-01
B 3.4-57	89	05-01-08
B 3.4-58	0	Initial
B 3.4-59	0	Initial
B 3.4-60	0	Initial
B 3.4-61	0	Initial
B 3.4-62	0	Initial
B 3.4-63	0	Initial
B 3.4-64	0	Initial
B 3.4-65	68	03-22-05
B 3.4-66	0	Initial
B 3.4-67	68	03-22-05
B 3.4-68	0	Initial
B 3.4-69	0	Initial
B 3.4-70	0	Initial

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LIST OF EFFECTIVE PAGES

PAGE	REVISION	DATE
B 3.4-71	7	09-28-96
B 3.4-72	7	09-28-96
B 3.4-73	89	05-01-08
B 3.4-74	82	11-17-06
B 3.4-75	82	11-17-06
B 3.4-76	82	11-17-06
B 3.4-77	82	11-17-06
B 3.4-78	82	11-17-06
B 3.4-79	82	11-17-06
B 3.4-80	82	11-17-06
B 3.4-81	0	Initial
B 3.4-82	0	Initial
B 3.4-83	0	Initial
B 3.4-84	0	Initial
B 3.4-85	89	05-01-08
B 3.4-86	89	05-01-08
B 3.4-87	12	09-10-97
B 3.4-88	12	09-10-97
B 3.4-89	12	09-10-97
B 3.4-90	68	03-22-05
B 3.4-91	68	03-22-05
B 3.4-92	12	09-10-97
B 3.4-93	52	12-20-02
B 3.4-94	52	12-20-02
B 3.4-95	52	12-20-02
B 3.4-96	68	03-22-05
B 3.4-97	52	12-20-02
B 3.4-98	0	Initial
B 3.4-99	82	11-17-06
B 3.4-100	82	11-17-06
B 3.4-101	82	11-17-06
B 3.4-102	82	11-17-06
B 3.4-103	82	11-17-06
B 3.4-104	82	11-17-06
B 3.5-1	0	Initial
B 3.5-2	39	03-17-00
B 3.5-3	39	03-17-00
B 3.5-4	39	03-17-00
B 3.5-5	0	Initial
B 3.5-6	0	Initial
B 3.5-7	29	03-13-00
B 3.5-8	0	Initial
B 3.5-9	29	03-13-00
B 3.5-10	61	10-14-03
B 3.5-11	0	Initial
B 3.5-12	39	03-17-00
B 3.5-13	39	03-17-00
B 3.5-14	68	03-22-05

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PAGE	REVISION	DATE
B 3.5-15	68	03-22-05
B 3.5-16	0	Initial
B 3.5-17	62	10-15-03
B 3.5-18	89	05-01-08
B 3.5-19	80	11-08-06
B 3.5-20	0	Initial
B 3.5-21	68	03-22-05
B 3.5-22	0	Initial
B 3.5-23	0	Initial
B 3.5-24	0	Initial
B 3.5-25	0	Initial
B 3.5-26	88	03-06-08
B 3.5-27	61	10-14-03
B 3.5-28	0	Initial
B 3.5-29	29	03-13-00
B 3.5-30	29	03-13-00
B 3.5-31	0	Initial
B 3.5-32	0	Initial
B 3.5-33	0	Initial
B 3.5-34	29	03-13-00
B 3.6-1	10	05-27-97
B 3.6-2	10	05-27-97
B 3.6-3	10	05-27-97
B 3.6-4	10	05-27-97
B 3.6-5	10	05-27-97
B 3.6-6	5	07-03-96
B 3.6-7	10	05-27-97
B 3.6-8	0	Initial
B 3.6-9	0	Initial
B 3.6-10	0	Initial
B 3.6-11	0	Initial
B 3.6-12	10	05-27-97
B 3.6-13	10	05-27-97
B 3.6-14	0	Initial
B 3.6-15	0	Initial
B 3.6-16	0	Initial
B 3.6-17	8	11-21-96
B 3.6-18	76	09-22-06
B 3.6-19	0	Initial
B 3.6-20	0	Initial
B 3.6-21	0	Initial
B 3.6-22	0	Initial
B 3.6-23	0	Initial
B 3.6-24	0	Initial
B 3.6-25	10	05-27-97
B 3.6-26	10	05-27-97
B 3.6-27	10	05-27-97

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PAGE	REVISION	DATE
B 3.6-28	76	09-22-06
B 3.6-29	0	Initial
B 3.6-30	71	02-01-06
B 3.6-31	0	Initial
B 3.6-32	0	Initial
B 3.6-33	29	03-13-00
B 3.6-34	29	03-13-00
B 3.6-35	0	Initial
B 3.6-36	0	Initial
B 3.6-37	76	09-22-06
B 3.6-38	0	Initial
B 3.6-39	0	Initial
B 3.6-40	89	05-01-08
B 3.6-41	83	11-20-06
B 3.6-42	89	05-01-08
B 3.6-43	0	Initial
B 3.6-44	61	10-14-03
B 3.6-45	0	Initial
B 3.6-46	68	03-22-05
B 3.6-47	0	Initial
B 3.6-48	0	Initial
B 3.6-49	0	Initial
B 3.6-50	0	Initial
B 3.6-51	0	Initial
B 3.6-52	0	Initial
B 3.6-53	0	Initial
B 3.6-54	16	06-09-98
B 3.6-55	0	Initial
B 3.6-56	85	03-22-07
B 3.6-57	71	02-01-06
B 3.6-58	29	03-13-00
B 3.6-59	29	03-13-00
B 3.6-60	0	Initial
B 3.6-61	0	Initial
B 3.6-62	0	Initial
B 3.6-63	0	Initial
B 3.6-64	0	Initial
B 3.6-65	81	11-15-06
B 3.6-66	0	Initial
B 3.6-67	0	Initial
B 3.6-68	0	Initial
B 3.6-69	29	03-13-00
B 3.6-70	81	11-15-06
B 3.6-71	36	08-23-00
B 3.6-72	36	08-23-00
B 3.6-72a	36	08-23-00

LIST OF EFFECTIVE PAGES

PAGE	REVISION	DATE
B 3.6-73	36	08-23-00
B 3.6-74	0	Initial
B 3.6-75	0	Initial
B 3.6-76	0	Initial
B 3.6-77	0	Initial
B 3.6-78	36	08-23-00
B 3.6-79	0	Initial
B 3.6-80	6	09-09-96
B 3.6-81	6	09-09-96
B 3.6-82	21	11-30-98
B 3.6-83	6	09-09-96
B 3.6-84	0	Initial
B 3.6-85	0	Initial
B 3.6-86	0	Initial
B 3.6-87	0	Initial
B 3.6-88	0	Initial
B 3.6-89	0	Initial
B 3.6-90	0	Initial
B 3.6-91	0	Initial
B 3.6-92	0	Initial
B 3.6-93	0	Initial
B 3.6-94	0	Initial
B 3.6-95	0	Initial
B 3.6-96	85	03-22-07
B 3.6-97	85	03-22-07
B 3.6-98	29	03-13-00
B 3.7-1	31	04-06-00
B 3.7-2	31	04-06-00
B 3.7-3	41	01-22-01
B 3.7-4	31	04-06-00
B 3.7-5	89	05-01-08
B 3.7-6	89	05-01-08
B 3.7-7	0	Initial
B 3.7-8	0	Initial
B 3.7-9	0	Initial
B 3.7-10	0	Initial
B 3.7-11	0	Initial
B 3.7-12	89	05-01-08
B 3.7-13	76	09-22-06
B 3.7-14	0	Initial
B 3.7-15	0	Initial
B 3.7-16	0	09-22-06
B 3.7-17	76	Initial
B 3.7-18	89	05-01-08
B 3.7-19	89	05-01-08

LIST OF EFFECTIVE PAGES

PAGE	REVISION	DATE
B 3.7-20	0	Initial
B 3.7-21	0	Initial
B 3.7-22	68	03-22-05
B 3.7-23	24	12-17-98
B 3.7-24	0	Initial
B 3.7-25	0	Initial
B 3.7-26	0	Initial
B 3.7-27	0	Initial
B 3.7-28	68	03-22-05
B 3.7-29	0	Initial
B 3.7-30	0	Initial
B 3.7-31	89	05-01-08
B 3.7-32	20	10-26-98
B 3.7-33	89	05-01-08
B 3.7-34	0	Initial
B 3.7-35	41	01-22-01
B 3.7-36	0	Initial
B 3.7-37	29	03-13-00
B 3.7-38	0	Initial
B 3.7-39	0	Initial
B 3.7-40	0	Initial
B 3.7-41	0	Initial
B 3.7-42	0	Initial
B 3.7-43	0	Initial
B 3.7-44	0	Initial
B 3.7-45	0	Initial
B 3.7-46	0	Initial
B 3.7-47	0	Initial
B 3.7-48	0	Initial
B 3.7-49	0	Initial
B 3.7-50	29	03-13-00
B 3.7-51	0	Initial
B 3.7-52	64	03-23-04
B 3.7-53	45	02-12-02
B 3.7-54	45	02-12-02
B 3.7-55	45	02-12-02
B 3.7-56	0	Initial
B 3.7-57	64	03-23-04
B 3.7-58	64	03-23-04
B 3.7-59	64	03-23-04
B 3.7-60	45	02-12-02
B 3.7-61	64	03-23-04
B 3.7-62	87	02-12-08
B 3.7-62A	87	02-12-08
B 3.7-63	87	02-12-08
B 3.7-64	0	Initial

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PAGE	REVISION	DATE
B 3.7-65	0	Initial
B 3.7-66	35	08-14-00
B 3.7-67	55	05-22-03
B 3.7-68	0	Initial
B 3.7-69	0	Initial
B 3.7-70	0	Initial
B 3.7-71	47	03-01-02
B 3.7-72	0	Initial
B 3.7-73	0	Initial
B 3.7-74	0	Initial
B 3.7-75	61	10-14-03
B 3.7-76	61	10-14-03
B 3.7-77	61	10-14-03
B 3.8-1	0	Initial
B 3.8-2	0	Initial
B 3.8-3	0	Initial
B 3.8-4	0	Initial
B 3.8-5	68	03-22-05
B 3.8-6	0	Initial
B 3.8-7	0	Initial
B 3.8-8	50	08-30-02
B 3.8-9	65	04-01-04
B 3.8-10	63	12-08-03
B 3.8-11	50	08-30-02
B 3.8-12	50	08-30-02
B 3.8-13	50	08-30-02
B 3.8-14	50	08-30-02
B 3.8-15	50	08-30-02
B 3.8-16	0	Initial
B 3.8-17	0	Initial
B 3.8-18	29	03-13-00
B 3.8-19	0	Initial
B 3.8-20	0	Initial
B 3.8-21	0	Initial
B 3.8-22	0	Initial
B 3.8-23	0	Initial
B 3.8-24	0	Initial
B 3.8-25	0	Initial
B 3.8-26	0	Initial
B 3.8-27	19	10-19-98
B 3.8-28	50	08-30-02
B 3.8-29	0	Initial
B 3.8-30	0	Initial
B 3.8-31	0	Initial

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PAGE	REVISION	DATE
B 3.8-32	0	Initial
B 3.8-33	0	Initial
B 3.8-34	0	Initial
B 3.8-35	50	08-30-02
B 3.8-36	63	12-08-03
B 3.8-37	0	Initial
B 3.8-38	0	Initial
B 3.8-39	0	Initial
B 3.8-40	0	Initial
B 3.8-41	0	Initial
B 3.8-42	0	Initial
B 3.8-43	0	Initial
B 3.8-44	0	Initial
B 3.8-45	0	Initial
B 3.8-46	0	Initial
B 3.8-47	55	05-22-03
B 3.8-48	55	05-22-03
B 3.8-49	0	Initial
B 3.8-50	0	Initial
B 3.8-51	29	03-13-00
B 3.8-52	0	Initial
B 3.8-53	29	03-13-00
B 3.8-54	0	Initial
B 3.8-55	0	Initial
B 3.8-56	0	Initial
B 3.8-57	0	Initial
B 3.8-58	0	Initial
B 3.8-59	0	Initial
B 3.8-60	0	Initial
B 3.8-61	69	04-04-05
B 3.8-62	0	Initial
B 3.8-63	0	Initial
B 3.8-64	66	05-21-04
B 3.8-65	19	10-19-98
B 3.8-66	19	10-19-98
B 3.8-67	19	10-19-98
B 3.8-68	0	Initial
B 3.8-69	0	Initial
B 3.8-70	0	Initial
B 3.8-71	0	Initial
B 3.8-72	0	Initial
B 3.8-73	0	Initial
B 3.8-74	0	Initial
B 3.8-75	0	Initial
B 3.8-76	0	Initial

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PAGE	REVISION	DATE
B 3.8-77	0	Initial
B 3.8-78	0	Initial
B 3.8-79	0	Initial
B 3.8-80	0	Initial
B 3.8-81	78	10-13-06
B 3.8-82	78	10-13-06
B 3.8-83	78	10-13-06
B 3.8-84	75	09-18-06
B 3.8-85	0	Initial
B 3.8-86	78	10-13-06
B 3.8-87	75	09-18-06
B 3.8-88	0	Initial
B 3.8-89	78	10-13-06
B 3.8-90	0	Initial
B 3.8-91	78	10-13-06
B 3.8-92	0	Initial
B 3.8-93	78	10-13-06
B 3.8-94	0	Initial
B 3.8-95	0	Initial
B 3.8-96	0	Initial
B 3.8-97	0	Initial
B 3.8-98	33	05-02-00
B 3.8-99	0	Initial
B 3.8-100	0	Initial
B 3.8-101	0	Initial
B 3.8-102	0	Initial
B 3.9-1	0	Initial
B 3.9-2	0	Initial
B 3.9-3	68	03-22-05
B 3.9-4	0	Initial
B 3.9-5	0	Initial
B 3.9-6	68	03-22-05
B 3.9-7	0	Initial
B 3.9-8	0	Initial
B 3.9-9	0	Initial
B 3.9-10	0	Initial
B 3.9-11	0	Initial
B 3.9-12	45	02-12-02
B 3.9-13	73	09-11-06
B 3.9-14	74	09-16-06
B 3.9-15	45	02-12-02
B 3.9-16	37	09-08-00
B 3.9-17	0	Initial
B 3.9-18	23	01-05-99
B 3.9-19	0	Initial

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PAGE	REVISION	DATE
B 3.9-20	0	Initial
B 3.9-21	0	Initial
B 3.9-22	68	03-22-05
B 3.9-23	68	03-22-05
B 3.9-24	0	Initial
B 3.9-25	55	05-22-03
B 3.9-26	45	02-12-02
B 3.9-27	55	05-22-03
B 3.9-28	45	02-12-02
B 3.9-29	0	Initial
B 3.9-30	87	02-12-08
B 3.9-31	87	02-12-08
B 3.9-32	0	Initial
B 3.9-33	86	01-31-08
B 3.9-34	0	Initial

TECHNICAL SPECIFICATION BASES - REVISION LISTING

(This listing is an administrative tool maintained by WBN Licensing and may be updated without formally revising the Technical Specification Bases Table-of-Contents)

REVISIONS	ISSUED	SUBJECT
NPF-20	11-09-95	Low Power Operating License
Revision 1	12-08-95	Slave Relay Testing
NPF-90	02-07-96	Full Power Operating License
Revision 2 (Amendment 1)	12-08-95	Turbine Driven AFW Pump Suction Requirement
Revision 3	03-27-96	Remove Cold Leg Accumulator Alarm Setpoints
Revision 4 (Amendment 2)	06-13-96	Ice Bed Surveillance Frequency And Weight
Revision 5	07-03-96	Containment Airlock Door Indication
Revision 6 (Amendment 3)	09-09-96	Ice Condenser Lower Inlet Door Surveillance
Revision 7	09-28-96	Clarification of COT Frequency for COMS
Revision 8	11-21-96	Admin Control of Containment Isol. Valves
Revision 9	04-29-97	Switch Controls For Manual CI-Phase A
Revision 10 (Amendment 5)	05-27-97	Appendix-J, Option B
Revision 11 (Amendment 6)	07-28-97	Spent Fuel Pool Rerack
Revision 12	09-10-97	Heat Trace for Radiation Monitors
Revision 13 (Amendment 7)	09-11-97	Cycle 2 Core Reload
Revision 14	10-10-97	Hot Leg Recirculation Timeframe
Revision 15	02-12-98	EGTS Logic Testing
Revision 16 (Amendment 10)	06-09-98	Hydrogen Mitigation System Temporary Specification

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REVISIONS	ISSUED	SUBJECT
Revision 17	07-31-98	SR Detectors (Visual/audible indication)
Revision 18 (Amendment 11)	09-09-98	Relocation of F(Q) Penalty to COLR
Revision 19 (Amendment 12)	10-19-98	Online Testing of the Diesel Batteries and Performance of the 24 Hour Diesel Endurance Run
Revision 20 (Amendment 13)	10-26-98	Clarification of Surveillance Testing Requirements for TDAFW Pump
Revision 21	11-30-98	Clarification to Ice Condenser Door ACTIONS and door lift tests, and Ice Bed sampling and flow blockage SRs
Revision 22 (Amendment 14)	11-10-98	COMS - Four Hour Allowance to Make RHR Suction Relief Valve Operable
Revision 23	01-05-99	RHR Pump Alignment for Refueling Operations
Revision 24 (Amendment 16)	12-17-98	New action for Steam Generator ADVs due to Inoperable ACAS.
Revision 25	02-08-99	Delete Reference to PORV Testing Not Performed in Lower Modes
Revision 26 (Amendment 17)	12-30-98	Slave Relay Surveillance Frequency Extension to 18 Months
Revision 27 (Amendment 18)	01-15-99	Deletion of Power Range Neutron Flux High Negative Rate Reactor Trip Function

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REVISIONS	ISSUED	SUBJECT
Revision 28	04-02-99	P2500 replacement with Integrated Computer System (ICS). Delete Reference to ERFDS as a redundant input signal.
Revision 29	03-13-00	Added notes to address instrument error in various parameters shown in the Bases. Also corrected the applicable modes for TS 3.6.5 from 3 and 4 to 2, 3 and 4.
Revision 30 (Amendment 23)	03-22-00	For SR 3.3.2.10, Table 3.3.2-1, one time relief from turbine trip response time testing. Also added Reference 14 to the Bases for LCO 3.3.2.
Revision 31 (Amendment 19)	03-07-00	Reset Power Range High Flux Reactor Trip Setpoints for Multiple Inoperable MSSVs.
Revision 32	04-13-00	Clarification to Reflect Core Reactivity and MTC Behavior.
Revision 33	05-02-00	Clarification identifying four distribution boards primarily used for operational convenience.
Revision 34 (Amendment 24)	07-07-00	Elimination of Response Time Testing
Revision 35	08-14-00	Clarification of ABGTS Surveillance Testing
Revision 36 (Amendments 22 and 25)	08-23-00	Revision of Ice Condenser sampling and flow channel surveillance requirements
Revision 37 (Amendment 26)	09-08-00	Administrative Controls for Open Penetrations During Refueling Operations

TECHNICAL SPECIFICATION BASES - REVISION LISTING

(This listing is an administrative tool maintained by WBN Licensing and may be updated without formally revising the Technical Specification Bases Table-of-Contents)

REVISIONS	ISSUED	SUBJECT
Revision 38	09-17-00	SR 3.2.1.2 was revised to reflect the area of the core that will be flux mapped.
Revision 39 (Amendments 21and 28)	09-13-00	Amendment 21 - Implementation of Best Estimate LOCA analysis. Amendment 28 - Revision of LCO 3.1.10, “Physics Tests Exceptions - Mode 2.”
Revision 40	09-28-00	Clarifies WBN’s compliance with ANSI/ANS-19.6.1 and deletes the detailed descriptions of Physics Tests.
Revision 41 (Amendment 31)	01-22-01	Power Uprate from 3411 MWt to 3459 MWt Using Leading Edge Flow Meter (LEFM)
Revision 42	03-07-01	Clarify Operability Requirements for Pressurizer PORVs
Revision 43	05-29-01	Change CVI Response Time from 5 to 6 Seconds
Revision 44 (Amendment 33)	01-31-02	Ice weight reduction from 1236 to 1110 lbs per basket and peak containment pressure revision from 11.21 to 10.46 psig.
Revision 45 (Amendment 35)	02-12-02	Relaxation of CORE ALTERATIONS Restrictions
Revision 46	02-25-02	Clarify Equivalent Isolation Requirements in LCO 3.9.4
Revision 47 (Amendment 38)	03-01-02	RCS operational LEAKAGE and SG Alternate Repair Criteria for Axial Outside Diameter Stress Corrosion Cracking (ODSCC)
Revision 48 (Amendment 36)	03-06-02	Increase Degraded Voltage Time Delay from 6 to 10 seconds.

TECHNICAL SPECIFICATION BASES - REVISION LISTING

(This listing is an administrative tool maintained by WBN Licensing and may be updated without formally revising the Technical Specification Bases Table-of-Contents)

REVISIONS	ISSUED	SUBJECT
Revision 49 (Amendment 34)	03-08-02	Deletion of the Post-Accident Sampling System (PASS) requirements from Section 5.7.2.6 of the Technical Specifications.
Revision 50 (Amendment 39)	08-30-02	Extension of the allowed outage time (AOT) for a single diesel generator from 72 hours to 14 days.
Revision 51	11-14-02	Clarify that Shutdown Banks C and D have only One Rod Group
Revision 52 (Amendment 41)	12-20-02	RCS Specific Activity Level reduction from <1.0 μCi/gm to <0.265 μCi/gm.
Revision 53 (Amendment 42)	01-24-03	Revise SR 3.0.3 for Missed Surveillances
Revision 54 (Amendment 43)	05-01-03	Exigent TS SR 3.5.2.3 to delete SI Hot Leg Injection lines from SR until U1C5 outage.
Revision 55	05-22-03	Editorial corrections (PER 02-015499), correct peak containment pressure, and revise I-131 gap inventory for an FHA.
Revision 56	07-10-03	TS Bases for SRs 3.8.4.8 through SR 3.8.4.10 clarification of inter-tier connection resistance test.
Revision 57 (Amendment 48)	08-11-03	TS Bases for B 3.5.2 Background information provides clarification when the 9 hrs for hot leg recirculation is initiated.
Revision 58 (Amendment 45)	09-26-03	The Bases for LCO 3.8.7 and 3.8.8 were revised to delete the Unit 2 Inverters.
Revision 59 (Amendment 46)	09-30-03	Address new DNB Correlation in B2.1.1 and B3.2.12 for Robust Fuel Assembly (RFA)-2.
Revision 60 (Amendment 47)	10-06-03	RCS Flow Measurement Using Elbow Tap Flow Meters (Revise Table 3.3.1-1(10) & SR 3.4.1.4).

REVISIONS	ISSUED	SUBJECT
Revision 61 (Amendments 40 and 48)	10-14-03	Incorporated changes required to implement the Tritium Program (Amendment 40) and Stepped Boron Concentration increases for RWST and CLAs (Amendment 48) depending on the number of TPBARS installed into the reactor core.
Revision 62	10-15-03	Clarified ECCS venting in Bases Section B 3.5.2 (WBN-TS-03-19)
Revision 63	12-08-03	The contingency actions listed in Bases Table 3.8.1-2 were reworded to be consistent with the NRC Safety Evaluation that approved Tech Spec Amendment 39.
Revision 64 (Amendment 50)	03-23-04	Incorporated Amendment 50 for the seismic qualification of the Main Control Room duct work. Amendment 50 revised the Bases for LCO 3.7.10, “CREVS,” and LCO 3.7.11, “CREATCS.” An editorial correction was made on Page B 3.7-61.
Revision 65	04-01-04	Revised the Bases for Action B.3.1 of LCO 3.8.1 to clarify that a common cause assessment is not required when a diesel generator is made inoperable due to the performance of a surveillance.
Revision 66	05-21-04	Revised Page B 3.8-64 (Bases for LCO 3.8.4) to add a reference to SR 3.8.4.13 that was inadvertently deleted by the changes made for Amendment 12.
Revision 67 (Amendment 45)	03-05-05	Revised the Bases for LCOs 3.8.7, 3.8.8 and 3.8.9 to incorporate changes to the Vital Inverters (DCN 51370). Refer to the changes made for Bases Revision 58 (Amendment 45)

TECHNICAL SPECIFICATION BASES - REVISION LISTING
(This listing is an administrative tool maintained by WBN Licensing and may be updated without formally revising the Technical Specification Bases Table-of-Contents)

REVISIONS	ISSUED	SUBJECT
Revision 68 (Amendment 55)	03-22-05	Amendment 55 modified the requirements for mode change limitations in LCO 3.0.4 and SR 3.0.4 by incorporating TSTF-359, Revision 9.
Revision 68 (Amendment 55 and 56)	03-22-05	Change MSLB primary to secondary leakage from 1 gpm to 3 gpm (WBN-TS-03-14).
Revision 69 (Amendment 54)	04-04-05	Revised the use of the terms inter-tier and inter-rack in the Bases for SR 3.8.4.10.
Revision 70 (Amendment 58)	10-17-05	Alternate monitoring process for a failed Rod Position Indicator (RPI) (TS-03-12).
Revision 71 (Amendment 59)	02-01-06	Temporary Use of Penetrations in Shield Building Dome During Modes 1-4 (WBN-TS-04-17)
Revision 72	08-31-06	Minor Revision (Corrects Typographical Error) – Changed LCO Bases Section 3.4.6 which incorrectly referred to Surveillance Requirement 3.4.6.2 rather than correctly identifying Surveillance Requirement 3.4.6.3.
Revision 73	09-11-06	Updated the Bases for LCO 3.9.4 to clarify that penetration flow paths through containment to the outside atmosphere must be limited to less than the ABSCE breach allowance. Also administratively removed from the Bases for LCO 3.9.4 a statement on core alterations that should have been removed as part of Amendment 35.

REVISIONS	ISSUED	SUBJECT
Revision 74	09-16-06	For the LCO section of the Bases for LCO 3.9.4, administratively removed the change made by Revision 73 to the discussion of an LCO note and placed the change in another area of the LCO section.
Revision 75 (Amendment 45)	09-18-06	Revised the Bases for LCOs 3.8.7, 3.8.8 and 3.8.9 to incorporate a spare inverter for Channel 1-II of the Vital Inverters (DCN 51370).
Revision 76 (Amendment 45)	09-22-06	Revised the Bases for LCOs 3.8.7, 3.8.8 and 3.8.9 to incorporate a spare inverter for Channel 1-IV of the Vital Inverters (DCN 51370).
Revision 77 (Amendment 45)	10-10-06	Revised the Bases for LCOs 3.8.7, 3.8.8 and 3.8.9 to incorporate a spare inverter for Channel 1-I of the Vital Inverters (DCN 51370).
Revision 78 (Amendment 45)	10-13-06	Revised the Bases for LCOs 3.8.7, 3.8.8 and 3.8.9 to incorporate a spare inverter for each of the Vital Inverters (DCN 51370).
Revision 79 (Amendment 60, 61 and 64)	11-03-06	Steam Generator Narrow Range Level Indication Increased from 6% to 32% (WBN-TS-05-06) Bases Sections 3.4.5, 3.4.6, and 3.4.7.
Revision 80	11-08-06	Revised the Bases for SR 3.5.2.8 to clarify that inspection of the containment sump strainer constitutes inspection of the trash rack and the screen functions.
Revision 81 (Amendment 62)	11-15-06	Revised the Bases for SR 3.6.11.2, 3.6.11.3, and 3.6.11.4 to address the Increase Ice Weight in Ice Condenser to Support Replacement Steam Generators (WBN-TS-05-09) [SGRP]

TECHNICAL SPECIFICATION BASES - REVISION LISTING

(This listing is an administrative tool maintained by WBN Licensing and may be updated without formally revising the Technical Specification Bases Table-of-Contents)

REVISIONS	ISSUED	SUBJECT
Revision 82 (Amendment 65)	11-17-06	Steam Generator (SG) Tube Integrity (WBN-TS-05-10) [SGRP]
Revision 83	11-20-06	Updated Surveillance Requirement (SR) 3.6.6.5 to clarify that the number of unobstructed spray nozzles is defined in the design bases.
Revision 84	11-30-06	Revised Bases 3.6.9 and 3.6.15 to show the operation of the EGTS when annulus pressure is not within limits.
Revision 85	03-22-07	Revised Bases 3.6.9 and 3.6.15 in accordance with TACF 1-07-0002-065 to clarify the operation of the EGTS.
Revision 86	01-31-08	Figure 3.7.15-1 was deleted as part of Amendment 40. A reference to the figure in the Bases for LCO 3.9.9 was not deleted at the time Amendment 40 was incorporated into the Technical Specifications. Bases Revision 86 corrected this error (refer to PER 130944).
Revision 87	02-12-08	Implemented Bases change package TS-07-13 for DCN 52220-A. This DCN ties the ABI and CVI signals together so that either signal initiates the other signal.
Revision 88 (Amendment 67)	03-06-08	Technical Specification Amendment 67 increased the number of TPBARs from 240 to 400.
Revision 89 (Amendment 66)	05-01-08	Update of Bases to be consistent with the changes made to Section 5.7.2.11 of the Technical Specifications to reference the ASME Operation and Maintenance Code

Enclosure 2

WBN Technical Specifications Bases - Changed Pages

B 3.3 INSTRUMENTATION

B 3.3.6 Containment Vent Isolation Instrumentation

BASES

BACKGROUND

Containment Vent Isolation Instrumentation closes the containment isolation valves in the Containment Purge System. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. The Reactor Building Purge System may be in use during reactor operation and with the reactor shutdown.

Containment vent isolation is initiated by a safety injection (SI) signal or by manual actuation. The Bases for LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," discuss initiation of SI signals.

Redundant and independent gaseous radioactivity monitors measure the radioactivity levels of the containment purge exhaust, each of which will initiate its associated train of automatic Containment Vent Isolation upon detection of high gaseous radioactivity.

The Reactor Building Purge System has inner and outer containment isolation valves in its supply and exhaust ducts. This system is described in the Bases for LCO 3.6.3, "Containment Isolation Valves."

The plant design basis requires that when moving irradiated fuel in the Auxiliary Building and/or Containment with the Containment open to the Auxiliary Building ABSCE spaces, a signal from the spent fuel pool radiation monitors 0-RE-90-102 and 103 will initiate a Containment Ventilation Isolation (CVI) in addition to their normal function. In addition, a signal from the containment purge radiation monitors 1 -RE-90-130, and -13 1 or other CVI signal will initiate that portion of the ABI normally initiated by the spent fuel pool radiation monitors. Therefore, the containment ventilation instrumentation must remain operable when moving irradiated fuel in the Auxiliary Building if the containment air locks, penetrations, equipment hatch, etc. are open to the Auxiliary Building ABSCE spaces.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The containment isolation valves for the Reactor Building Purge System close within six seconds following the DBA. The containment vent isolation radiation monitors act as backup to the SI signal to ensure closing of the purge air system supply and exhaust valves. They are also the primary means for automatically isolating containment in the event of a fuel handling accident during shutdown. Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accidental offsite radiological doses are below 10 CFR 100 (Ref. 1) limits.

The Containment Vent Isolation instrumentation satisfies Criterion 3 of the NRC Policy Statement.

When moving irradiated fuel inside containment or in the Auxiliary Building with containment air locks or penetrations open to the Auxiliary Building ABSCE spaces, or when moving fuel in the Auxiliary Building with the containment equipment hatch open, the provisions to initiate a CVI from the spent fuel pool radiation monitors and to initiate an ABI (i.e., the portion of an ABI normally initiated by the spent fuel pool radiation monitors) from a CVI, including a CVI generated by the containment purge monitors, in the event of a fuel handling accident (FHA) must be in place and functioning. The containment equipment hatch cannot be open when moving irradiated fuel inside containment in accordance with Technical Specification 3.9.4.

The ABGTS is required to be operable during movement of irradiated fuel in the Auxiliary Building during any mode and during movement of irradiated fuel in the Reactor Building when the Reactor Building is established as part of the ABSCE boundary (see TS 3.3.8, 3.7.12, & 3.9.4). When moving irradiated fuel inside containment, at least one train of the containment purge system must be operating or the containment must be isolated. When moving irradiated fuel in the Auxiliary Building during times when the containment is open to the Auxiliary Building ABSCE spaces, containment purge can be operated, but operation of the system is not required. However, whether the containment purge system is operated or not in this configuration, all containment ventilation isolation valves and associated instrumentation must remain operable. This requirement is necessary to ensure a CVI can be accomplished from the spent fuel pool radiation monitors in the event of a FHA in the Auxiliary Building.

(continued)

BASES (continued)

LCO

The LCO requirements ensure that the instrumentation necessary to initiate Containment Vent Isolation, listed in Table 3.3.6-1, is OPERABLE.

1. Manual Initiation

The LCO requires two channels OPERABLE. The operator can initiate Containment Vent Isolation at any time by using either of two switches in the control room or from local panel(s). Either switch actuates both trains. This action will cause actuation of all components in the same manner as any of the automatic actuation signals. These manual switches also initiate a Phase A isolation signal.

The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

Each channel consists of one selector switch and the interconnecting wiring to the actuation logic cabinet.

2. Automatic Actuation Logic and Actuation Relays

The LCO requires two trains of Automatic Actuation Logic and Actuation Relays OPERABLE to ensure that no single random failure can prevent automatic actuation.

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b, SI. The applicable MODES and specified conditions for the containment vent isolation portion of the SI Function is different and less restrictive than those for the SI role. If one or more of the SI Functions becomes inoperable in such a manner that only the Containment Vent Isolation Function is affected, the Conditions applicable to the SI Functions need not be entered. The less restrictive Actions specified for inoperability of the Containment Vent Isolation Functions specify sufficient compensatory measures for this case.

(continued)

BASES

LCO (continued)

3. Containment Radiation

The LCO specifies two required channels of radiation monitors to ensure that the radiation monitoring instrumentation necessary to initiate Containment Vent Isolation remains OPERABLE.

For sampling systems, channel OPERABILITY involves more than OPERABILITY of the channel electronics. OPERABILITY may also require correct valve lineups and sample pump operation, as well as detector OPERABILITY, if these supporting features are necessary for trip to occur under the conditions assumed by the safety analyses.

Only the Allowable Value is specified for the Containment Purge Exhaust Radiation Monitors in the LCO. The Allowable Value is based on expected concentrations for a small break LOCA, which is more restrictive than 10 CFR 100 limits. The Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis in order to account for instrument uncertainties appropriate to the trip function. The actual nominal Trip Setpoint is normally still more conservative than that required by the Allowable Value. If the setpoint does not exceed the Allowable Value, the radiation monitor is considered OPERABLE.

4. Safety Injection (SI)

Refer to LCO 3.3.2, Function 1, for all initiating Functions and requirements.

APPLICABILITY

The Manual Initiation, Automatic Actuation Logic and Actuation Relays, Safety Injection, and Containment Radiation Functions are required OPERABLE in MODES 1, 2, 3, and 4, and during movement of irradiated fuel assemblies within containment. Under these conditions, the potential exists for an accident that could release significant fission product radioactivity into containment. Therefore, the Containment Vent Isolation Instrumentation must be OPERABLE in these MODES. See additional discussion in the Background and Applicable Safety Analysis sections.

(continued)

BASES

APPLICABILITY (continued)

While in MODES 5 and 6 without fuel handling in progress, the Containment the Containment Vent Isolation Instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within the limits of Reference 1.

ACTIONS

The most common cause of channel inoperability is outright failure or drift sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.6-1. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies to the failure of one containment purge isolation radiation monitor channel. Since the two containment radiation monitors are both gaseous detectors, failure of a single channel may result in loss of the redundancy. Consequently, the failed channel must be restored to OPERABLE status. The 4 hours allowed to restore the affected channel is justified by the low likelihood of events occurring during this interval, and recognition that one or more of the remaining channels will respond to most events.

(continued)

BASES

ACTIONS (continued)

B.1

Condition B applies to all Containment Vent Isolation Functions and addresses the train orientation of the Solid State Protection System (SSPS) and the master and slave relays for these Functions. It also addresses the failure of multiple radiation monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1.

If a train is inoperable, multiple channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action for the applicable Conditions of LCO 3.6.3 is met for each valve made inoperable by failure of isolation instrumentation. A Note has been added above the Required Actions to allow one train of actuation logic to be placed in bypass and to delay entering the Required Actions for up to four hours to perform surveillance testing provided the other train is OPERABLE. The 4 hour allowance is consistent with the Required Actions for actuation logic trains in LCO 3.3.2, "Engineered Safety Features Actuation System Instrumentation" and allows periodic testing to be conducted while at power without causing an actual actuation. The delay for entering the Required Actions relieves the administrative burden of entering the Required Actions for isolation valves inoperable solely due to the performance of surveillance testing on the actuation logic and is acceptable based on the OPERABILITY of the opposite train.

A Note is added stating that Condition B is only applicable in MODE 1, 2, 3, or 4.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

Condition C applies to all Containment Vent Isolation Functions and addresses the train orientation of the SSPS and the master and slave relays for these Functions. It also addresses the failure of multiple radiation monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1. If a train is inoperable, multiple channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action to place and maintain containment purge and exhaust isolation valves in their closed position is met or the applicable Conditions of LCO 3.9.4, "Containment Penetrations," are met for each valve made inoperable by failure of isolation instrumentation. The Completion Time for these Required Actions is Immediately.

A Note states that Condition C is applicable during movement of irradiated fuel assemblies within containment.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.6-1 determines which SRs apply to which Containment Vent Isolation Functions.

SR 3.3.6.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value.

Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1 (continued)

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.6.2

SR 3.3.6.2 is the performance of an ACTUATION LOGIC TEST. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and there is an intact voltage signal path to the master relay coils. This test is performed every 31 days on a STAGGERED TEST BASIS. The Surveillance interval is acceptable based on instrument reliability and industry operating experience.

SR 3.3.6.3

SR 3.3.6.3 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity. This test is performed every 31 days on a STAGGERED TEST BASIS. The Surveillance interval is acceptable based on instrument reliability and industry operating experience.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.4

A COT is performed every 92 days on each required channel to ensure the entire channel will perform the intended Function. The Frequency is based on the staff recommendation for increasing the availability of radiation monitors according to NUREG-1366 (Ref. 2). This test verifies the capability of the instrumentation to provide the containment vent system isolation. The setpoint shall be left consistent with the current unit specific calibration procedure tolerance.

SR 3.3.6.5

SR 3.3.6.5 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation mode is either allowed to function or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment that may not be operated in the design mitigation mode is prevented from operation by the SLAVE RELAY TEST circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay. This test is performed every 92 days. The Frequency is acceptable based on instrument reliability and industry operating experience.

For ESFAS slave relays which are Westinghouse type AR relays, the SLAVE RELAY TEST is performed every 18 months. The frequency is based on the relay reliability assessment presented in Reference 3. This reliability assessment is relay specific and applies only to Westinghouse type AR relays with AC coils. Note that for normally energized applications, the relays may require periodic replacement in accordance with the guidance given in Reference 3.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.6

SR 3.3.6.6 is the performance of a TADOT. This test is a check of the Manual Actuation Functions and is performed every 18 months. Each Manual Actuation Function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.).

For these tests, the relay trip setpoints are verified and adjusted as necessary. The Frequency is based on the known reliability of the Function and the redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Functions tested have no setpoints associated with them.

SR 3.3.6.7

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

2. NUREG-1366, "Improvement to Technical Specification Surveillance Requirements," December 1992.

3. WCAP-13877, Rev. 1. “Reliability Assessment of Westinghouse Type AR Relays Used as SSPS Slave Relays.” August 1998.

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B 3.3 INSTRUMENTATION

B 3.3.8 Auxiliary Building Gas Treatment (ABGTS) Actuation Instrumentation

BASES

BACKGROUND

The ABGTS ensures that radioactive materials in the fuel building atmosphere following a fuel handling accident or a loss of coolant accident (LOCA) are filtered and adsorbed prior to exhausting to the environment. The system is described in the Bases for LCO 3.7.12, "Auxiliary Building Gas Treatment System." The system initiates filtered exhaust of air from the fuel handling area, ECCS pump rooms, and penetration rooms automatically following receipt of a fuel pool area high radiation signal or a Containment Phase A Isolation signal. Initiation may also be performed manually as needed from the main control room.

High area radiation, monitored by either of two monitors, provides ABGTS initiation. Each ABGTS train is initiated by high radiation detected by a channel dedicated to that train. There are a total of two channels, one for each train. High radiation detected by any monitor or a Phase A isolation signal from the Engineered Safety Features Actuation System (ESFAS) initiates auxiliary building isolation and starts the ABGTS. These actions function to prevent exfiltration of contaminated air by initiating filtered ventilation, which imposes a negative pressure on the Auxiliary Building Secondary Containment Enclosure (ABSCE).

The plant design basis requires that when moving irradiated fuel in the Auxiliary Building and/or Containment with the Containment and/or annulus open to the Auxiliary Building ABSCE spaces, a signal from the spent fuel pool radiation monitors 0-RE-90-102 and -103 will initiate a Containment Ventilation Isolation (CVI) in addition to their normal function. In addition, a signal from the containment purge radiation monitors 1-RE-90-130, and -131 or other CVI signal will initiate that portion of the ABI normally initiated by the spent fuel pool radiation monitors. Therefore, the containment ventilation instrumentation must remain operable when moving irradiated fuel in the Auxiliary Building if the containment and/or annulus air locks, penetrations, equipment hatch, etc. are open to the Auxiliary Building ABSCE spaces.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The ABGTS ensures that radioactive materials in the ABSCE atmosphere following a fuel handling accident or a LOCA are filtered and adsorbed prior to being exhausted to the environment. This action reduces the radioactive content in the auxiliary building exhaust following a LOCA or fuel handling accident so that offsite doses remain within the limits specified in 10 CFR 100 (Ref. 1).

The ABGTS Actuation Instrumentation satisfies Criterion 3 of the NRC Policy Statement.

When moving irradiated fuel inside containment or in the Auxiliary Building with containment air locks or penetrations open to the Auxiliary Building ABSCE spaces, or when moving fuel in the Auxiliary Building with the containment equipment hatch open, the provisions to initiate a CVI from the spent fuel pool radiation monitors and to initiate an ABI (i.e., the portion of an ABI normally initiated by the spent fuel pool radiation monitors) from a CVI, including a CVI generated by the containment purge monitors, in the event of a fuel handling accident (FHA) must be in place and functioning. The containment equipment hatch cannot be open when moving irradiated fuel inside containment in accordance with Technical Specification 3.9.4.

The ABGTS is required to be operable during movement of irradiated fuel in the Auxiliary Building during any mode and during movement of irradiated fuel in the Reactor Building when the Reactor Building is established as part of the ABSCE boundary (see TS 3.3.8, 3.7.12, & 3.9.4). When moving irradiated fuel inside containment, at least one train of the containment purge system must be operating or the containment must be isolated. When moving irradiated fuel in the Auxiliary Building during times when the containment is open to the Auxiliary Building ABSCE spaces, containment purge can be operated, but operation of the system is not required. However, whether the containment purge system is operated or not in this configuration, all containment ventilation isolation valves and associated instrumentation must remain operable. This requirement is necessary to ensure a CVI can be accomplished from the spent fuel pool radiation monitors in the event of a FHA in the Auxiliary Building.

(continued)

BASES (continued)

LCO

The LCO requirements ensure that instrumentation necessary to initiate the ABGTS is OPERABLE.

1. Manual Initiation

The LCO requires two channels OPERABLE. The operator can initiate the ABGTS at any time by using either of two switches in the control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

Each channel consists of one hand switch and the interconnecting wiring to the actuation logic relays.

2. Fuel Pool Area Radiation

The LCO specifies two required Fuel Pool Area Radiation Monitors to ensure that the radiation monitoring instrumentation necessary to initiate the ABGTS remains OPERABLE. One radiation monitor is dedicated to each train of ABGTS.

For sampling systems, channel OPERABILITY involves more than OPERABILITY of channel electronics. OPERABILITY may also require correct valve lineups, sample pump operation, and filter motor operation, as well as detector OPERABILITY, if these supporting features are necessary for trip to occur under the conditions assumed by the safety analyses.

Only the Allowable Value is specified for the Fuel Pool Area Radiation Monitors in the LCO. The Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis in order to account for instrument uncertainties appropriate to the trip function. The actual nominal Trip Setpoint is normally still more conservative than that required by the Allowable Value. If the measured setpoint does not exceed the Allowable Value, the radiation monitor is considered OPERABLE.

(continued)

BASES

LCO (continued)

3. Containment Phase A Isolation

Refer to LCO 3.3.2, Function 3.a, for all initiating Functions and requirements.

APPLICABILITY

The manual ABGTS initiation must be OPERABLE in MODES 1, 2, 3, and 4 and when moving irradiated fuel assemblies in the fuel handling area, to ensure the ABGTS operates to remove fission products associated with leakage after a LOCA or a fuel handling accident. The Phase A ABGTS Actuation is also required in MODES 1, 2, 3, and 4 to remove fission products caused by post LOCA Emergency Core Cooling Systems leakage.

High radiation initiation of the ABGTS must be OPERABLE in any MODE during movement of irradiated fuel assemblies in the fuel handling area to ensure automatic initiation of the ABGTS when the potential for a fuel handling accident exists.

While in MODES 5 and 6 without fuel handling in progress, the ABGTS instrumentation need not be OPERABLE since a fuel handling accident cannot occur. See additional discussion in the Background and Applicable Safety Analysis sections.

ACTIONS

The most common cause of channel inoperability is outright failure or drift sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.8-1 in the accompanying LCO. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

(continued)

BASES

ACTIONS (continued)

A.1

Condition A applies to the actuation logic train function from the Phase A Isolation, the radiation monitor functions, and the manual function. Condition A applies to the failure of a single actuation logic train, radiation monitor channel, or manual channel. If one channel or train is inoperable, a period of 7 days is allowed to restore it to OPERABLE status. If the train cannot be restored to OPERABLE status, one ABGTS train must be placed in operation. This accomplishes the actuation instrumentation function and places the unit in a conservative mode of operation. The 7 day Completion Time is the same as is allowed if one train of the mechanical portion of the system is inoperable. The basis for this time is the same as that provided in LCO 3.7.12.

B.1.1, B.1.2, B.2

Condition B applies to the failure of two ABGTS actuation logic signals from the Phase A Isolation, two radiation monitors, or two manual channels. The Required Action is to place one ABGTS train in operation immediately. This accomplishes the actuation instrumentation function that may have been lost and places the unit in a conservative mode of operation. The applicable Conditions and Required Actions of LCO 3.7.12 must also be entered for the ABGTS train made inoperable by the inoperable actuation instrumentation. This ensures appropriate limits are placed on train inoperability as discussed in the Bases for LCO 3.7.12.

Alternatively, both trains may be placed in the emergency radiation protection mode. This ensures the ABGTS Function is performed even in the presence of a single failure.

C.1

Condition C applies when the Required Action and associated Completion Time for Condition A or B have not been met and irradiated fuel assemblies are being moved in the fuel building. Movement of irradiated fuel assemblies in the

(continued)

BASES

ACTIONS

C.1 (continued)

fuel building must be suspended immediately to eliminate the potential for events that could require ABGTS actuation. Performance of these actions shall not preclude moving a component to a safe position.

D.1 and D.2

Condition D applies when the Required Action and associated Completion Time for Condition A or B have not been met and the plant is in MODE 1, 2, 3, or 4. The plant must be brought to a MODE in which the LCO requirements are not applicable. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.8-1 determines which SRs apply to which ABGTS Actuation Functions.

SR 3.3.8.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1 (continued)

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.8.2

A COT is performed once every 92 days on each required channel to ensure the entire channel will perform the intended function. This test verifies the capability of the instrumentation to provide the ABGTS actuation. The setpoints shall be left consistent with the unit specific calibration procedure tolerance. The Frequency of 92 days is based on the known reliability of the monitoring equipment and has been shown to be acceptable through operating experience.

SR 3.3.8.3

SR 3.3.8.3 is the performance of a TADOT. This test is a check of the manual actuation functions and is performed every 18 months. Each manual actuation function is tested up to, and including, the relay coils. In some instances, the test includes actuation of the end device (e.g., pump starts, valve cycles, etc.). The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Functions tested have no setpoints associated with them.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.8.4

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

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

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

Overpressure protection is required in MODES 1, 2, 3, 4, and 5; however, in MODE 4, MODE 5, and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Cold Overpressure Mitigation System (COMS)."

The upper and lower pressure limits are based on a $\pm$ 3% tolerance. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

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

(continued)

BASES

BACKGROUND (continued)

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

APPLICABLE SAFETY ANALYSES

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

a. Uncontrolled rod withdrawal from full power;

b. Loss of reactor coolant flow;

c. Loss of external electrical load;

d. Loss of normal feedwater;

e. Loss of all AC power to station auxiliaries;

f. Locked rotor; and

g. Feedwater line break.

Detailed analyses of the above transients are contained in Reference 2. Safety valve actuation is required in events c, d, e, f, and g (above) to limit the pressure increase. Compliance with this LCO is consistent with the design bases and accident analyses assumptions.

Pressurizer safety valves satisfy Criterion 3 of the NRC Policy Statement.

LCO

The three pressurizer safety valves are set to open at the RCS design pressure (2485 psig), and within the specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits are

(continued)

BASES

LCO (continued)

based on a ± 3% tolerance. The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

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

The LCO is not applicable in MODE 4 when all RCS cold leg temperatures are ≤ 350°F or in MODE 5 because COMS is provided. Overpressure protection is not required in MODE 6 with reactor vessel head detensioned.

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

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS Overpressure Protection System. An inoperable safety

(continued)

BASES

ACTIONS

A.1 (continued)

valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. With any RCS cold leg temperatures at or below 350°F, overpressure protection is provided by the COMS System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of the ASME OM Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is ± 3% for OPERABILITY, however, the valves are reset to ± 1% during the surveillance to allow for drift.

REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, NB 7000, 1971 Edition through Summer 1973.

(continued)

BASES

REFERENCES (continued)

2. Watts Bar FSAR, Section 15.0, "Safety Analyses."

3. WCAP-7769, Rev. 1, "Topical Report on Overpressure Protection for Westinghouse Pressurized Water Reactors," June 1972.

4. American Society of Mechanical Engineers (ASME) OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

BASES

BACKGROUND

The pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORVs are pilot-operated solenoid valves that are controlled to open at a specific set pressure when the pressurizer pressure increases and close when the pressurizer pressure decreases. The PORVs may also be manually operated from the control room.

Block valves, which are normally open, are located between the pressurizer and the PORVs. The block valves are used to isolate the PORVs in case of excessive leakage or a stuck open PORV. Block valve closure is accomplished manually using controls in the control room. A stuck open PORV is, in effect, a small break loss of coolant accident (LOCA). As such, block valve closure terminates the RCS depressurization and coolant inventory loss.

The PORVs and their associated block valves may be used by plant operators to depressurize the RCS to recover from certain transients if normal pressurizer spray is not available. Additionally, the series arrangement of the PORVs and their block valves permit performance of surveillances on the valves during power operation.

The PORVs may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.

The PORVs, their block valves, and their controls are powered from the vital buses that normally receive power from offsite power sources, but are also capable of being powered from emergency power sources in the event of a loss of offsite power. Two PORVs and their associated block valves are powered from two separate safety trains (Ref. 1).

The plant has two PORVs, each having a relief capacity of 210,000 lb/hr at 2485 psig. The functional design of the PORVs is based on maintaining pressure below the Pressurizer Pressure - High reactor trip setpoint following a step reduction of 50% of full load with steam dump. In addition, the PORVs

(continued)

BASES

BACKGROUND (continued)

minimize challenges to the pressurizer safety valves and also may be used for low temperature overpressure protection (LTOP). See LCO 3.4.12, "Cold Overpressure Mitigation System (COMS)."

APPLICABLE SAFETY ANALYSES

Plant operators employ the PORVs to depressurize the RCS in response to certain plant transients if normal pressurizer spray is not available. For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes that manual operator actions are required to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray is unavailable to reduce RCS pressure. The PORVs are assumed to be used for RCS depressurization, which is one of the steps performed to equalize the primary and secondary pressures in order to terminate the primary to secondary break flow and the radioactive releases from the affected steam generator.

The PORVs are modeled in safety analyses for events that result in increasing RCS pressure for which departure from nucleate boiling ratio (DNBR), pressurizer filling, or reactor coolant saturation criteria are critical (Ref. 2). By assuming PORV actuation, the primary pressure remains below the high pressurizer pressure trip setpoint; thus, the DNBR calculation is more conservative. As such, this actuation is not required to mitigate these events, and PORV automatic operation is, therefore, not an assumed safety function.

Pressurizer PORVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation to mitigate the effects associated with an SGTR.

By maintaining two PORVs and their associated block valves OPERABLE, the single failure criterion is satisfied. An OPERABLE block valve may be either open and energized with the capability to be closed, or closed and energized with the capability to be opened, since the required safety function is accomplished by manual operation, the block valves may be OPERABLE when closed to isolate the flow path of an

(continued)

BASES

LCO (continued)

inoperable PORV that is capable of being manually cycled (e.g., as in the case of excessive PORV leakage). Similarly, isolation of an OPERABLE PORV does not render that PORV or block valve inoperable provided the relief function remains available with manual action.

An OPERABLE PORV is required to be capable of manually opening and closing and not experiencing excessive seat leakage. Excessive seat leakage although not associated with a specific acceptance criteria, exists when conditions dictate closure of block valve to limit leakage.

Satisfying the LCO helps minimize challenges to fission product barriers.

APPLICABILITY

In MODES 1, 2, and 3, the PORV and its block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. The most likely cause for a PORV small break LOCA is a result of a pressure increase transient that causes the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2. The PORVs are also required to be OPERABLE in MODES 1, 2, and 3 for manual actuation to mitigate a steam generator tube rupture event.

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the LCO is applicable in MODES 1, 2, and 3. The LCO is not applicable in MODE 4, 5, and 6 with the reactor vessel head in place when both pressure and core energy are decreased and the pressure surges become much less significant. LCO 3.4.12 addresses the PROV requirements in these MODES.

ACTIONS

A Note has been added to clarify that all pressurizer PORVs are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis).

(continued)

BASES

ACTIONS (continued)

A.1

PORVs may be inoperable and capable of being manually cycled (e.g., due to excessive seat leakage). In this condition, either the PORV must be restored or the flow path isolated within 1 hour. The associated block valve is required to be closed, but power must be maintained to t he associated block valve, since removal of power would render the block valve inoperable. This permits operation of the plant until the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition.

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that has shown that minor problems can be corrected or closure accomplished in this time period.

B.1, B.2, and B.3

If one PORV is inoperable and not capable of being manually cycled, it must be either restored or isolated by closing the associated block valve and removing the power to the associated block valve. The Completion Times of 1 hour are reasonable, based on challenges to the PORVs during this time period, and provide the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is at least one PORV that remains OPERABLE, an additional 72 hours is provided to restore the inoperable PORV to OPERABLE status. If the PORV cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D.

C.1 and C.2

If one block valve is inoperable, then it is necessary to either restore the block valve to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in manual control. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

within 1 hour, the Required Action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted a Completion Time of 72 hours to restore the inoperable block valve to OPERABLE status. The time allowed to restore the block valve is based upon the Completion Time for restoring an inoperable PORV in Condition B, since the PORVs may not be capable of mitigating an event if the inoperable block valve is not full open. If the block valve is restored within the Completion Time of 72 hours, the PORV may be restored to automatic operation. If it cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D.

D.1 and D.2

If the Required Action of Condition A, B, or C is not met, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, automatic PORV OPERABILITY may be required. See LCO 3.4.12.

E.1, E.2, E.3, and E.4

If both PORVs are inoperable and not capable of being manually cycled, it is necessary to either restore at least one valve within the Completion Time of 1 hour or isolate the flow path by closing and removing the power to the associated block valves. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system

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E.1, E.2, E.3, and E.4 (continued)

during this time and provides the operator time to correct the situation. If no PORVs are restored within the Completion Time, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, automatic PORV OPERABILITY may be required. See LCO 3.4.12.

F.1 and F.2

If both block valves are inoperable, it is necessary to either restore the block valves within the Completion Time of 1 hour, or place the associated PORVs in manual control and restore at least one block valve within 2 hours. The Completion Times are reasonable, based on the small potential for challenges to the system during this time and provide the operator time to correct the situation.

G.1 and G.2

If the Required Actions of Condition F are not met, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.12.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be opened and closed if needed. The basis for the Frequency of 92 days is the ASME OM Code (Ref. 3). If the block valve is closed to isolate a PORV that is capable of being manually cycled, the OPERABILITY of the block valve is of importance, because opening the block valve is necessary to permit the PORV to be used for manual control of reactor pressure. If the block valve is closed to isolate an inoperable PORV that is incapable of being manually cycled, the maximum Completion Time to restore the PORV and open the block valve is 72 hours, which is well within the allowable limits (25%) to extend the block valve Frequency of 92 days. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to OPERABLE status.

The Note modifies this SR by stating that it is not required to be met with the block valve closed, in accordance with the Required Action of this LCO.

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. The Frequency of 18 months is based on a typical refueling cycle and industry accepted practice.

REFERENCES

1. Regulatory Guide 1.32, "Criteria for Safety Related Electric Power Systems for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, February 1977.

2. Watts Bar FSAR, Section 15.2, "Condition II - Faults of Moderate Frequency."

3. American Society of Mechanical Engineers (ASME) OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Cold Overpressure Mitigation System (COMS)

BASES

BACKGROUND

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

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

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

This LCO provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires all safety injection pumps and all but one charging pump incapable of injection into the RCS and isolating the accumulators. The pressure relief capacity requires either two redundant RCS relief valves or a depressurized RCS and an RCS vent of sufficient size. One RCS relief valve or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

(continued)

BACKGROUND (continued)

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the safety injection (SI) actuation circuits blocked. Due to the lower pressures in the COMS MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve. If conditions require the use of more than one charging pump or safety injection pump for makeup in the event of loss of inventory, then pumps can be made available through manual actions.

The COMS for pressure relief consists of two PORVs with reduced lift settings, or one PORV and the Residual Heat Removal (RHR) suction relief valve, or a depressurized RCS and an RCS vent of sufficient size. Two RCS relief valves are required for redundancy. One RCS relief valve has adequate relieving capability to keep from overpressurization for the required coolant input capability.

PORV Requirements

As designed for the COMS, each PORV is signaled to open if the RCS pressure approaches a limit determined by the COMS actuation logic. The COMS actuation logic monitors both RCS temperature and RCS pressure and determines when a condition not acceptable in the PTLR limits is approached. The wide range RCS temperature indications are auctioneered to select the lowest temperature signal.

The lowest temperature signal is processed through a function generator that calculates a pressure limit for that temperature. The calculated pressure limit is then compared with the indicated RCS pressure from a wide range pressure channel. If the indicated pressure meets or exceeds the calculated value, a PORV is signaled to open.

The PTLR presents the PORV setpoints for COMS. The setpoints are normally staggered so only one valve opens during a low temperature overpressure transient. Having the setpoints of both valves within the limits in the PTLR ensures that the Reference 1 limits will not be exceeded in any analyzed event.

(continued)

BACKGROUND

PORV Requirements (continued)

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

RHR Suction Relief Valve Requirements

During COMS MODES, the RHR System is operated for decay heat removal and low pressure letdown control. Therefore, the RHR suction isolation valves are open in the piping from the RCS hot leg to the inlet header of the RHR pumps. While these valves are open, the RHR suction relief valve is exposed to the RCS and is able to relieve pressure transients in the RCS.

The RHR suction isolation valves must be open to make the RHR suction relief valve OPERABLE for RCS overpressure mitigation. Autoclosure interlocks are not permitted to cause the RHR suction isolation valves to close. The RHR suction relief valve is a spring loaded, bellows type water relief valve with pressure tolerances and accumulation limits established by Section III of the American Society of Mechanical Engineers (ASME) Code (Ref. 3) for Class 2 relief valves.

RCS Vent Requirements

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 COMS 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.

(continued)

BASES

BACKGROUND

RCS Vent Requirements (continued)

For an RCS vent to meet the flow capacity requirement, it requires removing a pressurizer safety valve, removing a PORV, and disabling its block valve in the open position, or opening the pressurizer manway. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 4) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, and 3, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. At about 350EF and below, overpressure prevention falls to two OPERABLE RCS 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 PTLR curves are revised, the COMS must be re-evaluated to ensure its functional requirements can still be met using the RCS relief valve method or the depressurized and vented RCS condition.

The PTLR contains the acceptance limits that define the COMS requirements. Any change to the RCS must be evaluated against the Reference 4 analyses to determine the impact of the change on the COMS acceptance limits.

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

Mass Input Type Transients

a. Inadvertent safety injection; or

b. Charging/letdown flow mismatch.

(continued)

APPLICABLE SAFETY ANALYSES (continued)

Heat Input Type Transients

a. Inadvertent actuation of pressurizer heaters;

b. Loss of RHR cooling; or

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

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

a. Rendering all safety injection pumps and all but one charging pump incapable of injection;

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

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

The Reference 4 analyses demonstrate that either one RCS relief valve or the depressurized RCS and RCS vent can maintain RCS pressure below limits when no safety injection pumps and only one centrifugal charging pump is actuated. Thus, the LCO allows only one charging pump OPERABLE during the COMS MODES. Since neither one RCS relief valve nor the RCS vent can handle the pressure transient induced from accumulator injection, when RCS temperature is low, the LCO also requires the accumulators be isolated when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

Heat Input Type Transients (continued)

The isolated accumulators must have their discharge valves closed and the valve power supply breakers fixed in their open positions. The analyses show the effect of accumulator discharge is over a narrower RCS temperature range (175EF and below) than that of the LCO (350EF and below). Fracture mechanics analyses established the temperature of COMS Applicability at 350EF.

The consequences of a small break loss of coolant accident (LOCA) in COMS MODE 4 conform to 10 CFR 50.46 and 10 CFR 50, Appendix K (Refs. 5 and 6) requirements by having a maximum of one charging pump OPERABLE and SI actuation enabled.

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the limit shown in the PTLR. The setpoints are derived by analyses that model the performance of the COMS, assuming the mass injection COMS transient of no safety injection pumps and only one centrifugal charging pump injecting into the RCS and the heat injection COMS transient of starting a RCP with the RCS 50EF colder than the secondary side. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensures the Reference 1 P/T limits will be met.

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

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

RHR Suction Relief Valve Performance

The RHR suction relief valve does not have variable pressure and temperature lift setpoints like the PORVs. Analyses must show that the RHR suction relief valve with a setpoint at or between 436.5 psig and 463.5 psig will pass flow greater than that required for the limiting COMS transient while maintaining RCS pressure less than the P/T limit curve. Assuming all relief flow requirements during the limiting COMS event, the RHR suction relief valve will maintain RCS pressure to within the valve rated lift setpoint, plus an accumulation # 3% of the rated lift setpoint.

The RHR suction relief valve inclusion and location within the RHR System does not allow it to meet single failure criteria when spurious RHR suction isolation valve closure is postulated. Also, as the RCS P/T limits are decreased to reflect the loss of toughness in the reactor vessel materials due to neutron embrittlement, the RHR suction relief valves must be analyzed to still accommodate the design basis transients for COMS.

The RHR suction relief valve is considered an active component. Thus, the failure of this valve is assumed to represent the worst case single active failure.

RCS Vent Performance

With the RCS depressurized, analyses show a vent capable of relieving > 475 gpm water flow is capable of mitigating the allowed COMS overpressure transient. The capacity of 475 gpm is greater than the flow of the limiting transient for the COMS configuration, with one centrifugal charging pump OPERABLE, maintaining RCS pressure less than the maximum pressure on the P/T limit curve.

Three vent flow paths have been identified in the RCS which could serve as pressure release (vent) paths. With one safety or PORV removed, the open line could serve as one vent path. The pressurizer manway could serve as an alternative vent path with the manway cover removed. These flow paths are capable of discharging 475 gpm at low pressure in the RCS. Thus, any one of the openings can be used for relieving the pressure to prevent violating the P/T limits.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

RCS Vent Performance (continued)

The RCS vent size will be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance. The RCS vent is passive and is not subject to active failure.

The COMS satisfies Criterion 2 of the NRC Policy Statement.

LCO

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

To limit the coolant input capability, the LCO requires no safety injection pumps and a maximum of one charging pump be capable of injecting into the RCS, and all accumulator discharge isolation valves be closed and immobilized when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

The LCO is modified by two Notes. Note 1 allows two charging pumps to be made capable of injecting for less than or equal to 1 hour during pump swap operations. One hour provides sufficient time to safely complete the actual transfer and to complete the administrative controls and surveillance requirements associated with the swap. The intent is to minimize the actual time that more than one charging pump is physically capable of injection.

Note 2 states that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.

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

a. Two RCS relief valves, as follows:

(continued)

BASES

LCO (continued)

1. Two OPERABLE PORVs; or

   A PORV is OPERABLE for COMS when its block valve is open, its lift setpoint is set to the limit required by the PTLR and testing proves its ability to open at this setpoint, and motive power is available to the valve and its control circuit.

2. One OPERABLE PORV and the OPERABLE RHR suction relief valve; or

   An RHR suction relief valve is OPERABLE for COMS when both RHR suction isolation valves are open, its setpoint is at or between 436.5 psig and 463.5 psig, and testing has proven its ability to open at this setpoint.

b. A depressurized RCS and an RCS vent.

   An RCS vent is OPERABLE when capable of relieving > 475 gpm water flow.

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

APPLICABILITY

This LCO is applicable in MODE 4, MODE 5, and 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 350EF. When the reactor vessel head is off, overpressurization cannot occur.

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

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)

BASES

ACTIONS

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

A.1 and B.1

With two or more charging pumps or any safety injection pumps capable of injecting into the RCS, RCS overpressurization is possible.

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

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

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

If isolation is needed and cannot be accomplished in 1 hour, Required Action D.1 and Required Action D.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to > 350EF, an accumulator pressure specified in WAT-D-9448 (Ref. 9) cannot exceed the COMS limits if the accumulators are fully injected.

(continued)

BASES

ACTIONS

C.1, D.1, and D.2 (continued)

Depressurizing the accumulators below the COMS limit from the PTLR also gives this protection.

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

E.1

In MODE 4 with one required RCS relief valve inoperable, the RCS relief valve must be restored to OPERABLE status within a Completion Time of 7 days. Two RCS relief valves are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

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

F.1

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

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

(continued)

BASES

ACTIONS (continued)

G.1

The RCS must be depressurized and a vent must be established within 8 hours when:

a. Both required RCS relief valves are inoperable; or

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

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

This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

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

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, no safety injection pumps and all but one charging pump are verified incapable of injecting into the RCS and the accumulator discharge isolation valves are verified closed and locked out.

The safety injection pumps and charging pump are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control. Alternative methods of low temperature overpressure protection control may be employed using at least two independent means such that a single failure or single action will not result in an injection into the RCS. This may be accomplished through the pump control switch being placed in pull to lock and at least one valve in the discharge flow path being closed, or closing discharge MOV(s) and deenergizing the motor operator(s) under administrative control, or locking closed and tagging manual valve(s) in the discharge flow path.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3 (continued)

The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment. The additional Frequency for SR 3.4.12.1 and SR 3.4.12.2 is necessary to allow time during the transition from MODE 3 to MODE 4 to make the pumps inoperable.

SR 3.4.12.4

The RCS vent capable of relieving > 475 gpm water flow is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a vent path that cannot be locked.

b. Once every 31 days for a vent path that is locked, sealed, or secured in position. A removed safety or PORV fits this category.

The passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12b.

SR 3.4.12.5

The PORV block valve must be verified open every 72 hours to provide the flow path for each required PORV to perform its function when actuated. The valve must be remotely verified open in the main control room. This Surveillance is performed if the PORV satisfies the LCO.

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.5 (continued)

The 72 hour Frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify that the PORV block valve remains open.

SR 3.4.12.6

The required RHR suction relief valve shall be demonstrated OPERABLE by verifying both RHR suction isolation valves are open and by testing it in accordance with the Inservice Testing Program. This Surveillance is only performed if the RHR suction relief valve is being used to satisfy this LCO.

Every 31 days both RHR suction isolation valves are verified locked open, with power to the valve operator removed, to ensure that accidental closure will not occur. The "locked open" valves must be locally verified in the open position with the manual actuator locked. The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve position.

SR 3.4.12.7

The COT is required to be in frequency prior to decreasing RCS temperature to ≤ 350EF or be performed within 12 hours after decreasing RCS temperature to ≤ 350EF on each required PORV to verify and, as necessary, adjust its lift setpoint. The COT will verify the setpoint is within the PTLR allowed maximum limits in the PTLR. PORV actuation could depressurize the RCS and is not required. The COT is required to be performed every 31 days when RCS temperature is ≤ 350EF with the reactor head in place.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.7 (continued)

The 12 hour allowance to meet the requirement considers the unlikelihood of a low temperature overpressure event during this time.

A Note has been added indicating that this SR is required to be met within 12 hours after decreasing RCS cold leg temperature to ≤ 350EF.

SR 3.4.12.8

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required every 18 months to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix G, "Fracture Toughness Requirements."

2. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operation."

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

4. Watts Bar FSAR, Section 15.2, "Condition II - Faults of Moderate Frequency."

5. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

6. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."

(continued)

BASES

REFERENCES (continued)

7. Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability, and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light Water Reactors,' pursuant to 10 CFR 50.44(f)."

8. American Society of Mechanical Engineers (ASME) OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."

9. Letter WAT-D-9448, "Tennessee Valley Authority Watts Bar Nuclear Plant Units 1 & 2 Revised COMS PORV Setpoints", August 27, 1994.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3), define RCS PIVs as any two normally closed valves in series within the reactor coolant pressure boundary (RCPB), which separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV Leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

The PIV leakage limit applies to each individual valve.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed accident, that could degrade the ability for low pressure injection.

The basis for this LCO is the 1975 NRC "Reactor Safety Study" (Ref. 4) that identified potential intersystem LOCAs as a significant contributor to the risk of core melt. A subsequent study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs.

(continued)

BASES

BACKGROUND (continued)

PIVs are provided to isolate the RCS from the following typically connected systems:

a. Residual Heat Removal (RHR) System;

b. Safety Injection System; and

c. Chemical and Volume Control System.

The PIVs are listed in the FSAR, Section 3.9 (Ref. 6).

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier.

APPLICABLE SAFETY ANALYSES

Reference 4 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the RHR System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the RCPB, and the subsequent pressurization of the RHR System downstream of the PIVs from the RCS. Because the low pressure portion of the RHR System is typically designed for 600 psig, overpressurization failure of the RHR low pressure line would result in a LOCA outside containment and subsequent risk of core melt.

Reference 5 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA.

RCS PIV leakage satisfies Criterion 2 of the NRC Policy Statement.

LCO

RCS PIV leakage is LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that

(continued)

BASES

LCO (continued)

increases significantly suggests that something is operationally wrong and corrective action must be taken.

The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm. The previous criterion of 1 gpm for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and resulted in higher personnel radiation exposures. A study concluded a leakage rate limit based on valve size was superior to a single allowable value.

Reference 7 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 4, valves in the RHR flow path are not required to meet the requirements of this LCO when in or during the transition to or from the RHR mode of operation.

In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.

ACTIONS

The Actions are modified by two Notes. Note 1 provides clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability, or isolation of a leaking flow path with an alternate valve may have

(continued)

BASES

ACTIONS (continued)

degraded the ability of the interconnected system to perform its safety function.

A.1 and A.2

The flow path must be isolated. Required Actions A.1 and A.2 are modified by a Note that the valve used for isolation must meet the same leakage requirements as the PIVs and must be within the RCPB.

Required Action A.1 requires that the isolation with one valve must be performed within 4 hours. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the affected system if leakage cannot be reduced. The 4 hour Completion Time allows the actions and restricts the operation with leaking isolation valves.

The 72 hour Completion Time after exceeding the limit allows for the restoration of the leaking PIV to OPERABLE status. This timeframe considers the time required to complete this Action and the low probability of a second valve failing during this period.

B.1 and B.2

If leakage cannot be reduced, or the system isolated, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The allowed Completion Times are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action A.2 is required to verify that leakage is below the specified limit and to identify each leaking valve. The

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every 18 months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The 18 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within the frequency allowed by the American Society of Mechanical Engineers (ASME) OM Code (Ref. 7), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complementary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

In addition, this Surveillance is not required to be performed on the RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Section 50.2, "Definitions--Reactor Coolant Pressure Boundary."

2. Title 10, Code of Federal Regulations, Part 50, Section 50.55a, "Codes and Standards," Subsection (c), "Reactor Coolant Pressure Boundary."

3. Title 10, Code of Federal Regulations, Part 50, Appendix A, Section V, "Reactor Containment," General Design Criterion 55, "Reactor Coolant Pressure Boundary Penetrating Containment."

4. U.S. Nuclear Regulatory Commission (NRC), "Reactor Safety Study--An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," Appendix V, WASH-1400 (NUREG-75/014), October 1975.

5. U.S. NRC, "The Probability of Intersystem LOCA: Impact Due to Leak Testing and Operational Changes," NUREG-0677, May 1980.

6. Watts Bar FSAR, Section 3.9, "Mechanical Systems and Components" (Table 3.9-17).

7. American Society of Mechanical Engineers (ASME) OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."

8. Title 10, Code of Federal Regulations, Part 50, Section 50.55a, "Codes and Standards," Subsection (g), "Inservice Inspection Requirements."

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS - Operating

BASES

BACKGROUND

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

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

b. Rod ejection accident;

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

d. Steam generator tube rupture (SGTR).

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

There are three phases of ECCS operation: injection, cold leg recirculation, and hot leg recirculation. In the injection phase, water is taken from the refueling water storage tank (RWST) and injected into the Reactor Coolant System (RCS) through the cold legs. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment sumps have enough water to supply the required net positive suction head to the ECCS pumps, suction is switched to the containment sump for cold leg recirculation. Approximately 3.0 hours after event initiation, the ECCS flow is shifted to the hot leg recirculation phase to provide a backflush, which would reduce the boiling in the top of the core and any resulting boron precipitation.

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

(continued)

BASES

BACKGROUND (continued)

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

During the injection phase of LOCA recovery, a suction header supplies water from the RWST to the ECCS pumps. Separate piping supplies each subsystem and each train within the subsystem. The discharge from the SI and RHR pumps divides and feeds an injection line to each of the RCS cold legs. Throttle valves and piping hydraulic design are set to balance the flow to the RCS and prevent pump runout. This balance ensures sufficient flow to the core to meet the analysis assumptions following a LOCA in one of the RCS cold legs.

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

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

(continued)

BASES

BACKGROUND (continued)

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

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

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

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

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46, Paragraph b (Ref. 2), will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

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

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Refs. 3 and 4). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The centrifugal charging pumps and SI pumps are credited in a small break LOCA event. This event establishes the flow and discharge head at the design point for the centrifugal charging pumps. The SGTR and MSLB events also credit the centrifugal charging pumps. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

a. A large break LOCA event, with or without loss of offsite power and with a single failure disabling one ECCS train (in the containment pressure analysis, both EDG trains are conservatively assumed to operate due to requirements for modeling full active containment heat removal system operation); and

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

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

The effects on containment mass and energy releases are accounted for in appropriate analyses (Refs. 3 and 4). The LCO ensures that an ECCS train will deliver sufficient water to match boil off rates soon enough to minimize the consequences of the core being uncovered following a large LOCA. It also ensures that the centrifugal charging and SI pumps will deliver sufficient water

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

The ECCS trains satisfy Criterion 3 of the NRC Policy Statement.

LCO

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

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

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the four cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to supply its flow to the RCS hot and cold legs.

The flow path for each train must maintain its designed independence to ensure that no single failure can disable both ECCS trains.

As indicated in Note 1, the SI flow paths may be isolated for 2 hours in MODE 3, under controlled conditions, to perform pressure isolation valve testing per SR 3.4.14.1. The flow path is readily restorable from the control room. As indicated in Note 2, operation in MODE 3 with safety injection pumps and charging pumps made incapable of injecting in order to facilitate entry into or exit from the Applicability of LCO 3.4.12, “Cold Overpressure Mitigation System (COMS)” is necessary with a COMS arming temperature at or near the MODE 3 boundary temperature of 350°F. LCO 3.4.12 requires that certain pumps be rendered incapable of injecting at and below the COMS arming temperature. When this temperature is at or near the MODE 3 boundary temperature, time is needed to make pumps incapable of injecting prior to entering the COMS Applicability, and provide time to restore the inoperable pumps to OPERABLE status on exiting the COMS Applicability.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, the ECCS OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The centrifugal charging pump performance is based on a small break LOCA, which establishes the pump performance curve and has less dependence on power. The SI pump performance requirements are based on a small break LOCA. MODE 2 and MODE 3 requirements are bounded by the MODE 1 analysis.

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

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level."

ACTIONS

A.1

With one or more trains inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 5) and is a reasonable time for repair of many ECCS components.

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

(continued)

BASES

ACTIONS

A.1 (continued)

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

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 5) has shown that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

Reference 6 describes situations in which one component, such as an RHR crossover valve, can disable both ECCS trains. With one or more component(s) inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered.

B.1 and B.2

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Misalignment of these valves could render both ECCS trains inoperable. Securing these valves in position by removal of power or by key locking the control in the correct position ensures that they cannot change position as a result of an active failure or be inadvertently misaligned. These valves are of the type, described in Reference 6, that can disable the function of both ECCS trains and invalidate the accident analyses. A 12-hour Frequency is considered reasonable in view of other administrative controls that will ensure a mispositioned valve is unlikely.

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.2.3

With the exception of the operating centrifugal charging pump, the ECCS pumps are normally in a standby, nonoperating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water by venting the ECCS pump casings and accessible suction and discharge piping high points ensures that the system will perform properly, injecting its full capacity into the RCS upon demand.* This will also prevent water hammer, pump cavitation, and pumping of noncondensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SI signal or during shutdown cooling. The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the procedural controls governing system operation. A note is added to the FREQUENCY that surveillance performance is not required for safety injection hot leg injection lines until startup from the Fall 2003 Refueling Outage. (Ref. 7)

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.3 (continued)

*For the accessible locations, UT may be substituted to demonstrate the piping is full of water. An accessible ECCS high point is defined as one that:

1) Has a vent connection installed.
2) The high point can be vented with the dose received remaining within ALARA expectations. ALARA for venting ECCS high point vents is considered to not be within ALARA expectations when the planned, intended collective dose for the activity is unjustifiably higher than industry norm, or the licensee's past experience, for this (or similar) work activity.
3) The high point can be vented with industrial safety expectations remaining within the industry norm.

SR 3.5.2.4

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

SR 3.5.2.5 and 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative control. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.7

Realignment of valves in the flow path on an SI signal is necessary for proper ECCS performance. These valves are secured in a throttled position for restricted flow to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow. The 18 month Frequency is based on the same reasons as those stated in SR 3.5.2.5 and SR 3.5.2.6.

SR 3.5.2.8

Periodic inspections of the containment sump suction inlet ensure that it is unrestricted and stays in proper operating condition. The advanced sump strainer design installed at WBN incorporates both the trash rack function and the screen function. Inspection of the advanced strainer constitutes fulfillment of the trash rack/screen inspection. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, on the need to have access to the location, and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency has been found to be sufficient to detect abnormal degradation and is confirmed by operating experience.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 35, "Emergency Core Cooling System."

2. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plant."

3. Watts Bar FSAR, Section 6.3, "Emergency Core Cooling System."

4. FSAR Bar FSAR, Section 15.0, "Accident Analysis."

5. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

6. IE Information Notice No. 87-01, "RHR Valve Misalignment Causes Degradation of ECCS in PWRs," January 6, 1987.

7. WBN License Amendment Request WBN-TS-03-11 dated April 8, 2003.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 Refueling Water Storage Tank (RWST)

BASES

BACKGROUND

The RWST supplies borated water to the Chemical and Volume Control System (CVCS) during abnormal operating conditions, to the refueling pool during refueling, and to the ECCS and the Containment Spray System during accident conditions.

The RWST supplies both trains of the ECCS and the Containment Spray System through a common supply header during the injection phase of a loss of coolant accident (LOCA) recovery. Motor operated isolation valves are provided to isolate the RWST from the ECCS once the system has been transferred to the recirculation mode. The recirculation mode is entered when pump suction is transferred to the containment sump following receipt of the RWST-Low coincident with Containment Sump Level-High signal. Use of a single RWST to supply both trains of the ECCS and Containment Spray System is acceptable since the RWST is a passive component, and passive failures are not required to be assumed to occur coincidentally with Design Basis Events until after transfer to the recirculation mode.

The switchover from normal operation to the injection phase of ECCS operation requires changing centrifugal charging pump suction from the CVCS volume control tank (VCT) to the RWST through the use of isolation valves. Each set of isolation valves is interlocked so that the VCT isolation valves will begin to close once the RWST isolation valves are fully open. Since the VCT is under pressure, the preferred pump suction will be from the VCT until the tank is isolated. This will result in a delay in obtaining the RWST borated water. The effects of this delay are discussed in the Applicable Safety Analyses section of these Bases.

During normal operation in MODES 1, 2, and 3, the safety injection (SI) and residual heat removal (RHR) pumps are aligned to take suction from the RWST.

The ECCS and Containment Spray System pumps are provided with recirculation lines that ensure each pump can maintain minimum flow requirements when operating at or near shutoff head conditions.

(continued)

BASES

BACKGROUND (continued)

When the suction for the ECCS and Containment Spray System pumps is transferred to the containment sump, the RWST flow paths must be isolated to prevent a release of the containment sump contents to the RWST, which could result in a release of contaminants to the atmosphere and the eventual loss of suction head for the ECCS pumps.

This LCO ensures that:

a. The RWST contains sufficient borated water to support the ECCS during the injection phase;

b. Sufficient water volume exists in the containment sump to support continued operation of the ECCS and Containment Spray System pumps at the time of transfer to the recirculation mode of cooling; and

c. The reactor remains subcritical following a LOCA.

Insufficient water in the RWST could result in insufficient cooling capacity when the transfer to the recirculation mode occurs. Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following the LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the containment.

APPLICABLE SAFETY ANALYSES

During accident conditions, the RWST provides a source of borated water to the ECCS and Containment Spray System pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown (Ref. 1). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of B 3.5.2, “ECCS-Operating;” B 3.5.3, “ECCS-Shutdown;” and B 3.6.6, “Containment Spray Systems.” These analyses are used to assess changes to the RWST in order to evaluate their effects in relation to the acceptance limits in the analyses.

The RWST must also meet volume, boron concentration, and temperature requirements for non-LOCA events. The volume is not an explicit assumption in non-LOCA events since the required volume is a small fraction of the available

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BASES

APPLICABLE SAFETY ANALYSES (continued)

volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is different from the total volume contained since, due to the design of the tank, more water can be contained than can be delivered. The minimum boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability. The maximum boron concentration is an explicit assumption in the inadvertent ECCS actuation analysis, although it is typically a nonlimiting event and the results are very insensitive to boron concentrations. The maximum temperature ensures that the amount of cooling provided from the RWST during the heatup phase of a feedline break is consistent with safety analysis assumptions; the minimum is an assumption in both the MSLB and inadvertent ECCS actuation analyses, although the inadvertent ECCS actuation event is typically nonlimiting.

The MSLB analysis has considered a delay associated with the interlock between the VCT and RWST isolation valves, and the results show that the departure from nucleate boiling design basis is met. The delay has been established as 27 seconds, with offsite power available, or 37 seconds without offsite power.

Technical Specification Surveillance Requirements 3.5.1.4, “Accumulators,” and 3.5.4.3, “RWST,” match boron concentrations to the number of tritium producing burnable absorbers rods (TPBARs) installed in the reactor core. Watts Bar is authorized to place a maximum of 400 TPBARs into the reactor in an operating cycle. Generally, TPBARs act as burnable absorber rods normally found in similar reactor core designs. However, unlike burnable absorber rods which lose their poison effects over the life of the cycle, some residual effect remains in the TPBARs at the end of the cycle. When larger amounts of excess neutron poisons (as in the case with larger loads of TPBARs) are added to a core, there is competition for neutrons from all the poison and the negative worth of each poison (including the reactor coolant system (RCS) boron) decreases. The positive reactivity insertion due to the negative moderator coefficient that occurs during the cooldown from hot full power to cold conditions following a loss of coolant accident (LOCA) must be overcome by RCS boron. Because the RCS boron is worth less, it takes a higher concentration to maintain subcriticality.

For a large break LOCA Analysis, the minimum water volume limit of 370,000 gallons and the minimum boron concentration limit is used to compute the post LOCA sump boron concentration necessary to assure subcriticality. This

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

minimum value depends on the number of TPBARs in the core as specified in the Core Operating Limits Report (COLR) for each operating cycle. The large break LOCA is the limiting case since the safety analysis assumes least negative reactivity insertion.

The upper limit on boron concentration of 3300 ppm is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg injection is to avoid boron precipitation in the core following the accident.

In the ECCS analysis, the containment spray temperature is assumed to be equal to the RWST lower temperature limit of 60°F. If the lower temperature limit is violated, the containment spray further reduces containment pressure, which decreases the rate at which steam can be vented out the break and increases peak clad temperature. The acceptable temperature range of 60°F to 105°F is assumed in the large break LOCA analysis, and the small break analysis value bounds the upper temperature limit of 105°F. The upper temperature limit of 105°F is also used in the containment OPERABILITY analysis. Exceeding the upper temperature limit will result in a higher peak clad temperature, because there is less heat transfer from the core to the injected water following a LOCA and higher containment pressures due to reduced containment spray cooling capacity. For the containment response following an MSLB, the lower limit on boron concentration and the upper limit on RWST water temperature are used to maximize the total energy release to containment.

The RWST satisfies Criterion 3 of the NRC Policy Statement.

LCO

The RWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment sump to support ECCS and Containment Spray System pump operation in the recirculation mode.

To be considered OPERABLE, the RWST must meet the water volume, boron concentration, and temperature limits established in the SRs.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, RWST OPERABILITY requirements are dictated by ECCS and Containment Spray System OPERABILITY requirements. Since both the ECCS and the Containment Spray System must be OPERABLE in MODES 1, 2, 3, and 4, the RWST must also be OPERABLE to support their operation. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level."

ACTIONS

A.1

With RWST boron concentration or borated water temperature not within limits, they must be returned to within limits within 8 hours. Under these conditions neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE condition. The 8 hour limit to restore the RWST temperature or boron concentration to within limits was developed considering the time required to change either the boron concentration or temperature and the fact that the contents of the tank are still available for injection.

B.1

With the RWST inoperable for reasons other than Condition A (e.g., water volume), it must be restored to OPERABLE status within 1 hour.

In this Condition, neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in which the RWST is not required. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting redundant trains.

C.1 and C.2

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

The RWST borated water temperature should be verified every 24 hours to be within the limits assumed in the accident analyses band. The specified temperature range is ≥ 60°F and ≤ 105°F and does not account for instrument error (Ref. 2). The 24 hour Frequency is sufficient to identify a temperature change that would approach either limit and has been shown to be acceptable through operating experience.

The SR is modified by a Note that eliminates the requirement to perform this Surveillance when ambient air temperatures are within the operating limits of the RWST. With ambient air temperatures within the band, the RWST temperature should not exceed the limits.

SR 3.5.4.2

The required minimum RWST water level is ≥370,000 gallons (value does not account for instrument error, Ref. 2). Verification every 7 days of the presence of this water volume ensures that a sufficient initial supply of water is available for injection and to support continued ECCS and Containment Spray System pump operation on recirculation. Since the RWST volume is normally stable and is protected by an alarm, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

SR 3.5.4.3

The boron concentration of the RWST should be verified every 7 days to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. Since the RWST volume is normally stable, a 7 day sampling Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

(continued)

BASES (continued)

REFERENCES

1. Watts Bar FSAR, Section 6.3, “Emergency Core Cooling System,” and Section 15.0, “Accident Analysis.”

2. Watts Bar Drawing 1-47W605-243, “Electrical Tech Spec Compliance Tables.”

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Containment Spray System

BASES

BACKGROUND

The Containment Spray System provides containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure helps reduce the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA). The Containment Spray System is designed to meet the requirements of 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal Systems," and GDC 40, "Testing of Containment Heat Removal Systems," (Ref. 1), or other documents that were appropriate at the time of licensing (identified on a plant specific basis).

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the system design basis spray coverage. Each train includes a containment spray pump, one containment spray heat exchanger, a spray header, nozzles, valves, and piping. Each train is powered from a separate Engineered Safety Feature (ESF) bus. The refueling water storage tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWST to the containment recirculation sump(s).

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

The Containment Spray System and RHR System provide a spray of subcooled borated water into the upper region of containment to limit the containment pressure and temperature during a DBA. In the recirculation mode of

(continued)

BASES

BACKGROUND (continued)

operation, heat is removed from the containment sump water by the Containment Spray System and RHR heat exchangers. Each train of the Containment Spray System, supplemented by a train of RHR spray, provides adequate spray coverage to meet the system design requirements for containment heat removal.

The Containment Spray System is actuated either automatically by a containment High-High pressure signal or manually. An automatic actuation starts the two containment spray pumps, opens the containment spray pump discharge valves, and begins the injection phase. A manual actuation of the Containment Spray System requires the operator to actuate two separate switches on the main control board to begin the same sequence. The injection phase continues until an RWST level Low-Low alarm is received. The Low-Low alarm for the RWST signals the operator to manually align the system to the recirculation mode. The Containment Spray System in the recirculation mode maintains an equilibrium temperature between the containment atmosphere and the recirculated sump water. Operation of the Containment Spray System in the recirculation mode is controlled by the operator in accordance with the emergency operation procedures.

The RHR spray operation is initiated manually, when required by the emergency operating procedures, after the Emergency Core Cooling System (ECCS) is operating in the recirculation mode. The RHR sprays are available to supplement the Containment Spray System, if required, in limiting containment pressure. This additional spray capacity would typically be used after the ice bed has been depleted and in the event that containment pressure rises above a predetermined limit.

The Containment Spray System is an ESF system. It is designed to ensure that the heat removal capability required during the post accident period can be attained.

(continued)

BASES

BACKGROUND (continued)

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

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

APPLICABLE SAFETY ANALYSES

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

The DBA analyses show that the maximum peak containment pressure of 11.01 psig results from the LOCA analysis, and is calculated to be less than the containment design pressure. The maximum peak containment atmosphere temperature results from the SLB analysis. The calculated transient containment atmosphere temperatures are acceptable for the DBA SLB.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The modeled Containment Spray System actuation from the containment analysis is based on a response time associated with exceeding the containment High-High pressure signal setpoint to achieving full flow through the containment spray nozzles. A delayed response time initiation provides conservative analyses of peak calculated containment temperature and pressure responses. The Containment Spray System total response time of 221 seconds is composed of signal delay, diesel generator startup, and system startup time.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the ECCS cooling effectiveness during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 3).

Inadvertent actuation of the Containment Spray System is evaluated in the analysis, and the resultant reduction in containment pressure is calculated. The maximum calculated steady state pressure differential relative to the Shield Building annulus is 1.4 psid, which is below the containment design external pressure load of 2.0 psid.

The Containment Spray System satisfies Criterion 3 of the NRC Policy Statement.

LCO

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

Each containment spray train typically includes a spray pump, header, valves, a heat exchanger, nozzles, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an ESF actuation signal and automatically transferring suction to the containment sump.

(continued)

BASES

LCO (continued)

Each RHR spray train includes a pump, header, valves, a heat exchanger, nozzles, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the containment sump and supplying flow to the spray header.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the Containment Spray System. A Note has been added which states the RHR spray trains are not required in MODE 4. The containment spray system does not require supplemental cooling from the RHR spray in MODE 4.

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

ACTIONS

A.1 and B.1

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

C.1 and C.2

If the affected containment spray train and/or RHR spray train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Times are reasonable, based

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time and is reasonable when considering that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment of manual, power operated, and automatic valves, excluding check valves, in the Containment Spray System provides assurance that the proper flow path exists for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since they were verified in the correct position prior to being secured. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned, are in the correct position.

SR 3.6.6.2

Verifying that each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by the American Society of Mechanical Engineers (ASME) OM Code. (Ref. 4). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on bypass flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.3 and SR 3.6.6.4

These SRs require verification that each automatic containment spray valve actuates to its correct position and each containment spray pump starts upon receipt of an actual or simulated containment spray actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative control. Containment spray pump start verification may be performed by testing breaker actuation without pump start (breaker is racked out in its "test position") and observation of the local or remote pump start lights (breaker energization light). The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillances when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The surveillance of containment sump isolation valves is also required by SR 3.6.6.3. A single surveillance may be used to satisfy both requirements.

SR 3.6.6.5

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle required by the design bases is unobstructed and that spray coverage of the containment during an accident is not degraded. Because of the passive design of the nozzle, a test at the first refueling and at 10 year intervals are considered adequate to detect obstruction of the spray nozzles.

SR 3.6.6.6

The Surveillance descriptions from Bases 3.5.2 for SR 3.5.2.2 and 3.5.2.4 apply as applicable to the RHR spray system.

(continued)

BASES (continued)

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criterion (GDC) 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal System," GDC 40, "Testing of Containment Heat Removal Systems, and GDC 50, "Containment Design Basis."

2. Watts Bar FSAR, Section 6.2, "Containment Systems."

3. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."

4. American Society of Mechanical Engineers (ASME) OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

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

Five MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the FSAR, Section 10.3.2 (Ref. 1). The MSSVs must have sufficient capacity to limit the secondary system pressure to ≤ 110% of the steam generator design pressure in order to meet the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-2 in the accompanying LCO, so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine reactor trip.

APPLICABLE SAFETY ANALYSES

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

The events that challenge the relieving capacity of the MSSVs, and thus Main Steam System pressure, are those characterized as decreased heat removal events, which are presented in the FSAR, Section 15.2 and 15.4 (Ref. 3). Of these, the full power loss of normal feedwater is the limiting AOO. The transient response for this event presents no hazard to the integrity of the RCS or the Main Steam System.

Following the loss of continued subcooled feedwater addition, the primary and secondary-side temperatures increase, resulting in a secondary-side pressure increase that proceeds all the way up to the lowest safety valve setpoint. The receipt of a low-low steam generator water level reactor trip signal releases the RCCAs to fall into the core and provides a turbine trip signal. Following the turbine trip, all MSSVs are briefly actuated while rods fall into the core and

(continued)

APPLICABLE ANALYSES (continued)

the hot leg inventory is purged of hot reactor coolant. After the SAFETY core is shutdown, the required relief capacity is reduced, and one MSSV per steam generator remains open during the remainder of the transient.

In addition to the decreased heat removal events, reactivity insertion events may also challenge the relieving capacity of the MSSVs. The uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power event is characterized by an increase in core power and steam generation rate until reactor trip occurs when either the Overtemperature ΔT or Power Range Neutron Flux-High setpoint is reached. Steam flow to the turbine will not increase from its initial value for this event. The increased heat transfer to the secondary side causes an increase in steam pressure and may result in opening of the MSSVs prior to reactor trip, assuming no credit for operation of the atmospheric or condenser steam dump valves. The FSAR Section 15.2 safety analysis of the RCCA bank withdrawal at power event for a range of initial core power levels demonstrates that the MSSVs are capable of preventing secondary side overpressurization for this AOO.

The FSAR safety analyses discussed above assume that all of the MSSVs for each steam generator are OPERABLE. If there are inoperable MSSV(s), it is necessary to limit the primary system power during steady-state operation and AOOs to a value that does not result in exceeding the combined steam flow capacity of the turbine (if available) and the remaining OPERABLE MSSVs. The required limitation on primary system power necessary to prevent secondary system overpressurization may be determined by system transient analyses or conservatively arrived at by a simple heat balance calculation. In some circumstances it is necessary to limit the primary side heat generation that can be achieved during an AOO by reducing the setpoint of the Power Range Neutron Flux-High reactor trip function. For example, if more than one MSSV on a single steam generator is inoperable, an uncontrolled RCCA bank withdrawal at power event occurring from a partial power level may result in an increase in reactor power that exceeds the combined steam flow capacity of the turbine and the remaining OPERABLE MSSVs. Thus, for multiple inoperable MSSVs on the same steam generator it is necessary to prevent this power increase by lowering the Power Range Neutron Flux-High setpoint to an appropriate value.

The MSSVs are assumed to have two active failure modes. The active failure modes are spurious opening, and failure to reclose once opened.

The MSSVs satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

The accident analysis requires that five MSSVs per steam generator be OPERABLE to provide overpressure protection for design basis transients occurring at 100.6% RTP. The LCO requires that five MSSVs per steam generator be OPERABLE in compliance with Reference 2 and the DBA analysis.

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

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

APPLICABILITY

In MODES 1, 2, and 3, five MSSVs per steam generator are required to be OPERABLE to prevent Main Steam System overpressuration.

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

ACTIONS

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

With one or more MSSVs inoperable, action must be taken so that the available MSSV relieving capacity meets Reference 2 requirements.

Operation with less than all five MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator.

(continued)

BASES

ACTIONS (continued)

A.1

In the case of only a single inoperable MSSV on one or more steam generators a reactor power reduction alone is sufficient to limit primary side heat generation such that overpressurization of the secondary side is precluded for any RCS heatup event. Furthermore, for this case there is sufficient total steam flow capacity provided by the turbine and remaining OPERABLE MSSVs to preclude overpressuration in the event of an increased reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. Therefore, Required Action A.1, requires an appropriate reduction in reactor power within 4 hours.

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined using a conservative heat balance between the reactor coolant system heat generation and the steam relief through the OPERABLE MSSVs, as shown below and described in the attachment to Reference 6:

$$\textit{Allowable THERMAL POWER Level (\%)} = 100\frac{4\,w_s h_{fg}}{QK}$$

where: w_s = Minimum total steam relief capacity of the OPERABLE MSSVs on any one steam generator, in lbm/sec.

h_{fg} = heat of vaporization at the highest MSSV full-open pressure, in Btu/lbm.

Q = NSSS power rating of the plant (includes reactor coolant pump heat) in MWt.

K = Unit conversion factor: 947.82 Btu/sec/MWt.

Note: The values in Specification 3.7.1 include an allowance for instrument and channel uncertainties to the allowable RTP obtained with this algorithm.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

In the case of multiple inoperable MSSVs on one or more steam generators, with a reactor power reduction alone there may be insufficient total steam flow capacity provided by the turbine and remaining OPERABLE MSSVs to preclude overpressurization in the event of an increased reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. The 4 hour Completion Time for Required Action B.1 is consistent with A.1. An additional 32 hours is allowed in Required Action B.2 to reduce the setpoints. The Completion Time of 36 hours is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction, operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined using a conservative heat balance calculation as described above (Action A.1) and in the attachment to Reference 6. The values in Specification 3.7.1 include an allowance for instrument and channel uncertainties to the allowable RTP obtained with this algorithm.

Required Action B.2 is modified by a Note, indicating that the Power Range Neutron Flux-High reactor trip setpoint reduction is only required in MODE 1. In MODES 2 and 3 the reactor protection system trips specified in LCO 3.3.1, “Reactor Trip System Instrumentation,” provide sufficient protection.

C.1 and C.2

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

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME OM Code (Ref. 4) requires that safety and relief valve tests be

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

performed as follows:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting); and

d. Compliance with owner's seat tightness criteria;

The ASME OM Code requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. Additional test frequency requirements apply during the initial five year period. The ASME OM Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift. The lift settings, according to Table 3.7.1-2 correspond to ambient conditions of the valve at nominal operating temperature and pressure.

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

REFERENCES

1. Watts Bar FSAR, Section 10.3, "Main Steam Supply System."

2. American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, Section III, Article NC-7000, "Overpressure Protection," Class 2 Components.

3. Watts Bar FSAR, Section 15.2, "Condition II - Faults of Moderate Frequency," and Section 15.4, "Condition IV - Limiting Faults."

4. American Society of Mechanical Engineers (ASME) OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."

5. NRC Information Notice 94-60, “Potential Overpressurization of the Main Steam System,” August 22, 1994.

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

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

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

The MSIVs close on a main steam isolation signal generated by either low steam line pressure, high negative steam pressure rate (below P-11), or high-high containment pressure. The MSIVs fail closed on loss of control or actuation power.

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

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

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, discussed in the FSAR, Section 6.2 (Ref. 2). It is also affected by the accident analysis of the SLB events presented in the FSAR, Section 15.4.2.1 (Ref. 3). The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limiting case for the containment analysis is the SLB inside containment, with a loss of offsite power following turbine trip, and failure of the MSIV on the affected steam generator to close. At lower powers, the steam generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the containment. Due to reverse flow and failure of the MSIV to close, the additional mass and energy in the steam headers downstream from the other MSIV contribute to the total release. With the most reactive rod cluster control assembly assumed stuck in the fully withdrawn position, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the boric acid injection delivered by the Emergency Core Cooling System.

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

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

a. An HELB inside containment. In order to maximize the mass and energy release into containment, the analysis assumes that the MSIV in the affected steam generator remains open. For this accident scenario, steam is discharged into containment from all steam generators until the remaining MSIVs close. After MSIV closure, steam is discharged into containment only from the affected steam generator and from the residual steam in the main steam header downstream of the closed

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

MSIVs in the unaffected loops. Closure of the MSIVs isolates the break from the unaffected steam generators.

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

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

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

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

The MSIVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

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

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 (Ref. 4) limits or the NRC staff approved licensing basis.

APPLICABILITY

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

(continued)

BASES

APPLICABILITY (continued)

In MODE 4, normally most of the MSIVs are closed, and the steam generator energy is low.

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

ACTIONS

A.1

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

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

B.1

If the MSIV cannot be restored to OPERABLE status within 8 hours, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in MODE 2 within 6 hours and Condition C would be entered. The Completion Times are reasonable, based on operating experience, to reach MODE 2 and to close the MSIVs in an orderly manner and without challenging plant systems.

C.1 and C.2

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

Since the MSIVs are required to be OPERABLE in MODES 2 and 3, the inoperable MSIVs may either be restored to OPERABLE status or closed.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

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

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

D.1 and D.2

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

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that MSIV closure time is ≤ 6.0 seconds on an actual or simulated actuation signal. The MSIV closure time is assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1 (continued)

The Frequency is in accordance with the Inservice Testing Program or 18 months. The 18 month Frequency for valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

This test is conducted in MODE 3 with the unit at operating temperature and pressure, as discussed in Reference 5 exercising requirements. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

REFERENCES

1. Watts Bar FSAR, Section 10.3, "Main Steam Supply System."

2. Watts Bar FSAR, Section 6.2, "Containment Systems."

3. Watts Bar FSAR, Section 15.4.2.1, "Major Rupture of a Main Steam Line."

4. 10 CFR 100.11.

5. American Society of Mechanical Engineers (ASME) OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Isolation Valves (MFIVs) and Main Feedwater Regulation Valves (MFRVs) and Associated Bypass Valves

BASES

BACKGROUND

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

The MFIVs and associated bypass valves, isolate the nonsafety-related portions from the safety related portions of the system. In the event of a secondary side pipe rupture inside containment, the valves limit the quantity of high energy fluid that enters containment through the break.

One MFIV and one MFRV are located on each 16 inch MFW line. One bypass MFRV and one bypass MFIV are located on a smaller 6 inch startup flow feedwater line. Both the MFIV and bypass MFIV are located in the main steam valve vault close to containment.

(continued)

BASES

BACKGROUND (continued)

The MFIVs and associated bypass valves, and MFRVs and associated bypass valves, close on receipt of a T_{avg} Low coincident with reactor trip (P-4), safety injection signal, or steam generator water level-high high signal. They may also be closed manually except for the bypass MFIV which has no handswitch. In addition to the MFIVs and associated bypass valves, and the MFRVs and associated bypass valves, a check valve on the 16-inch MFW line is located just outside containment in the main steam valve vault. The check valve terminates flow from the steam generator for breaks upstream of the check valve.

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

APPLICABLE SAFETY ANALYSES

The design basis of the MFIVs and MFRVs and associated bypass valves is established by the analyses for the large SLB. It is also influenced by the accident analysis for the large FWLB. Closure of the MFIVs and associated bypass valves, or MFRVs and associated bypass valves, may also be relied on to mitigate an SLB for core response analysis and excess feedwater event.

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

The MFIVs and MFRVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

This LCO ensures that the MFIVs, MFRVs, and their associated bypass valves will isolate MFW flow to the steam generators, following an FWLB or SLB. The MFIVs and bypass MFIVs will also isolate the nonsafety related portions from the safety related portions of the system.

This LCO requires that four MFIVs and associated bypass valves and four MFRVs and associated bypass valves be OPERABLE.

(continued)

BASES

LCO (continued)

The MFIVs and MFRVs and the associated bypass valves are considered OPERABLE when isolation times are within limits and they close on an isolation actuation signal.

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

APPLICABILITY

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

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

ACTIONS

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

A.1 and A.2

With one MFIV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

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

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

B.1 and B.2

With one MFRV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

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

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

(continued)

BASES

ACTIONS (continued)

C.1

With one MFIV or MFRV bypass valve in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status within 72 hours. The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

D.1

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

E.1

With two bypass valves in the same flow path inoperable, there may be no redundant system to operate automatically and perform the required safety function. Under these conditions, at least one valve in the flow path must be restored to OPERABLE status within 8 hours. The Completion Time of 8 hours is consistent with Condition D.

(continued)

BASES

ACTIONS (continued)

F.1 and F.2

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

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFIV, MFRV, and associated bypass valves is ≤ 6.5 seconds on an actual or simulated actuation signal. The MFIV and MFRV closure times are assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. These valves should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. This is consistent with the American Society of Mechanical Engineers (ASME) OM Code (Ref. 2), quarterly stroke requirements during operation in MODES 1 and 2.

The Frequency for this SR is in accordance with the Inservice Testing Program or 18 months. The 18 month Frequency for valve closure is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 10.4.7, "Condensate and Feedwater Systems."

2. American Society of Mechanical Engineers (ASME) OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply. The AFW pumps take suction from the condensate storage tank (CST) (LCO 3.7.6) and pump to the steam generator secondary side via separate connections to the main feedwater (MFW) bypass line piping. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1) or atmospheric dump valves (LCO 3.7.4). If the main condenser is available, steam may be released via the steam dump valves and recirculated to the CST.

The AFW System consists of two motor driven AFW pumps and one steam turbine driven pump configured into three trains. Each motor driven pump provides 410 gpm of AFW flow, and the turbine driven pump provides 720 gpm to the steam generators, as assumed in the accident analysis. The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system. Each motor driven AFW pump is powered from an independent Class 1E power supply and feeds two steam generators. The steam turbine driven AFW pump receives steam from one of two main steam lines upstream of the main steam isolation valves. Each of the steam feed lines will supply 100% of the requirements of the turbine driven AFW pump.

The AFW System is capable of supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions, however, the Main Feedwater System will normally perform these functions.

The turbine driven AFW pump supplies a common header capable of feeding all steam generators. One pump at full flow is sufficient to remove decay heat and cool the unit to residual heat removal (RHR) entry conditions. Thus, the requirement for diversity in motive power sources for the AFW System is met.

(continued)

BASES

BACKGROUND (continued)

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

The AFW System actuates automatically on steam generator water level - low-low by the ESFAS (LCO 3.3.2). The motor driven pumps start on a two-out-of-three low-low level signal in any steam generator and the turbine driven pump starts on a two-out-of-three low-low level signal in any two steam generators. The system also actuates on loss of offsite power, safety injection, and trip of both turbine-driven MFW pumps.

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

APPLICABLE SAFETY ANALYSES

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

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

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

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

a. Feedwater Line Break (FWLB); and

b. Loss of MFW.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

In addition, the minimum available AFW flow and system characteristics are serious considerations in the analysis of a small break loss of coolant accident (LOCA).

The AFW System design is such that it can perform its function following an FWLB between the MFW check valves and the steam generators, combined with a loss of offsite power following turbine trip, and a single active failure of the steam turbine driven AFW pump. One motor driven AFW pump would deliver to the faulted steam generator. Sufficient flow would be delivered to the intact steam generators by the redundant AFW pump.

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

Each motor-driven auxiliary feedwater pump (one Train A and one Train B) supplies flow paths to two steam generators. Each flow path contains automatic air-operated level control valves (LCVs). The LCVs have the same train designation as the associated pump and are provided trained air. The turbine-driven auxiliary feedwater pump supplies flow paths to all four steam generators. Each of these flow paths contains an automatic air-operated LCV, two of which are designated as Train A, receive A-train air and provide flow to the same steam generators that are supplied by the B-train motor-driven auxiliary feedwater pump. The remaining two LCVs are designated as Train B, receive B-train air, and provide flow to the same steam generators that are supplied by the A-train motor-driven pump. This design provides the required redundancy to ensure that at least two steam generators receive the necessary flow assuming any single failure. It can be seen from the description provided above that the loss of a single train of air (A or B) will not prevent the auxiliary feedwater system from performing its intended safety function and is no more severe than the loss of a single auxiliary feedwater pump. Therefore, the loss of a single train of auxiliary air only affects the capability of a single motor-driven auxiliary feedwater pump because the turbine-driven pump is still capable of providing flow to the two steam generators that are separated from the other motor-driven pump.

The AFW System satisfies the requirements of Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

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

The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the steam generators are OPERABLE. This requires that the two motor driven AFW pumps be OPERABLE in two diverse paths, each supplying AFW to separate steam generators. The turbine driven AFW pump is required to be OPERABLE with redundant steam supplies from each of two main steam lines upstream of the MSIVs, and shall be capable of supplying AFW to any of the steam generators. The piping, valves, instrumentation, and controls in the required flow paths also are required to be OPERABLE.

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

APPLICABILITY

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

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

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

(continued)

BASES (continued)

ACTIONS

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

A.1

If one of the two steam supplies to the turbine driven AFW train is inoperable, action must be taken to restore OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

a. The redundant OPERABLE steam supply to the turbine driven AFW pump;

b. The availability of redundant OPERABLE motor driven AFW pumps; and

c. The low probability of an event occurring that requires the inoperable steam supply to the turbine driven AFW pump.

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

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

B.1

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

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

(continued)

BASES

ACTIONS

B.1 (continued)

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

C.1 and C.2

When Required Action A.1 or B.1 cannot be completed within the required Completion Time, or if two AFW trains are inoperable in MODE 1, 2, or 3, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 18 hours.

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

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

D.1

If all three AFW trains are inoperable in MODE 1, 2, or 3, the plant is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety related equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one AFW train to OPERABLE status.

(continued)

BASES

ACTIONS

D.1 (continued)

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

E.1

In MODE 4, either the reactor coolant pumps or the RHR loops can be used to provide forced circulation. This is addressed in LCO 3.4.6, "RCS Loops - MODE 4." With one required AFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status. The immediate Completion Time is consistent with LCO 3.4.6. Automatic actuation of AFW is not required in MODE 4, therefore, AFW/ERCW interface valves are not required to be in service.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW System water and steam supply flow paths provides assurance that the proper flow paths exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by the American Society of Mechanical Engineers (ASME) OM Code (Ref. 2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME OM Code (Ref. 2) (only required at 3 month intervals) satisfies this requirement. The 31 day Frequency on a STAGGERED TEST BASIS results in testing each pump once every 3 months, as required by Reference 2.

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there may be insufficient steam pressure to perform the test.

SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative control. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is acceptable based on operating experience and the design reliability of the equipment. This SR is modified by a Note that states that the SR is not required in MODE 4. MODE 4 does not require automatic activation of the AFW because there is a sufficient time frame for operator action. This is based on the fact that even at 0% power (MODE 3) there is approximately a 10 minute trip delay before actuation of the AFW system to allow for operator action. In MODE 4 the heat removal requirements would be less providing more time for operator action.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.4

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an ESFAS by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

This SR is modified by two Notes. Note 1 indicates that the SR be deferred until suitable test conditions are established. This deferral is required because there may be insufficient steam pressure to perform the test. Note 2 states that the SR is not required in MODE 4. MODE 4 does not require automatic activation of the AFW because there is a sufficient time frame for operator action. This is based on the fact that even at 0% power (MODE 3) there is approximately a 10 minute trip delay before actuation of the AFW system to allow for operator action. In MODE 4 the heat removal requirements would be less providing more time for operator action.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.5

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

REFERENCES

1. Watts Bar FSAR, Section 10.4.9, "Auxiliary Feedwater System."

2. American Society of Mechanical Engineers (ASME) OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."

B 3.7 PLANT SYSTEMS

B 3.7.12 Auxiliary Building Gas Treatment System (ABGTS)

BASES

BACKGROUND

The ABGTS filters airborne radioactive particulates from the area of the fuel pool following a fuel handling accident and from the area of active Unit 1 ECCS components and Unit 1 penetration rooms following a loss of coolant accident (LOCA).

The ABGTS consists of two independent and redundant trains. Each train consists of a heater, a prefilter, moisture separator, a high efficiency particulate air (HEPA) filter, two activated charcoal adsorber sections for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case the main HEPA filter bank fails. The downstream HEPA filter is not credited in the analysis. The system initiates filtered ventilation of the Auxiliary Building Secondary Containment Enclosure (ABSCE) exhaust air following receipt of a Phase A containment isolation signal or a high radiation signal from the spent fuel pool area.

The ABGTS is a standby system, not used during normal plant operations. During emergency operations, the ABSCE dampers are realigned and ABGTS fans are started to begin filtration. Air is exhausted from the Unit 1 ECCS pump rooms, Unit 1 penetration rooms, and fuel handling area through the filter trains. The prefilters or moisture separators remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The plant design basis requires that when moving irradiated fuel in the Auxiliary Building and/or Containment with the Containment open to the Auxiliary Building ABSCE spaces, a signal from the spent fuel pool radiation monitors 0-RE-90-102 and -103 will initiate a Containment Ventilation Isolation (CVI) in addition to their normal function. In addition, a signal from the containment purge radiation monitors 1-RE-90-130, and -131 or other CVI signal will initiate that portion of the ABI normally initiated by the spent fuel pool radiation monitors. Therefore, the containment ventilation instrumentation must remain operable when moving irradiated fuel in the Auxiliary Building if the containment air locks, penetrations, equipment hatch, etc. are open to the Auxiliary Building ABSCE spaces. In addition, the ABGTS must remain operable if these containment penetrations are open to the Auxiliary Building during movement of irradiated fuel inside containment.

The ABGTS is discussed in the FSAR, Sections 6.5.1, 9.4.2, 15.0, and 6.2.3 (Refs. 1, 2, 3, and 4, respectively).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The ABGTS design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident. The analysis of the fuel handling accident, given in Reference 3, assumes that all fuel rods in an assembly are damaged. The analysis of the LOCA assumes that radioactive materials leaked from the Emergency Core Cooling System (ECCS) are filtered and adsorbed by the ABGTS. The DBA analysis of the fuel handling accident assumes that only one train of the ABGTS is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the one remaining train of this filtration system. The amount of fission products available for release from the ABSCE is determined for a fuel handling accident and for a LOCA. The assumptions and the analysis for a fuel handling accident follow the guidance provided in Regulatory Guide 1.25 (Ref. 5) and NUREG/CR-5009 (Ref. 11). The assumptions and analysis for a LOCA follow the guidance provided in Regulatory Guide 1.4 (Ref. 6).

The ABGTS satisfies Criterion 3 of the NRC Policy Statement.

When moving irradiated fuel inside containment or in the Auxiliary Building with containment air locks or penetrations open to the Auxiliary Building ABSCE spaces, or when moving fuel in the Auxiliary Building with the containment equipment hatch open, the provisions to initiate a CVI from the spent fuel pool radiation monitors and to initiate an ABI (i.e., the portion of an ABI normally initiated by the spent fuel pool radiation monitors) from a CVI, including a CVI initiated by the containment purge monitors, in the event of a fuel handling accident (FHA) must be in place and functioning. The containment equipment hatch cannot be open when moving irradiated fuel inside containment in accordance with Technical Specification 3.9.4.

The ABGTS is required to be operable during movement of irradiated fuel in the Auxiliary Building during any mode and during movement of irradiated fuel in the Reactor Building when the Reactor Building is established as part of the ABSCE boundary (see TS 3.3.8, 3.7.12, & 3.9.4). When moving irradiated fuel inside containment, at least one train of the containment purge system must be operating or the containment must be isolated. When moving irradiated fuel in the Auxiliary Building during times when the containment is open to the Auxiliary Building ABSCE spaces, containment purge can be operated, but operation of the system is not required. However, whether the containment purge system is operated or not in this configuration, all containment ventilation isolation valves and associated instrumentation must remain operable. This requirement is necessary to ensure a CVI can be accomplished from the spent fuel pool radiation monitors in the event of a FHA in the Auxiliary Building.

(continued)

BASES (continued)

LCO

Two independent and redundant trains of the ABGTS are required to be OPERABLE to ensure that at least one train is available, assuming a single failure that disables the other train, coincident with a loss of offsite power. Total system failure could result in the atmospheric release from the ABSCE exceeding the 10 CFR 100 (Ref. 7) limits in the event of a fuel handling accident or LOCA.

The ABGTS is considered OPERABLE when the individual components necessary to control exposure in the fuel handling building are OPERABLE in both trains. An ABGTS train is considered OPERABLE when its associated:

a. Fan is OPERABLE;

b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration function; and

c. Heater, moisture separator, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

APPLICABILITY

In MODE 1, 2, 3, or 4, the ABGTS is required to be OPERABLE to provide fission product removal associated with ECCS leaks due to a LOCA and leakage from containment and annulus.

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

During movement of irradiated fuel in the fuel handling area, the ABGTS is required to be OPERABLE to alleviate the consequences of a fuel handling accident. See additional discussion in the Background and Applicable Safety Analysis sections.

(continued)

BASES (continued)

ACTIONS

A.1

With one ABGTS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this period, the remaining OPERABLE train is adequate to perform the ABGTS function. The 7 day Completion Time is based on the risk from an event occurring requiring the inoperable ABGTS train, and the remaining ABGTS train providing the required protection.

B.1 and B.2

In MODE 1, 2, 3, or 4, when Required Action A.1 cannot be completed within the associated Completion Time, or when both ABGTS trains are inoperable, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in MODE 3 within 6 hours, and in MODE 5 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

When Required Action A.1 cannot be completed within the required Completion Time, during movement of irradiated fuel assemblies in the fuel handling area, the OPERABLE ABGTS train must be started immediately or fuel movement suspended. This 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 placed in operation, this action requires suspension of fuel movement, which precludes a fuel handling accident. This does not preclude the movement of fuel assemblies to a safe position.

(continued)

BASES

ACTIONS

D.1

When two trains of the ABGTS are inoperable during movement of irradiated fuel assemblies in the fuel handling area, action must be taken to place the unit in a condition in which the LCO does not apply. Action must be taken immediately to suspend movement of irradiated fuel assemblies in the fuel handling area. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

Standby systems should be checked periodically to ensure that they function properly. As the environmental and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system.

Monthly heater operation dries out any moisture accumulated in the charcoal from humidity in the ambient air. The system must be operated for ≥ 10 continuous hours with the heaters energized. The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available.

SR 3.7.12.2

This SR verifies that the required ABGTS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The ABGTS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 8). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.3

This SR verifies that each ABGTS train starts and operates on an actual or simulated actuation signal. The 18 month Frequency is consistent with Reference 8.

SR 3.7.12.4

This SR verifies the integrity of the ABSCE. The ability of the ABSCE to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the ABGTS. During the post accident mode of operation, the ABGTS is designed to maintain a slight negative pressure in the ABSCE, to prevent unfiltered LEAKAGE. The ABGTS is designed to maintain a negative pressure between -0.25 and -0.5 inches water gauge (value does not account for instrument error, Ref. 10) with respect to atmospheric pressure at a nominal flow rate ≥ 9300 and ≤ 9900 cfm. The Frequency of 18 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 9).

An 18 month Frequency (on a STAGGERED TEST BASIS) is consistent with Reference 8.

REFERENCES

1. Watts Bar FSAR, Section 6.5.1, "Engineered Safety Feature (ESF) Filter Systems."

2. Watts Bar FSAR, Section 9.4.2, "Fuel Handling Area Ventilation System."

3. Watts Bar FSAR, Section 15.0, "Accident Analysis."

4. Watts Bar FSAR, Section 6.2.3, "Secondary Containment Functional Design."

(continued)

REFERENCES (continued)

5. Regulatory Guide 1.25, March 1972, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

6. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors."

7. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

8. Regulatory Guide 1.52 (Rev. 2), "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Power Plants."

9. NUREG-0800, Section 6.5.1, "Standard Review Plan," Rev. 2, "ESF Atmosphere Cleanup System," July 1981.

10. Watts Bar Drawing 1-47W605-242, "Electrical Tech Spec Compliance Tables."

11. NUREG/CR-5009, “Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors,” U. S. Nuclear Regulatory Commission, February 1988.

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B 3.9 REFUELING OPERATIONS

B 3.9.8 Reactor Building Purge Air Cleanup Units

BASES

BACKGROUND

The Reactor Building Purge Air Cleanup Units are an engineered safety feature of the Reactor Building Purge Ventilation System which is a non-safety feature ventilation system. The air cleanup units contain prefilters, HEPA filters, 2-inch-thick charcoal adsorbers, housings and ductwork. Anytime fuel handling operations are being carried on inside the primary containment, either the containment ventilation will be isolated or the Reactor Building Purge air cleanup units will be OPERABLE (Ref. 1).

`

The Reactor Building Purge Ventilation System provides mechanical ventilation of the primary containment, the instrument room located within the containment, and the annulus. The system is designed to supply fresh air for breathing and contamination control to allow personnel access for maintenance and refueling operations. The exhaust air is filtered by the Reactor Building Purge Air Cleanup Units to limit the release of radioactivity to the environment.

The containment upper and lower compartments are purged with fresh air by the Reactor Building Purge Ventilation System before occupancy. The annulus can be purged with fresh air during reactor shutdown or at times when the annulus vacuum control system of the Emergency Gas Treatment System is shut down. The instrument room is purged with fresh air during operation of the Reactor Building Purge Ventilation System or is separately purged by the Instrument Room Purge Subsystem. All purge ventilation functions are non-safety related.

The Reactor Building Purge Ventilation System is sized to provide adequate ventilation for personnel to perform work inside the primary containment and the annulus during all normal operations. In the event of a fuel handling accident, the Reactor Building Purge Ventilation System is isolated. The Reactor Building Purge Air Cleanup Units are always available as passive inline components to perform their function immediately after a fuel handling accident to process activity contained in exhaust air before it reaches the outside environment.

(continued)

BASES

BACKGROUND (continued)

The Primary containment exhaust is monitored by a radiation detector which provides automatic containment purge ventilation system isolation upon detecting the setpoint radioactivity in the exhaust air stream. The containment purge ventilation isolation valves will be automatically closed upon the actuation of a Containment Vent Isolation (CVI) signal whenever the primary containment is being purged during normal operation or upon manual actuation from the Main Control Room (Ref. 2). Requirements for Containment Vent Isolation Instrumentation are covered by LCO 3.3.6.

APPLICABLE SAFETY ANALYSES

The Reactor Building Purge Ventilation System air cleanup units ensure that the release of radioactivity to the environment is limited by cleaning up containment exhaust during a fuel handling accident before the containment purge exhaust valves are isolated. Reactor Building Purge Ventilation System filter efficiency is one of the inputs for the analysis of the environmental consequences of a fuel handling accident. Containment isolation can only result in smaller releases of radioactivity to the environment (Ref. 1). The Containment Vent Isolation System ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the containment (Ref. 2). Containment Vent Isolation Instrumentation is addressed by LCO 3.3.6.

The Reactor Building Purge Air Cleanup Units satisfy Criterion 3 of the NRC Policy Statement.

In addition, during movement of irradiated fuel in the Auxiliary Building when containment is open to the Auxiliary Building spaces, a high radiation signal from the spent fuel pool accident radiation monitors will initiate a CVI.

LCO

The safety function of the Reactor Building Purge Air Cleanup Unit is related to the initial control of offsite radiation exposures resulting from a fuel handling accident inside containment. During a fuel handling accident inside containment, the Reactor Building Purge Air Cleanup Unit provides a filtered path for cleaning up any air leaving the containment until the containment ventilation is isolated.

The plant design basis requires that when moving irradiated fuel in the Auxiliary Building and/or Containment with the Containment open to the Auxiliary Building ABSCE spaces, a signal from the spent fuel radiation monitors 0-RE-90-102 and -103 will initiate a CVI in addition to their normal function. In addition, a signal from the containment purge radiation monitors 1-RE-90-130, and -131 or other

(continued)

BASES

LCO (continued)

CVI signal will initiate that portion of the ABI normally initiated by the spent fuel pool radiation monitors. Therefore, the containment ventilation instrumentation must remain operable when moving irradiated fuel in the Auxiliary Building if the containment air locks, penetrations, equipment hatch, etc. are open to the Auxiliary Building ABSCE spaces. In addition, the ABGTS must remain operable if these containment penetrations are open to the Auxiliary Building during movement of irradiated fuel in side containment.

APPLICABILITY

An initial assumption in the analysis of a fuel handling accident inside containment is that the accident occurs while irradiated fuel is being handled. Therefore, LCO 3.9.8 is applicable only at this time. See additional discussion in the Applicable Safety Analysis and LCO sections.

ACTIONS

A.1 and A.2

If one Reactor Building Purge Air Cleanup Unit is inoperable, that air cleanup unit must be isolated. This places the system in the required accident configuration, thus allowing refueling to continue after verifying the remaining air cleanup unit is aligned and OPERABLE.

The immediate Completion Time is consistent with the required times for actions to be performed without delay and in a controlled manner.

B.1

With two Reactor Building Purge Air Cleanup Units inoperable, movement of irradiated fuel assemblies within containment must be suspended. This precludes the possibility of a fuel handling accident in containment with both Reactor Building Purge Air Cleanup Units inoperable. Performance of this action shall not preclude moving a component to a safe position.

The immediate Completion Time is consistent with the required times for actions to be performed without delay and in a controlled manner.

SURVEILLANCE REQUIREMENTS

SR 3.9.8.1

The Ventilation Filter Testing Program (VFTP) encompasses the Reactor Building Purge Air Cleanup Unit filter tests in accordance with Regulatory Guide 1.52 (Ref. 3). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.8.1 (continued)

the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

REFERENCES

1. Watts Bar FSAR, Section 15.5.6, "Environmental Consequences of a Postulated Fuel Handling Accident."

2. Watts Bar FSAR, Section 9.4.6, "Reactor Building Purge Ventilating System."

3. Regulatory Guide 1.52 (Rev. 02), "Design, Testing and Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

B 3.9 REFUELING OPERATIONS

B 3.9.9 Spent Fuel Pool Boron Concentration

BASES

BACKGROUND

The spent fuel storage rack criticality analysis assumes 2000 ppm soluble boron in the fuel pool during a dropped/misplaced fuel assembly event.

APPLICABLE SAFETY ANALYSES

This requirement ensures the presence of at least 2000 ppm soluble boron in the spent fuel pool water as assumed in the spent fuel rack criticality analysis for dropped/misplaced fuel assembly event.

The RCS boron concentration satisfies Criterion 2 of the NRC Policy Statement.

LCO

The LCO requires that the boron concentration in the spent fuel pool be greater than or equal to 2000 ppm during fuel movement.

APPLICABILITY

This LCO is applicable when the spent fuel pool is flooded and fuel is being moved. Once fuel movement begins, the movement is considered in progress until the configuration of the assemblies in the storage racks is verified to comply with the criticality loading criteria specified in Specification 4.3.1.1.

ACTIONS

A.1

If the spent fuel pool boron concentration does not meet the above requirements, fuel handling in the spent fuel pool must be suspended immediately. This action precludes a fuel handling accident, when conditions are outside those assumed in the accident analysis.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.9.1

This SR requires that the spent fuel pool boron concentration be verified greater than or equal to 2000 ppm. This surveillance is to be performed prior to movement of fuel in the spent fuel pool and at least once every 72 hours thereafter during the movement of fuel in the spent fuel pool.

The Frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of the sample. The Frequency is based on operating experience, which has shown 72 hours to be adequate.

REFERENCES

1. Watts Bar FSAR, Section 15, "Accident Analysis."

Enclosure 3

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Core Operating Limits Report

LIST OF ACRONYMS

Acronym	Title
ABGTS	Auxiliary Building Gas Treatment System
ACRP	Auxiliary Control Room Panel
ASME	American Society of Mechanical Engineers
AFD	Axial Flux Difference
AFW	Auxiliary Feedwater System
ARO	All Rods Out
ARFS	Air Return Fan System
ARV	Atmospheric Relief Valve
BOC	Beginning of Cycle
CCS	Component Cooling Water System
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CREVS	Control Room Emergency Ventilation System
CSS	Containment Spray System
CST	Condensate Storage Tank
DNB	Departure from Nucleate Boiling
ECCS	Emergency Core Cooling System
EFPD	Effective Full-Power Days
EGTS	Emergency Gas Treatment System
EOC	End of Cycle
ERCW	Essential Raw Cooling Water
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Features Actuation System
HEPA	High Efficiency Particulate Air
HVAC	Heating, Ventilating, and Air-Conditioning
LCC	Lower Compartment Cooler
LCO	Limiting Condition For Operation
MFIV	Main Feedwater Isolation Valve
MFRV	Main Feedwater Regulation Valve
MSIV	Main Steam Line Isolation Valve
MSSV	Main Steam Safety Valve
MTC	Moderator Temperature Coefficient
NMS	Neutron Monitoring System
ODCM	Offsite Dose Calculation Manual
PCP	Process Control Program
PIV	Pressure Isolation Valve
PORV	Power-Operated Relief Valve
PTLR	Pressure and Temperature Limits Report
QPTR	Quadrant Power Tilt Ratio
RAOC	Relaxed Axial Offset Control
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RTP	Rated Thermal Power
RTS	Reactor Trip System
RWST	Refueling Water Storage Tank
SG	Steam Generator
SI	Safety Injection
SL	Safety Limit
SR	Surveillance Requirement
UHS	Ultimate Heat Sink

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B 3.1-8	20	B 3.4-4	0
B 3.1-9	38	B 3.4-5	0
B 3.1-10	41	B 3.4-6	0
B 3.1-11	0	B 3.4-7	0
B 3.1-12	0	B 3.4-8	0
B 3.1-13	41	B 3.4-9	0
B 3.1-14	0	B 3.4-10	0
B 3.1-15	20	B 3.4-11	0
B 3.1-16	37	B 3.4-12	0
B 3.1-17	37	B 3.4-13	0
B 3.1-18	0	B 3.4-14	41
B 3.1-19	0	B 3.4-15	38
B 3.1-20	20	B 3.4-16	0
B 3.1-21	27	B 3.6-1	0
B 3.1-22	37	B 3.6-2	20
B 3.1-23	0	B 3.6-3	20
B 3.1-24	0	B 3.6-4	0
B 3.1-25	8	B 3.6-5	0
B 3.3-1	0	B 3.6-6	10
B 3.3-2	0	B 3.6-7	20
B 3.3-3	0	B 3.6-8	10
B 3.3-4	22	B 3.6-9	0
B 3.3-5	22	B 3.6-10	0
B 3.3-6	0	B 3.6-11	0
B 3.3-7	0	B 3.6-12	0
B 3.3-8	0	B 3.7-1	36

TECHNICAL REQUIREMENTS BASES

LIST OF EFFECTIVE PAGES

Page Number	Revision Number	Page Number	Revision Number
B 3.7-2	38	B 3.8-22	18
B 3.7-3	36	B 3.9-1	0
B 3.7-4	17	B 3.9-2	0
B 3.7-5	17	B 3.9-3	0
B 3.7-6	17	B 3.9-4	0
B 3.7-7	17	B 3.9-5	28
B 3.7-8	17	B 3.9-6	0
B 3.7-9	17	B 3.9-7	28
B 3.7-10	17	B 3.9-8	0
B 3.7-11	17	B 3.9-9	0
B 3.7-12	0		
B 3.7-13	5		
B 3.7-14	29		
B 3.7-15	4		
B 3.7-16	5		
B 3.7-17	29		
B 3.7-18	0		
B 3.7-19	0		
B 3.7-20	0		
B 3.7-21	0		
B 3.7-22	0		
B 3.7-23	20		
B 3.7-24	40		
B 3.7-25	40		
B 3.8-1	0		
B 3.8-2	0		
B 3.8-3	0		
B 3.8-4	0		
B 3.8-5	0		
B 3.8-6	25		
B 3.8-7	25		
B 3.8-8	0		
B 3.8-9	0		
B 3.8-10	0		
B 3.8-11	0		
B 3.8-12	0		
B 3.8-13	25		
B 3.8-14	25		
B 3.8-15	0		
B 3.8-16	0		
B 3.8-17	0		
B 3.8-18	0		
B 3.8-19	0		
B 3.8-20	0		
B 3.8-21	0		

LIST OF EFFECTIVE PAGES - REVISION LISTING

Revisions	Issued	SUBJECT
Revision 0	09-30-95	Initial Issue
Revision 1	12-06-95	Submerged Component Circuit Protection
Revision 2	01-04-96	Area Temperature Monitoring - Change in MSSV Limit
Revision 3	02-28-96	Turbine Driven AFW Pump Suction Requirement
Revision 4	08-18-97	Time-frame for Snubber Visual Exams
Revision 5	08-29-97	Performance of Snubber Functional Tests at Power
Revision 6	09-08-97	Revised Actions for Turbine Overspeed Protection
Revision 7	09-12-97	Change OPΔT/OTΔT Response Time
Revision 8	09-22-97	Clarification of Surveillance Frequency for Position Indication System
Revision 9	10-10-97	Revised Boron Concentration for Borated Water Sources
Revision 10	12-17-98	ICS Inlet Door Position Monitoring - Channel Check
Revision 11	01-08-99	Computer-Based Analysis for Loose Parts Monitoring
Revision 12	01-15-99	Removal of Process Control Program from TRM
Revision 13	03-30-99	Deletion of Power Range Neutron Flux High Negative Rate Reactor Trip Function
Revision 14	04-07-99	Submerged Component Circuit Protection
Revision 15	04-07-99	Submerged Component Circuit Protection
Revision 16	04-13-99	Submerged Component Circuit Protection
Revision 17	05-25-99	Flood Protection Plan
Revision 18	08-03-99	Submerged Component Circuit Protection
Revision 19	10-12-99	Upgrade Seismic Monitoring Instruments
Revision 20	03/13/00	Added Notes to Address Instrument Error for Various Parameters
Revision 21	04/13/00	COLR, Cycle 3, Rev 2
Revision 22	07/07/00	Elimination of Response Time Testing

LIST OF EFFECTIVE PAGES - REVISION LISTING

Revisions	Issued	SUBJECT
Revision 23	01/22/01	Plant Calorimetric (LEFM)
Revision 24	03/19/01	TRM Change Control Program per 50.59 Rule
Revision 25	05/15/01	Change in Preventive Maintenance Frequency for Molded Case Circuit Breakers
Revision 26	05/29/01	Change CVI Response Time from 5 to 6 Seconds
Revision 27	01/31/02	Change pH value in the borated water sources due to TS change for ice weight reduction
Revision 28	02/05/02	Refueling machine upgrade under DCN D-50991-A
Revision 29	02/26/02	Added an additional action to TR 3.7.3 to perform an engineering evaluation of inoperable snubber's impact on the operability of a supported system.
Revision 30	06/05/02	Updated TR 3.3.5.1 to reflect implementation of the TIPTOP program in a Technical Instruction (TI).
Revision 31	10/31/02	Correct RTP to 3459 MWt (PER 02-9519-000)
Revision 32	09/17/03	Editorial correction to Bases for TSR 3.1.5.3.
Revision 33	10/14/03	Updated TRs 3.1.5 and 3.1.6 and their respective bases to incorporate boron concentration changes in accordance with change packages WBN-TS-02-14 and WBN-TS-03-017.
Revision 34	05/14/04	Revised Item 5, "Source Range, Neutron Flux" function of Table 3.3.1-1 to provide an acceptable response time of less than or equal 0.5 seconds. (Reference TS Amendment 52.)
Revision 35	04/06/05	Revised Table 3.3.2-1, "Engineered Safety Features Actuation systems Response Times," to revise containment spray response time and to add an asterisk note to notation 13 of the table via Change Package WBN-TS-04-16.
Revision 36	09/25/06	Revised the response time for Containment Spray in Table 3.3.2-1 and the RT_{NDT} values in the Bases for TR 3.7.1. Both changes result from the replacement of the steam generators.
Revision 37	11/08/06	Revised TR 3.1.5 and 3.1.6 and the Bases for these TRs to update the boron concentration limits of the RWST and the BAT.

LIST OF EFFECTIVE PAGES
REVISION LISTING

Revisions	Issued	SUBJECT
Revision 38	11/29/06	Updated the TRM to be consistent with Tech Spec Amendment 55. TRM Revision 38 modified the requirements for mode change limitations in TR 3.0.4 and TSR 3.0.4 by incorporating changes similar to those outlined in TSTF-359, Revision 9. (TS-06-24)
Revision 39	04/16/07	Updated the TRM to be consistent with Tech Spec Amendment 42. TRM Revision 39 modified the requirements of TSR 3.0.3 by incorporating changes similar to those outlined in TSTF-358. (TS-07-03)
Revision 40	05/24/07	Updated the TRM and Bases to remove the various requirements for the submittal of reports to the NRC. (TS-07-06)
Revision 41	05/25/07	Revision 41 updates the Bases of TR 3.1.3, 3.1.4 and 3.4.5 to be consistent with Technical Specification Amendment 66. This amendment replaces the references to Section XI of the ASME Boiler and Pressure Vessel Code with the ASME Operation and Maintenance Code for Inservice Testing (IST) activities and removes reference to “applicable supports” from the IST program.
Revision 42	03/20/2008	Revision 42 updates Figure 3.1.6 to remove the 240 TPBAR Limit.

Enclosure 4

WBN Technical Requirements Manual - Changed Pages

TR 3.1 REACTIVITY CONTROL SYSTEMS

TR 3.1.6 Borated Water Sources, Operating

TR 3.1.6 The following borated water sources shall be OPERABLE as required by TR 3.1.2:

a. A Boric Acid Storage System, and

b. The Refueling Water Storage Tank (RWST).

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required Boric Acid Storage System, inoperable.	A.1 Restore Boric Acid Storage System, to OPERABLE status.	72 hours
	OR	
	A.2.1 Be in MODE 3.	78 hours
	AND	
	A.2.2 Borate to a SDM equivalent to ≥ 1% Δk/k at 200°F.	78 hours
	AND	
	A.2.3 Restore Boric Acid Storage System to OPERABLE status.	246 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. RWST boron concentration not within limits. OR RWST borated water temperature not within limits.	C.1 Restore RWST to OPERABLE status.	8 hours
D. RWST inoperable for reasons other than Condition C.	D.1 Restore RWST to OPERABLE status.	1 hour
E. Required Action and associated Completion Time of Condition C or D not met.	E.1 Be in MODE 3 AND E.2 Be in MODE 4 with one or more RCS cold leg temperatures ≤ 310 °F.	6 hours 12 hours

TECHNICAL SURVEILLANCE REQUIREMENTS

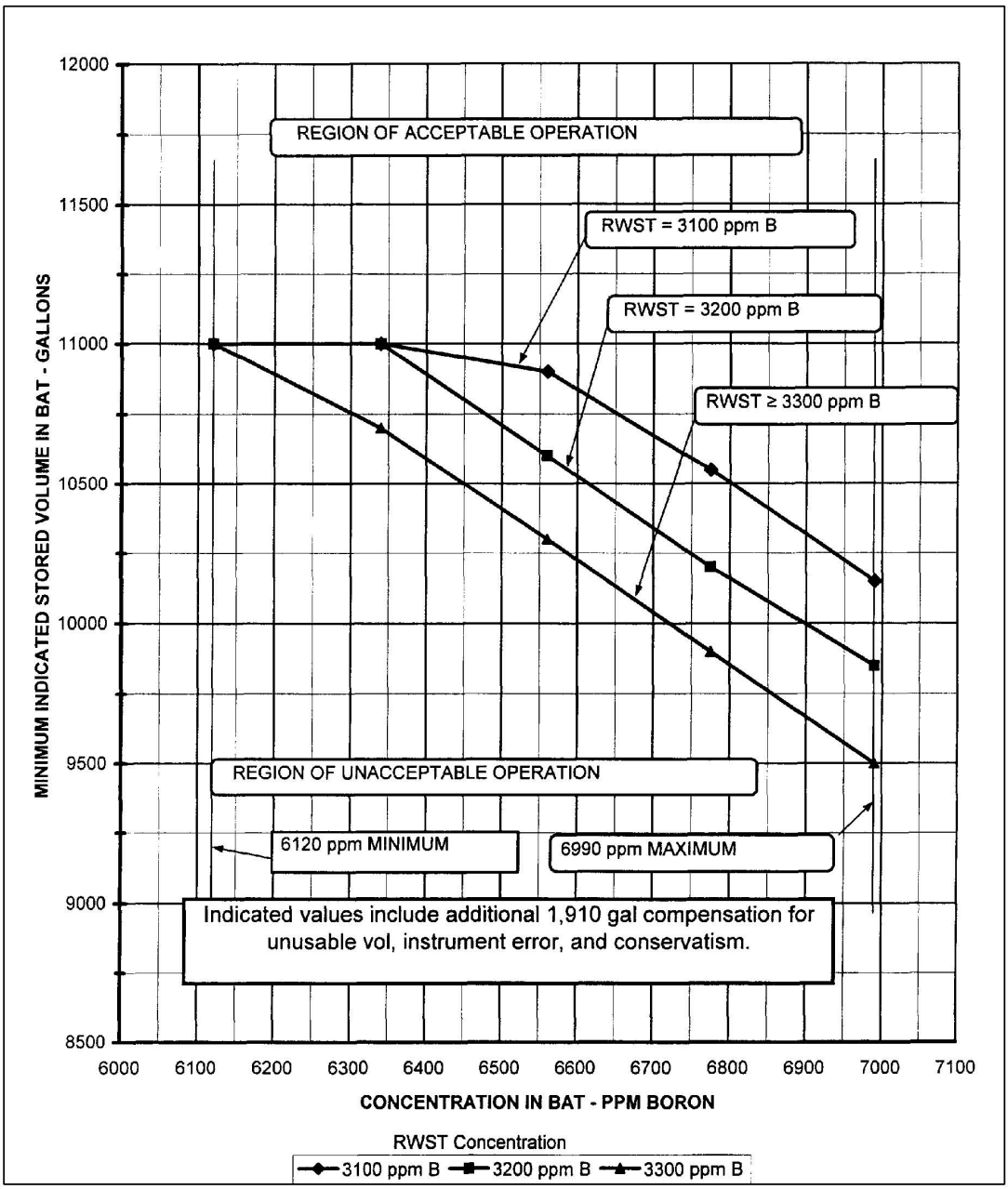
SURVEILLANCE		FREQUENCY
TSR 3.1.6.1	----------NOTE---------- Only required when outside air temperature is < 60 °F or >105 °F. ---------- Verify RWST solution temperature is ≥ 60° F and ≤ 105 °F.	24 hours
TSR 3.1.6.2	Verify RWST boron concentration is ≥ 3,100 ppm and ≤ 3,300 ppm.	7 days

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
TSR 3.1.6.3	Verifiy RWST borated water volume is ≥ 370,000 gallons.	7 days
TSR 3.1.6.4	----------NOTE---------- Only required if the BAT is required OPERABLE. ---------- Verify Boric Acid Tank (BAT) solution temperature is ≥ 63°F.	24 hours
TSR 3.1.6.5	----------NOTE---------- Only required if the BAT is required OPERABLE. ---------- Verify BAT boron concentration is in accordance with Figure 3.1.6.	7 days
TSR 3.1.6.6	----------NOTE---------- Only required if the BAT is required OPERABLE. ---------- Verify BAT borated water volume is in accordance with Figure 3.1.6.	7 days

TECHNICAL REQUIREMENTS FIGURE 3.1.6
BORIC ACID TANK LIMITS
BASED ON RWST BORON CONCENTRATION



TR 3.3 INSTRUMENTATION

TR 3.3.4 Seismic Instrumentation

TR 3.3.4 The seismic monitoring instrumentation shown in Table 3.3.4-1 shall be OPERABLE.

APPLICABILITY: At all times.

----------NOTE----------
TR 3.0.3 is not applicable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more seismic monitoring instruments in Panel 0-R-113 or foundation instrument 0-XT-52-75A in the Containment annulus inoperable for > 30 days, OR One or more remaining seismic monitoring instruments inoperable for >60 days.	A.1 Document in accordance with the Corrective Action Program.	In accordance with the Corrective Action Program.

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. ----------NOTE---------- All Required Actions must be completed whenever this Condition is entered. ----------	B.1 Document in accordance with the Corrective Action Program.	In accordance with the Corrective Action Program.
	AND	
One or more seismic monitoring instruments actuated during a seismic event.	B.2 Analyze data retrieved from 0-XT-52-75A to determine the magnitude of the vibratory ground motion.	4 hours
	AND	
	B.3 If OBE exceedance is verified, perform walkdowns of key plant equipment and structures to determine extent of damage.	8 hours
	AND	
	B.4 Restore each actuated monitoring instrument to OPERABLE status.	24 hours
	AND	
	B.5 Perform a CHANNEL CALIBRATION on each actuated monitoring instrument.	10 days
	AND	
	B.6 Analyze data retrieved from remaining seismic monitoring instruments.	14 days

TECHNICAL SURVEILLANCE REQUIREMENTS

----------NOTE----------
Refer to Table 3.3.4-1 to determine which Technical Surveillance Requirements apply for each seismic monitoring instrument.

SURVEILLANCE		FREQUENCY
TSR 3.3.4.1	Perform CHANNEL CHECK.	31 days
TSR 3.3.4.2	Perform CHANNEL OPERATIONAL TEST.	184 days
TSR 3.3.4.3	Perform CHANNEL CALIBRATION.	18 months

Table 3.3.4-1 (Page 1 of 1)

Seismic Monitoring Instrumentation

INSTRUMENTS AND SENSOR LOCATIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	MEASUREMENT RANGE
1. Strong Motion Triaxial Accelerometers[1] [5]			
a. 0-XT-52-75A Annulus El. 703	1	TSR 3.3.4.1[2] TSR 3.3.4.2[4] TSR 3.3.4.3[3]	0 - 1.0 g
b. 0-XT-52-75B Reactor Bldg. El. 757	1	TSR 3.3.4.1[2] TSR 3.3.4.2[4] TSR 3.3.4.3[3]	0 - 1.0 g
c. 0-XT-52-75D D/G Bldg. El. 742	1	TSR 3.3.4.1[2] TSR 3.3.4.2[4] TSR 3.3.4.3[3]	0 - 1.0 g
2. Triaxial Strong Motion Accelerograph			
a. 0-XR-52-80 Aux. Cont. Room. El 757	1	TSR 3.3.4.1[2] TSR 3.3.4.2[4] TSR 3.3.4.3[3]	0 - 2.0 g

(1) With associated acceleration triggers, and control room indication on 0-XR-52-82A, -82B, -83.
(2) Except acceleration trigger.
(3) Includes acceleration trigger.
(4) Except setpoint verification.
(5) Includes recording and analyzing components on 0-R-113.

TR 3.3 INSTRUMENTATION

TR 3.3.6 Loose-Part Detection System

TR 3.3.6 The Loose-Part Detection System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

----------NOTE----------
TR 3.0.3 is not applicable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels of Loose-Part Detection System inoperable > 30 days.	A.1 Document in accordance with the Corrective Action Program.	In accordance with the Corrective Action Program.

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.3.6.1	Perform CHANNEL CHECK.	24 hours
TSR 3.3.6.2	Perform CHANNEL OPERATIONAL TEST.	31 days
TSR 3.3.6.3	Perform CHANNEL CALIBRATION.	18 months

TR 3.7 PLANT SYSTEMS

TR 3.7.5 Area Temperature Monitoring

TR 3.7.5 The normal temperature limit of each area shown in Table 3.7.5-1 shall not be exceeded for > 8 hours and the abnormal temperature limits shall not be exceeded.

APPLICABILITY: Whenever the affected equipment in an area is required to be OPERABLE.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more areas exceeding normal temperature limits for > 8 hours.	A.1 ----------NOTE---------- TR 3.0.3 is not applicable. ---------- Document in accordance with the Corrective Action Program and include a record of the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s) and an analysis to demonstrate OPERABILITY of the affected equipment.	In accordance with the Corrective Action Program

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more areas exceeding abnormal temperature limits except for the IPS Mechanical or Electrical Equipment Rooms (Areas 31, 32, or 34).	B.1.1 Restore the area(s) to within normal temperature limits. OR B.1.2 Declare the affected equipment in the affected area(s) inoperable. AND B.2 Document in accordance with the Corrective Action Program and include a record of the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s) and an analysis to demonstrate OPERABILITY of the affected equipment.	4 hours 4 hours In accordance with the Corrective Action Program
C. Mechanical or Electrical Equipment Rooms in Intake Pumping Station (Areas 31, 32, or 34) less than 40°F and greater than 32°F.	C.1 Initiate action to maintain temperatures greater than 32°F. AND C.2 Restore temperatures to within normal limits.	24 hours 7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Mechanical or Electrical Equipment Rooms in Intake Pumping Station (Areas 31, 32, or 34) 32°F or less.	D.1 Declare the affected equipment in the affected area(s) inoperable. AND	Immediately
	D.2 Document in accordance with the Corrective Action Program and include a record of the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s) and an analysis to demonstrate OPERABILITY of the affected equipment.	In accordance with the Corrective Action Program

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.7.5.1	Verify each area temperature is within limits.	12 hours

Table 3.7.5-1 (Page 1 of 2)
Area Temperature Monitoring

	AREA	NORMAL LIMIT °F	ABNORMAL LIMIT °F
1.	Aux Bldg el 772 next to 480V Sd Bd transformer 1A2-A.	≤ 104	≤ 110
2.	Aux Bldg el 772 next to 480V Sd Bd transformer 1B1-B.	≤ 104	≤ 110
3.	Aux Bldg el 772 next to 480V Sd Bd transformer 2A2-A.	≤ 104	≤ 110
4.	Aux Bldg el 772 next to 480V Sd Bd transformer 2B2-B.	≤ 104	≤ 110
5.	Aux Bldg el 772 next to 480V Rx MOV Bd 1A2-A.	≤ 83	≤ 104
6.	Aux Bldg el 772 next to 480V Rx MOV Bd 2A2-A.	≤ 83	≤ 104
7.	Aux Bldg el 772 next to 480V Rx MOV Bd 2B2-B.	≤ 83	≤ 104
8.	Aux Bldg el 772 across from spare 125V vital battery charger 1-S.	≤ 83	≤ 104
9.	Aux Bldg el 772 U1 Mech Equip Room.	≤ 91	≤ 104
10.	Aux Bldg el 757 U1 Sd Bd room behind stairs S-A3.	≤ 85	≤ 104
11.	Aux Bldg el 757 U2 Sd Bd room behind stairs S-A13.	≤ 85	≤ 104
12.	Aux Bldg el 757 U1 Refueling beside Aux boration makeup tk.	≤ 104	≤ 115
13.	Aux Bldg el 737 U1 outside supply fan room.	≤ 104	≤ 110
14.	Aux Bldg el 713 U1 across from AFW pumps.	≤ 104	≤ 110
15.	Aux Bldg el 692 U1 outside AFW pump room door.	≤ 104	≤ 110
16.	Aux Bldg el 692 U2 near boric acid concentrate filter vault.	≤ 104	≤ 110
17.	Aux Bldg el 676 next to O-L-629.	≤ 104	≤ 110
18.	North steam valve vault room U1. (at affected MSSVs)	≥ 50	≥ 50

(continued)

Table 3.7.5-1 (Page 2 of 2)
Area Temperature Monitoring

	AREA	NORMAL LIMIT °F	ABNORMAL LIMIT °F
19.	South steam valve vault room U1. (at affected MSSVs)	≥ 50	≥ 50
20.	Add Equip Bldg U1 el 729 between UHI accumulators.	≤ 92	≤ 110
21.	CB Main Control Room south wall.	≤ 80	≤ 104
22.	CB Main Control Room across from 1-M-9.	≤ 80	≤ 104
23.	CB Computer room el 708 center of room.	≤ 74	≤ 104
24.	CB Aux. Instrument Room el 708.	≤ 90	≤ 104
25.	D/G Bldg el 742 2B-B D/G room on wall by battery charger.	≤ 104	≤ 120
26.	D/G Bldg el 742 1A-A D/G Room near D/G set.	≥ 50	≥ 50
27.	D/G Bldg el 742 1B-B D/G Room near D/G set.	≥ 50	≥ 50
28.	D/G Bldg el 742 2A-A D/G Room near D/G set.	≥ 50	≥ 50
29.	D/G Bldg el 742 2B-B D/G Room near D/G set.	≥ 50	≥ 50
30.	D/G Bldg el 760.5 next to 480V diesel Aux Bd 2B1-B.	≤ 104	≤ 120
31.	IPS Mechanical Equipment Room 1 el 722 near ERCW and HPFP Instruments and sense lines.	≥50≤104	≥40≤115
32.	IPS Mechanical Equipment Room 2 el 722 near ERCW and HPFP Instruments and sense lines.	≥50≤104	≥40≤115
33.	IPS el 741 in B train ERCW pump room.	≤ 120	≤ 120
34.	IPS el 711 next to 480V IPS board and transformer (A bus).	≥50≤104	≥40≤115
35.	IPS el 711 next to 480V IPS board and transformer (B bus).	≤ 104	≤ 115
36.	Add D/G Bldg el 742 C-S D/G Room near D/G set.	≥ 50	≥ 50

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Charging Pump, Shutdown

BASES

BACKGROUND	A description of the Boration Systems Flow Paths, which include charging pumps, is provided in the Bases for Technical Requirement 3.1.1, "Boration Systems Flow Paths, Shutdown."
APPLICABLE SAFETY ANALYSES	The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or Transient. In the case of a malfunction of the Chemical and Volume Control System, which causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system and/or stop the primary water pumps. This action is required before the SDM is lost. Operation of the boration subsystem is not assumed to mitigate this event (Ref. 1). OPERABILITY of the charging pumps, the refueling water storage tank, and the appropriate flow paths is required as part of the Emergency Core Cooling System (ECCS). The Technical Specifications for the ECCS address the requirements of these components. Technical Specification 3.4.12, "Cold Overpressure Mitigation System", places restrictions on maximum number of charging pumps allowed OPERABLE for overpressure concerns.
TR	TR 3.1.3 requires one charging pump in the required boron injection flow path to be OPERABLE and capable of being powered from an OPERABLE emergency power source during MODES 4, 5, and 6 in order to provide the driving force to accomplish (1) normal makeup, (2) chemical shim reactivity control, and (3) miscellaneous fill and transfer operations.
APPLICABILITY	The OPERABILITY of one charging pump in the required boron injection flow path ensures that this system is available for reactivity control while in MODES 4, 5, and 6. The APPLICABILITY statement is modified by the following Note to ensure the restrictions imposed by Technical Specification LCO 3.0.4.b are considered: For Mode 4, Technical Specification LCO 3.0.4.b is not applicable to ECCS high head (centrifugal charging) subsystem.

(continued)

BASES

APPLICABILITY (continued)

Charging pump OPERABILITY requirements for MODES 1, 2, and 3 are covered in Technical Requirement 3.1.4, "Charging Pumps - Operating."

ACTIONS

A.1 and A.2

With the required charging pump inoperable or not capable of being powered by an OPERABLE emergency power source, the plant must be placed in a condition where negative reactivity addition is not required. This is accomplished by suspending all CORE ALTERATIONS and positive reactivity additions immediately. One OPERABLE charging pump in the required boron injection flow path is required to meet the TR and to ensure that negative reactivity control is available during Modes 4, 5, and 6. Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.1.3.1

Periodic surveillance testing of charging pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is performed in accordance with the American Society of Mechanical Engineers (ASME) OM Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program, which encompasses the ASME OM Code. The ASME OM Code provides the activities and Frequencies necessary to satisfy the requirements.

REFERENCES

1. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Charging Pumps, Operating

BASES

BACKGROUND	A description of the Boration Systems Flow Paths is provided in the Bases for Technical Requirement 3.1.1, "Boration Systems Flow Paths, Shutdown."
APPLICABLE SAFETY ANALYSES	The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the Chemical and Volume Control System (CVCS), which causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system and/or stop the primary water pumps. This action is required before the shutdown margin is lost. Operation of the boration subsystem is not assumed to mitigate this event (Ref. 1). OPERABILITY of the charging pumps, the refueling water storage tank, and the appropriate flow paths is required as part of the Emergency Core Cooling System (ECCS). The Technical Specifications for the ECCS address the requirements of these components.
TR	TR 3.1.4 requires at least two charging pumps to be OPERABLE during MODES 1, 2, and 3 in order to assure redundant pumps to the two redundant flow paths to accomplish (1) normal makeup, (2) chemical shim reactivity control, and (3) miscellaneous fill and transfer operations.
APPLICABILITY	The OPERABILITY of two charging pumps ensures that the CVCS system is available for reactivity control while in MODES 1, 2, and 3. Two charging pumps are required to ensure single functional capability in the event an assumed failure renders one of the pumps inoperable. Charging pump OPERABILITY requirements for MODES 4, 5, and 6 are covered in Technical Requirement 3.1.3, "Charging Pumps - Shutdown".

(continued)

BASES (continued)

ACTIONS

A.1

If one of the required charging pumps is inoperable, action must be taken to restore a required charging pump to OPERABLE status. The 72-hour Completion Time was developed taking into account the redundant capabilities afforded by the OPERABLE charging pump and reasonable time for repairs. The Completion Time is consistent with the time allowed to restore an ECCS train or to restore a boron injection flow path to OPERABLE status (see Technical Specification 3.5.2, "ECCS-Operating" and Technical Requirement 3.1.2, "Boration Systems Flow Paths, Operating").

A.2.1, A.2.2, and A.2.3

An alternative to Required Action A.1 is to place the plant in MODE 3 and borate to a SDM equivalent to $\geq$ 1% $\Delta k/k$ at 200°F within 78 hours, and restore the required charging pump to OPERABLE status within 246 hours. This precludes the need for a flow path/charging pump for load follow and fuel burnup compensation, allowing the additional 7 days to restore two charging pumps to OPERABLE status. An additional 6 hours (78 hours total) are allowed to reach MODE 3 from full power in an orderly manner and without challenging plant systems. The allowed Completion Time to reach MODE 3 is reasonable, based on operating experience.

B.1

If two charging pumps cannot be restored to OPERABLE status or the Required Actions of Condition A are not met within the associated Completion Times, the plant must be placed in a MODE in which the TR does not apply. This is done by placing the plant in at least MODE 4 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.1.4.1

Periodic surveillance testing of charging pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is performed in accordance with the American Society of Mechanical Engineers (ASME) OM Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program, which encompasses the ASME OM Code. The ASME OM Code provides the activities and Frequencies necessary to satisfy the requirements.

REFERENCES

1. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

B 3.3 INSTRUMENTATION

B 3.3.4 Seismic Instrumentation

BASES

BACKGROUND

The seismic instrumentation is made up of several instruments such as accelerometers, an accelerograph, recorders, etc. These instruments are placed in several appropriate locations throughout the plant in order to provide data on the seismic input to containment, data on the frequency, amplitude and phase relationship of the seismic response of the containment structure, and data on the seismic input to other Seismic Category I structures (Ref. 1).

The seismic instrumentation is used to promptly determine the nature and severity of a seismic event and to predict the impact (i.e., potential for damage) on nuclear power plant features which are important to safety. This is required to permit comparison of the measured response to that used in the design basis for the unit to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Reference 1.

The original seismic instrumentation was replaced with state of the art digital instrumentation in order to permit application of EPRI OBE exceedance criteria delineated in References 4 and 5. Use of these criteria is permitted by Reference 6 provided that upgraded instrumentation is used. The replacement instrumentation is capable of recording a seismic event and performing appropriate analyses of the recorded data to provide an immediate basis for determining whether an OBE exceedance has occurred. Reference 6 directs that this information must be evaluated within 4 hours after an event and a walkdown of critical plant features must be accomplished within 8 hours after an event in order to make a determination as to whether a plant shutdown in warranted.

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and to determine the impact on those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the unit to determine if plant equipment inspection is required pursuant to Appendix A of 10 CFR part 100 prior to restart. Seismic risks which appear as dominant sequences in PRAs occur for very severe earthquakes with magnitudes which are a factor of two or three above the Safe Shutdown Earthquake and Design Basis Earthquake. The Seismic Instrumentation System was not designed to function or to provide comparative information for such severe earthquakes. This instrumentation is more pertinent to determining the need to shut down following a seismic event and the ability to restart the plant after seismic events which are not risk contributors, and is therefore not of prime importance in risk dominant sequences (Ref. 2).

(continued)

BASES (continued)

TR

TR 3.3.4 requires that the seismic monitoring instrumentation which is shown in Table 3.3.4-1 shall be OPERABLE. This requirement ensures that an assessment can be made of the effects on the plant of earthquakes which may occur that exceed the design basis spectra for the Operating Basis Earthquake (Ref. 3).

APPLICABILITY

Since the possibility of earthquakes is not MODE dependent, OPERABILITY of the seismic instrumentation is required at all times. The Applicability has been modified by a Note stating that the provisions of TR 3.0.3 do not apply.

ACTIONS

A.1

The determination as to whether an OBE exceedance has occurred is made by comparing the calculated spectra for the event with the applicable design basis spectra for that building and location. Reference 6 requires that this determination be made considering the data from instruments located on the Containment foundation. Therefore, the exceedance determination for WBN will be made using event data from 0-XT-52-75A in the Containment annulus. Data from this instrument is recorded at panel 0-R-113, which also contains the computer used to calculate the spectral content and the alarm panel used to annunciate in the control room. These devices are the key components used to detect the event and make a shutdown determination. With one or more of these required seismic monitoring instruments inoperable for more than 30 days, the inoperability of the instruments must be documented in accordance with the Corrective Action Program.

With one or more of the remaining seismic instruments inoperable for more than 60 days, the inoperability of the instruments must be documented in accordance with the Corrective Action Program. A longer period of inoperability is allowed for these instruments since they are used only for evaluating plant condition following an event and not for input to the shutdown decision.

B.1, B.2 and B.3

When one or more seismic monitoring instruments actuate during a seismic event with greater than or equal to 0.01g ground acceleration, all of the Required Actions under Condition B must be completed. The data retrieved from the actuated instruments must be analyzed to determine the magnitude of the vibratory ground motion. The replacement digital instrumentation provides the capability to analyze the event data onsite and generate event spectra to be used in determining whether an OBE exceedance has occurred. If an OBE exceedance has occurred, Reference 6 directs that this evaluation should occur within 4 hours after the event. Reference 6 also requires performance of a limited scope walkdown per Reference 7 to determine the extent of actual damage within 8 hours following the event. The information provided by this

(continued)

BASES

ACTIONS

B.1, B.2 and B.3 (continued)

walkdown and the spectral analysis are to be used in making a determination as to whether to proceed with plant shutdown. In addition, the seismic event must be documented in accordance with the Corrective Action Program.

B.4 and B.5

Each actuated monitoring instrument must be restored to OPERABLE status within 24 hours. Within 10 days of the actuation, a CHANNEL CALIBRATION must be performed on each actuated monitoring instrument. The Completion Time of 10 days to perform Required Action B.2 is reasonable and is based on engineering judgment.

B.6

Subsequent analysis must then be performed using data from the remaining seismic monitoring instruments to evaluate the plant response in comparison with previously generated design basis spectra at the locations of those instruments. The Completion Time of 14 days to perform Required Action B.6 is reasonable and based upon the typical time necessary to analyze data.

TECHNICAL SURVEILLANCE REQUIREMENTS

The SRs for each seismic monitoring Function are identified by the SRs column of Table 3.3.4-1.

A Note has been added to the TSRs to clarify that Table 3.3.4-1 determines which SRs apply to which seismic monitoring instruments.

Performance of a CHANNEL CHECK on the seismic instrumentation once every 31 days ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a check of external system status indications that the seismic monitoring equipment is in a state of readiness to properly function should an earthquake occur. A CHANNEL CHECK will detect gross system failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL OPERATIONAL TEST.

The Surveillance Frequency of 31 days is based on operating experience related to instrumentation systems, which demonstrates that gross instrumentation system failure in any 31-day interval is a rare event. The CHANNEL CHECK supplements the loss of power annunciation for the equipment in the auxiliary instrument room. The equipment in the auxiliary control room does not have a loss of power alarm but only provides supplemental data.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.3.4.2

A CHANNEL OPERATIONAL TEST is to be performed on each required channel to ensure the entire channel will perform the intended function. A CHANNEL OPERATIONAL TEST is the comparison of the response of the instrumentation, including all components of the instrument, to a known signal. Although the seismic trigger is functionally checked, its setpoint is not verified. The Surveillance Frequency of 184 days is based upon the known reliability of the monitoring instrumentation and has been shown to be acceptable through operating experience.

TSR 3.3.4.3

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor by comparing the response of the instrument to a known input on the sensor. This test verifies the capability of the seismic instrumentation to correctly determine the magnitude of a seismic event and evaluate the response of those features important to safety. The Surveillance Frequency of 18 months is based upon operating experience and consistency with the typical industry refueling cycle.

REFERENCES

1. Regulatory Guide 1.12, “Instrumentation for Earthquakes,” Revision 1, April 1974.

2. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

3. Watts Bar FSAR, Section 3.7.4, "Seismic Instrumentation Program."

4. EPRI NO-5930, July 1988, “A Criterion for Determining Exceedance of the Operating Basis Earthquake”

5. EPRI TR-104239, June 1994, “Seismic Instrumentation in Nuclear Power Plants for Response to OBE Exceedance: Guideline for Implementation”

6. Regulatory Guide 1.166, “Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post-Earthquake Actions”, Revision 0, March 1997.

7. EPRI NP-6695, December 1989, “Guidelines for Nuclear Plant Response to an Earthquake”

B 3.3 INSTRUMENTATION

B 3.3.6 Loose-Part Detection System

BASES

BACKGROUND

The Loose-Part Detection System consists of six sensors, a system cabinet, alarm units, a frequency-modulated tape recorder, an audio monitor, and calibration devices. The sensors are located in the six natural collection regions. These regions consist of the top and bottom plenums of the reactor vessel and the primary coolant inlet plenum to each steam generator. There are installed spares at each sensor location. The entire system is described in Reference 1.

The Loose-Part Detection System provides the capability to detect acoustic disturbances indicative of loose parts within the Reactor Coolant System (RCS) pressure boundary. This system is provided to avoid or mitigate damage to RCS components that could occur from these loose parts. The Loose-Part Detection System Technical Requirement is consistent with the recommendations of Reference 2.

APPLICABLE SAFETY ANALYSES

The presence of a loose part in the RCS can be indicative of degraded reactor safety resulting from failure or weakening of a safety-related component. A loose part, whether it be from a failed or weakened component, or from an item inadvertently left in the primary system during construction, refueling, or maintenance, can contribute to component damage and material wear by frequent impacting with other parts in the system. Also, a loose part increases the potential for control-rod jamming and for accumulation of increased levels of radioactive crud in the primary system (Ref. 2).

The Loose Part Detection System provides the capability to detect loose parts in the RCS which could cause damage to some component in the RCS. Loose parts are not assumed to initiate any DBA, and the detection of a loose part is not required for mitigation of any DBA (Ref. 3).

(continued)

BASES (continued)

TR

TR 3.3.6 requires the Loose-Part Detection System to be OPERABLE. This is necessary to ensure that sufficient capability is available to detect loose metallic parts in the RCS and avoid or mitigate damage to the RCS components. This requirement is provided in Reference 2.

APPLICABILITY

TR 3.3.6 is required to be met in MODES 1 and 2 as stated in Reference 2. These MODES of applicability are provided in Reference 2.

The Applicability has been modified by a Note stating that the provisions of TR 3.0.3 do not apply.

ACTIONS

A.1

If one or more required channels of the Loose-Part Detection System are inoperable for more than 30 days, document the inoperability of the channel in accordance with Corrective Action Program.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.3.6.1

Performance of a CHANNEL CHECK for the Loose-Part Detection System once every 24 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or of even something more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.3.6.1 (continued)

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the match criteria, it may be an indication that the sensor or the signal-processing equipment has drifted outside its limit.

The Surveillance and the Surveillance Frequency are provided in Reference 2.

TSR 3.3.6.2

A CHANNEL OPERATIONAL TEST is to be performed every 31 days on each required channel to ensure the entire channel will perform the intended function. This test verifies the capability of the Loose-Part Detection System to detect impact signals which would indicate a loose part in the RCS. The Surveillance and the Surveillance Frequency are provided in Reference 2.

TSR 3.3.6.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The Surveillance Frequency of 18 months is based upon operating experience and is consistent with the typical industry refueling cycle. The Surveillance and the Surveillance Frequency are provided in Reference 2. Reference 1 describes the use of a computer-based analytical system to verify proper channel calibration. This is an acceptable option to using a mechanical impact device for sensors located in plant areas where plant personnel radiation exposure is considered by Plant Management to be excessive.

REFERENCES

1. Watts Bar FSAR, Section 7.6.7, "Loose Part Monitoring System (LPMS) System Description."

2. Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors."

3. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 Piping System Structural Integrity

BASES

BACKGROUND	Inservice inspection of ASME Code Class 1, 2, and 3 components and pressure testing of ASME Code Class 1, 2, and 3 pumps and valves are performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code (Ref. 1) and applicable Addenda, as required by 10 CFR 50.55a(g) (Ref. 2). Exception to these requirements apply where relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i) and (a)(3). In general, the surveillance intervals specified in Section XI of the ASME Code apply. However, the Inservice Inspection Program includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Code. This clarification is provided to ensure consistency in surveillance intervals throughout the Technical Specifications. Each reactor coolant pump flywheel is, in addition, inspected as recommended in Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975 (Ref. 3). Additionally, programmatic information on Inservice Inspection is provided in Technical Specifications, Chapter 5.0, Administrative Controls, Section 5.7.2.11, Inservice Inspection Program.
APPLICABLE SAFETY ANALYSES	Certain components which are designed and manufactured to the requirements of specific sections of the ASME Boiler and Pressure Vessel Code are part of the primary success path and function to mitigate DBAs and transients. However, the operability of these components is addressed in the relevant specifications that cover individual components. In addition, this particular Requirement covers only structural integrity inspection/testing requirements for these components, which is not a consideration in designing the accident sequences for theoretical hazard evaluation (Ref.4).
TR	TR 3.4.5 requires that the structural integrity of the ASME Code Class 1, 2, and 3 components be maintained in accordance with TSR 3.4.5.1 and TSR 3.4.5.2. In those areas where conflict may exist between the Technical Specifications and the ASME Boiler and Pressure Vessel Code, the Technical Specifications take precedence.

(continued)

BASES (continued)

APPLICABILITY

The structural integrity of the ASME Code Class 1 components is required in all MODES, when the temperature is above the minimum temperature required by NDT considerations. For ASME Code Class 2 components, the structural integrity is required when the temperature is above 200 °F. For ASME Code Class 3 components, the structural integrity is required at all times when the particular component is in service.

ACTIONS

A.1 and A.2

Required Actions A.1 and A.2 apply to ASME Code Class 1 components. Required Action A.1 stipulates that structural integrity should be restored before the temperature of the component is increased more than 50°F above the minimum temperature required by NDT considerations. Alternatively, the component could be isolated before the temperature reaches 50°F above the minimum temperature required by NDT considerations.

B.1 and B.2

Required Actions B.1 and B.2 apply to ASME Code Class 2 components. Required Actions B.1 stipulates that structural integrity should be restored before the temperature of the component is increased more than 200°F. Alternatively, the component could be isolated before the temperature reaches 200°F.

C.1, C.2.1, and C.2.2

Required Actions C.1, C.2.1, and C.2.2 apply to ASME Code Class 3 components. Required Action C.1 requires that the applicable Conditions and Required Actions for the affected components be entered immediately. Additionally, the structural integrity of all components must be satisfied or the particular component which does not satisfy the required structural integrity must be isolated from the system within the Completion Time specified in the affected components LCO or TR.

(continued)

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.4.5.1

This surveillance stipulates inspection of the coolant pump flywheel in accordance with Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1. This inspection verifies the structural integrity of the flywheel.

TSR 3.4.5.2

TSR 3.4.5.2 requires the verification of structural integrity of ASME Code Class 1, 2, and 3 components are in accordance with the Inservice Inspection Program.

REFERENCES

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

2. 10 CFR 50.55a, "Codes and Standards."

3. Regulatory Guide 1.14, Revision 1, 1975.

4. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

B 3.7 PLANT SYSTEMS

B 3.7.5 Area Temperature Monitoring

BASES

BACKGROUND

Thermal-life of various electrical and mechanical equipment is one of several important aging concerns in the qualification of hardware. The requirement is that the equipment remains functional during and after specified design basis events. Design basis events consist of loss of offsite power and design basis accidents (DBA). In general, the following three groups of hardware are subjected to qualification:

a. Safety related equipment
b. Non-safety related equipment (failure of which could prevent safety related equipment to operate as designed)
c. Specific post-accident monitoring equipment.

The normal service temperatures of concern are relatively low, hence, most of the equipment requiring consideration are components in the electrical power supply and the instrumentation systems. Some of these components are designed for relatively low temperature with very little margin to normal operating temperatures in cabinets and buildings. The procedure for thermal qualification is normally to subject prototypes from the production line to life tests by natural or artificial (accelerated) aging to its end-of-installed life condition. Analyses with justifications of methods and assumptions are used to qualify the prototypes to the actual service conditions, which may differ from the test conditions. Although the equipment is qualified for an environment expected after a DBA, the components are only subjected to normal operating conditions for most of the design life. Therefore, the thermal aging due to normal operating conditions is of major importance and is the parameter which is controlled by the Technical Requirements. Accordingly, this particular requirement establishes temperature limits during normal operation for specific locations in various buildings, except the containment. The temperature limits are related to the expected thermal-life for the hardware which operates in the areas where the temperatures are monitored and controlled.

(continued)

BASES

BACKGROUND (continued)

Due to valve design, ambient temperatures can affect the setpoints of the main steam safety valves (MSSVs), whereby a decrease in valve body temperature causes an increase in setpoint, resulting in non-conservative relief pressure. Ambient temperatures are monitored within the main steam valve vaults to ensure that the MSSVs minimum temperatures are maintained to meet the 1% code allowable variance on setpoints. Detailed BASES for the MSSVs is provided in Technical Specification B 3.7.1.

The general guidelines, which are followed for the qualification of electrical equipment, are provided in 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants" (Ref. 1). Detailed requirements for the implementation of the general guidelines are provided in various Regulatory Guides and IEEE Standards. Basic requirements for the qualification of mechanical equipment are outlined in General Design Criteria 4 (Ref. 2).

APPLICABLE SAFETY ANALYSIS

Certain components, which have the service temperatures controlled by this requirement, are part of the primary success path and function to mitigate DBAs and transients. However, the integrity/OPERABILITY of these components is addressed in the relevant specifications that cover individual components. The service temperatures and the thermal aging, which are controlled by observing the requirements of this TR, are not inputs to the safety analysis. Further, Probabilistic Risk Assessment studies performed to date, do not explicitly model the function of area temperature monitors. In addition, this particular requirement covers only service temperatures and thermal aging of these components, which are not considerations in designing the accident sequences for theoretical hazard evaluations (Ref. 3).

TR

TR 3.7.5 provides nominal temperature limits in the vicinity of major equipment. The TR allows for each area shown in Table 3.7.5-1 to be higher or lower than the normal limit for a maximum of eight hours. Note that the temperature values listed in Table 3.7.5-1 do not account for instrument error.

APPLICABILITY

The limits on temperature and time apply whenever the affected equipment in an affected area is required to be OPERABLE.

(continued)

BASES (continued)

ACTIONS

A.1

Whenever the temperature in one or more areas have exceeded the normal temperature limits for more than eight hours, document the exceedance in accordance with the Corrective Action Program. The report must contain the cumulative time and the amount by which the temperature has exceeded the limits.

Condition A has been modified by a Note stating that the provisions of TR 3.0.3 do not apply.

B.1.1, B.1.2, and B.2

Whenever the temperature in one or more areas exceeds the abnormal temperature limits, the temperature must be restored to within the normal limits in 4 hours. The Completion Time of 4 hours is based on operator experience and is a reasonable time for restoring the temperature. Alternatively, the affected equipment must be declared inoperable and the inoperability documented in accordance with the Corrective Action Program along with the cumulative time and the amount by which the temperature has exceeded the limits. In addition, an analysis shall be prepared which demonstrates OPERABILITY of the affected equipment.

C.1 and C.2

Whenever the temperature in the Intake Pumping Station mechanical or electrical equipment rooms exceeds the lower limit of 40 °F, actions must be initiated within 24 hours to ensure the temperature does not decrease below 32 °F. The Completion Time of 24 hours is based on temperature analysis. Within 7 days, restore normal temperatures within the areas affected. The 7 day Completion Time is based on a reasonable repair duration, and compensatory actions available during the interim period to maintain temperatures above 32 °F.

(continued)

BASES

ACTIONS (continued)

D.1 and D.2

If the temperature in the Intake Pumping Station mechanical or electrical equipment rooms decrease to 32 °F or lower, the affected equipment must be immediately declared inoperable. The Completion Time is based on potential freezing of safety-related components. The inoperability of the equipment must be documented in the Corrective Action Program along with the cumulative time and amount by which the temperature has exceeded the limits. In addition, an analysis shall be prepared which demonstrates OPERABILITY of the affected equipment.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.5.1

The temperatures for the areas listed in Table 3.7.5-1 must be determined every 12 hours to ensure compliance with the limits. The 12 hour Frequency is based on engineering experience and is reasonable considering the time required for performing the surveillance and the probability for changes in the area temperatures. Note that the temperature values listed in Table 3.7.5-1 do not account for instrument error.

REFERENCES

1. 10 CFR 50.49 "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."

2. 10 CFR 50 Appendix A, General Design Criteria 4, "Environmental and Dynamic Effects Design Bases."

3. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.