MANUAL HARD COPY DISTRIBUTION
DOCUMENT TRANSMITTAL 2006-18643

USER INFORMATION:

GERLACH*ROSE M EMPL#:028401 CA#: 0363

Address: NUCSA2

Phone#: 254-3194

TRANSMITTAL INFORMATION:

TO: GERLACH*ROSE M 04/13/2006
LOCATION: USNRC
FROM: NUCLEAR RECORDS DOCUMENT CONTROL CENTER (NUCSA-2)
THE FOLLOWING CHANGES HAVE OCCURRED TO THE HARDCOPY OR ELECTRONIC MANUAL ASSIGNED TO YOU. HARDCOPY USERS MUST ENSURE THE DOCUMENTS PROVIDED MATCH THE INFORMATION ON THIS TRANSMITTAL. WHEN REPLACING THIS MATERIAL IN YOUR HARDCOPY MANUAL, ENSURE THE UPDATE DOCUMENT ID IS THE SAME DOCUMENT ID YOU'RE REMOVING FROM YOUR MANUAL. TOOLS FROM THE HUMAN PERFORMANCE TOOL BAG SHOULD BE UTILIZED TO ELIMINATE THE CHANCE OF ERRORS.
ATTENTION: "REPLACE" directions do not affect the Table of Contents, Therefore no TOC will be issued with the updated material.

TSB1 - TECHNICAL SPECIFICATION BASES UNIT 1 MANUAL

REMOVE MANUAL TABLE OF CONTENTS DATE: 04/02/2006

ADD MANUAL TABLE OF CONTENTS DATE: 04/12/2006

CATEGORY: DOCUMENTS TYPE: TSB1

A001

ID: TEXT 3.10.8
REMOVE: REV:0

ADD: REV: 1

CATEGORY: DOCUMENTS TYPE: TSB1
ID: TEXT 3.2.4
REMOVE: REV:1

ADD: REV: 2

CATEGORY: DOCUMENTS TYPE: TSB1
ID: TEXT 3.3.1.1
REMOVE: REV:2

ADD: REV: 3

CATEGORY: DOCUMENTS TYPE: TSB1
ID: TEXT 3.3.1.2
REMOVE: REV:0

ADD: REV: 1

CATEGORY: DOCUMENTS TYPE: TSB1
ID: TEXT 3.3.1.3
REMOVE: REV:0

CATEGORY: DOCUMENTS TYPE: TSB1
ID: TEXT 3.3.2.1
REMOVE: REV:1

ADD: REV: 2

CATEGORY: DOCUMENTS TYPE: TSB1
ID: TEXT 3.3.3.1
REMOVE: REV:2

ADD: REV: 3

CATEGORY: DOCUMENTS TYPE: TSB1
ID: TEXT 3.4.1
REMOVE: REV:2

ADD: REV: 3

CATEGORY: DOCUMENTS TYPE: TSB1
ID: TEXT LOES
REMOVE: REV:69

ADD: REV: 70

CATEGORY: DOCUMENTS TYPE: TSB1
ID: TEXT TOC
REMOVE: REV:7

ADD: REV: 8

ANY DISCREPANCIES WITH THE MATERIAL PROVIDED, CONTACT DCS @ X3107 OR X3136 FOR ASSISTANCE. UPDATES FOR HARDCOPY MANUALS WILL BE DISTRIBUTED WITHIN 5 DAYS IN ACCORDANCE WITH DEPARTMENT PROCEDURES. PLEASE MAKE ALL CHANGES AND ACKNOWLEDGE COMPLETE IN YOUR NIMS INBOX UPON COMPLETION OF UPDATES. FOR ELECTRONIC MANUAL USERS, ELECTRONICALLY REVIEW THE APPROPRIATE DOCUMENTS AND ACKNOWLEDGE COMPLETE IN YOUR NIMS INBOX.

Manual Name: TSB1

Manual Title: TECHNICAL SPECIFICATION BASES UNIT 1 MANUAL

Table Of Contents

Issue Date: 04/12/2006

Procedure Name	Rev	Issue Date	Change ID	Change Number
TEXT LOES	70	04/12/2006		
Title: LIST OF EFFECTIVE SECTIONS				
TEXT TOC	8	04/12/2006		
Title: TABLE OF CONTENTS				
TEXT 2.1.1	1	04/27/2004		
Title: SAFETY LIMITS (SLS) REACTOR CORE SLS				
TEXT 2.1.2	0	11/15/2002		
Title: SAFETY LIMITS (SLS) REACTOR COOLANT SYSTEM (RCS) PRESSURE SL				
TEXT 3.0	1	04/18/2005		
Title: LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY				
TEXT 3.1.1	0	11/15/2002		
Title: REACTIVITY CONTROL SYSTEMS SHUTDOWN MARGIN (SDM)				
TEXT 3.1.2	0	11/15/2002		
Title: REACTIVITY CONTROL SYSTEMS REACTIVITY ANOMALIES				
TEXT 3.1.3	1	07/06/2005		
Title: REACTIVITY CONTROL SYSTEMS CONTROL ROD OPERABILITY				
TEXT 3.1.4	1	07/06/2005		
Title: REACTIVITY CONTROL SYSTEMS CONTROL ROD SCRAM TIMES				
TEXT 3.1.5	1	07/06/2005		
Title: REACTIVITY CONTROL SYSTEMS CONTROL ROD SCRAM ACCUMULATORS				
TEXT 3.1.6	1	02/17/2005		
Title: REACTIVITY CONTROL SYSTEMS ROD PATTERN CONTROL				

Manual Name: TSB1

Manual Title: TECHNICAL SPECIFICATION BASES UNIT 1 MANUAL

TEXT 3.1.7 1 08/30/2005
Title: REACTIVITY CONTROL SYSTEMS STANDBY LIQUID CONTROL (SLC) SYSTEM

TEXT 3.1.8 1 10/19/2005
Title: REACTIVITY CONTROL SYSTEMS SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN VALVES

TEXT 3.2.1 0 11/15/2002
Title: POWER DISTRIBUTION LIMITS AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

TEXT 3.2.2 0 11/15/2002
Title: POWER DISTRIBUTION LIMITS MINIMUM CRITICAL POWER RATIO (MCPR)

TEXT 3.2.3 0 11/15/2002
Title: POWER DISTRIBUTION LIMITS LINEAR HEAT GENERATION RATE (LHGR)

TEXT 3.2.4 2 04/12/2006
Title: POWER DISTRIBUTION LIMITS AVERAGE POWER RANGE MONITOR (APRM) GAIN AND SETPOINTS

TEXT 3.3.1.1 3 04/12/2006
Title: INSTRUMENTATION REACTOR PROTECTION SYSTEM (RPS) INSTRUMENTATION

TEXT 3.3.1.2 1 04/12/2006
Title: INSTRUMENTATION SOURCE RANGE MONITOR (SRM) INSTRUMENTATION

TEXT 3.3.2.1 2 04/12/2006
Title: INSTRUMENTATION CONTROL ROD BLOCK INSTRUMENTATION

TEXT 3.3.2.2 0 11/15/2002
Title: INSTRUMENTATION FEEDWATER - MAIN TURBINE HIGH WATER LEVEL TRIP INSTRUMENTATION

TEXT 3.3.3.1 3 04/12/2006
Title: INSTRUMENTATION POST ACCIDENT MONITORING (PAM) INSTRUMENTATION

TEXT 3.3.3.2 1 04/18/2005
Title: INSTRUMENTATION REMOTE SHUTDOWN SYSTEM

Manual Name: TSB1

Manual Title: TECHNICAL SPECIFICATION BASES UNIT 1 MANUAL

TEXT 3.3.4.1 0 11/15/2002
Title: INSTRUMENTATION END OF CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) INSTRUMENTATION

TEXT 3.3.4.2 0 11/15/2002
Title: INSTRUMENTATION ANTICIPATED TRANSIENT WITHOUT SCRAM RECIRCULATION PUMP TRIP (ATWS-RPT) INSTRUMENTATION

TEXT 3.3.5.1 2 07/06/2005
Title: INSTRUMENTATION EMERGENCY CORE COOLING SYSTEM (ECCS) INSTRUMENTATION

TEXT 3.3.5.2 0 11/15/2002
Title: INSTRUMENTATION REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM INSTRUMENTATION

TEXT 3.3.6.1 1 11/09/2004
Title: INSTRUMENTATION PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

TEXT 3.3.6.2 1 11/09/2004
Title: INSTRUMENTATION SECONDARY CONTAINMENT ISOLATION INSTRUMENTATION

TEXT 3.3.7.1 0 11/15/2002
Title: INSTRUMENTATION CONTROL ROOM EMERGENCY OUTSIDE AIR SUPPLY (CREOAS) SYSTEM INSTRUMENTATION

TEXT 3.3.8.1 1 09/02/2004
Title: INSTRUMENTATION LOSS OF POWER (LOP) INSTRUMENTATION

TEXT 3.3.8.2 0 11/15/2002
Title: INSTRUMENTATION REACTOR PROTECTION SYSTEM (RPS) ELECTRIC POWER MONITORING

TEXT 3.4.1 3 04/12/2006
Title: REACTOR COOLANT SYSTEM (RCS) RECIRCULATION LOOPS OPERATING

TEXT 3.4.2 0 11/15/2002
Title: REACTOR COOLANT SYSTEM (RCS) JET PUMPS

TEXT 3.4.3 1 01/16/2006
Title: REACTOR COOLANT SYSTEM (RCS) SAFETY/RELIEF VALVES (S/RVS)

Manual Name: TSB1

Manual Title: TECHNICAL SPECIFICATION BASES UNIT 1 MANUAL

TEXT 3.4.4 0 11/15/2002
Title: REACTOR COOLANT SYSTEM (RCS) RCS OPERATIONAL LEAKAGE

TEXT 3.4.5 1 01/16/2006
Title: REACTOR COOLANT SYSTEM (RCS) RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

TEXT 3.4.6 1 04/18/2005
Title: REACTOR COOLANT SYSTEM (RCS) RCS LEAKAGE DETECTION INSTRUMENTATION

TEXT 3.4.7 1 04/18/2005
Title: REACTOR COOLANT SYSTEM (RCS) RCS SPECIFIC ACTIVITY

TEXT 3.4.8 1 04/18/2005
Title: REACTOR COOLANT SYSTEM (RCS) RESIDUAL HEAT REMOVAL (RHR) SHUTDOWN COOLING SYSTEM - HOT SHUTDOWN

TEXT 3.4.9 0 11/15/2002
Title: REACTOR COOLANT SYSTEM (RCS) RESIDUAL HEAT REMOVAL (RHR) SHUTDOWN COOLING SYSTEM - COLD SHUTDOWN

TEXT 3.4.10 0 11/15/2002
Title: REACTOR COOLANT SYSTEM (RCS) RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

TEXT 3.4.11 0 11/15/2002
Title: REACTOR COOLANT SYSTEM (RCS) REACTOR STEAM DOME PRESSURE

TEXT 3.5.1 2 01/16/2006
Title: EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM ECCS - OPERATING

TEXT 3.5.2 0 11/15/2002
Title: EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM ECCS - SHUTDOWN

TEXT 3.5.3 1 04/18/2005
Title: EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM RCIC SYSTEM

TEXT 3.6.1.1 0 11/15/2002
Title: CONTAINMENT SYSTEMS PRIMARY CONTAINMENT

Manual Name: TSB1

Manual Title: TECHNICAL SPECIFICATION BASES UNIT 1 MANUAL

TEXT 3.6.1.2 0 11/15/2002
Title: CONTAINMENT SYSTEMS PRIMARY CONTAINMENT AIR LOCK

TEXT 3.6.1.3 3 12/08/2005
Title: CONTAINMENT SYSTEMS PRIMARY CONTAINMENT ISOLATION VALVES (PCIVS)

TEXT 3.6.1.4 0 11/15/2002
Title: CONTAINMENT SYSTEMS CONTAINMENT PRESSURE

TEXT 3.6.1.5 1 10/05/2005
Title: CONTAINMENT SYSTEMS DRYWELL AIR TEMPERATURE

TEXT 3.6.1.6 0 11/15/2002
Title: CONTAINMENT SYSTEMS SUPPRESSION CHAMBER-TO-DRYWELL VACUUM BREAKERS

TEXT 3.6.2.1 0 11/15/2002
Title: CONTAINMENT SYSTEMS SUPPRESSION POOL AVERAGE TEMPERATURE

TEXT 3.6.2.2 0 11/15/2002
Title: CONTAINMENT SYSTEMS SUPPRESSION POOL WATER LEVEL

TEXT 3.6.2.3 1 01/16/2006
Title: CONTAINMENT SYSTEMS RESIDUAL HEAT REMOVAL (RHR) SUPPRESSION POOL COOLING

TEXT 3.6.2.4 0 11/15/2002
Title: CONTAINMENT SYSTEMS RESIDUAL HEAT REMOVAL (RHR) SUPPRESSION POOL SPRAY

TEXT 3.6.3.1 1 04/18/2005
Title: CONTAINMENT SYSTEMS PRIMARY CONTAINMENT HYDROGEN RECOMBINERS

TEXT 3.6.3.2 1 04/18/2005
Title: CONTAINMENT SYSTEMS DRYWELL AIR FLOW SYSTEM

TEXT 3.6.3.3 0 11/15/2002
Title: CONTAINMENT SYSTEMS PRIMARY CONTAINMENT OXYGEN CONCENTRATION

Manual Name: TSB1

Manual Title: TECHNICAL SPECIFICATION BASES UNIT 1 MANUAL

TEXT 3.6.4.1 5 03/16/2006
Title: CONTAINMENT SYSTEMS SECONDARY CONTAINMENT

TEXT 3.6.4.2 2 01/03/2005
Title: CONTAINMENT SYSTEMS SECONDARY CONTAINMENT ISOLATION VALVES (SCIVS)

TEXT 3.6.4.3 3 10/24/2005
Title: CONTAINMENT SYSTEMS STANDBY GAS TREATMENT (SGT) SYSTEM

TEXT 3.7.1 0 11/15/2002
Title: PLANT SYSTEMS RESIDUAL HEAT REMOVAL SERVICE WATER (RHRSW) SYSTEM AND THE ULTIMATE HEAT SINK (UHS)

TEXT 3.7.2 1 11/09/2004
Title: PLANT SYSTEMS EMERGENCY SERVICE WATER (ESW) SYSTEM

TEXT 3.7.3 0 11/15/2002
Title: PLANT SYSTEMS CONTROL ROOM EMERGENCY OUTSIDE AIR SUPPLY (CREOAS) SYSTEM

TEXT 3.7.4 0 11/15/2002
Title: PLANT SYSTEMS CONTROL ROOM FLOOR COOLING SYSTEM

TEXT 3.7.5 0 11/15/2002
Title: PLANT SYSTEMS MAIN CONDENSER OFFGAS

TEXT 3.7.6 1 01/17/2005
Title: PLANT SYSTEMS MAIN TURBINE BYPASS SYSTEM

TEXT 3.7.7 0 11/15/2002
Title: PLANT SYSTEMS SPENT FUEL STORAGE POOL WATER LEVEL

TEXT 3.8.1 3 10/05/2005
Title: ELECTRICAL POWER SYSTEMS AC SOURCES - OPERATING

TEXT 3.8.2 0 11/15/2002
Title: ELECTRICAL POWER SYSTEMS AC SOURCES - SHUTDOWN

Manual Name: TSB1

Manual Title: TECHNICAL SPECIFICATION BASES UNIT 1 MANUAL

TEXT 3.8.3 0 11/15/2002
Title: ELECTRICAL POWER SYSTEMS DIESEL FUEL OIL, LUBE OIL, AND STARTING AIR

TEXT 3.8.4 0 11/15/2002
Title: ELECTRICAL POWER SYSTEMS DC SOURCES - OPERATING

TEXT 3.8.5 0 11/15/2002
Title: ELECTRICAL POWER SYSTEMS DC SOURCES - SHUTDOWN

TEXT 3.8.6 0 11/15/2002
Title: ELECTRICAL POWER SYSTEMS BATTERY CELL PARAMETERS

TEXT 3.8.7 1 10/05/2005
Title: ELECTRICAL POWER SYSTEMS DISTRIBUTION SYSTEMS - OPERATING

TEXT 3.8.8 0 11/15/2002
Title: ELECTRICAL POWER SYSTEMS DISTRIBUTION SYSTEMS - SHUTDOWN

TEXT 3.9.1 0 11/15/2002
Title: REFUELING OPERATIONS REFUELING EQUIPMENT INTERLOCKS

TEXT 3.9.2 0 11/15/2002
Title: REFUELING OPERATIONS REFUEL POSITION ONE-ROD-OUT INTERLOCK

TEXT 3.9.3 0 11/15/2002
Title: REFUELING OPERATIONS CONTROL ROD POSITION

TEXT 3.9.4 0 11/15/2002
Title: REFUELING OPERATIONS CONTROL ROD POSITION INDICATION

TEXT 3.9.5 0 11/15/2002
Title: REFUELING OPERATIONS CONTROL ROD OPERABILITY - REFUELING

TEXT 3.9.6 0 11/15/2002
Title: REFUELING OPERATIONS REACTOR PRESSURE VESSEL (RPV) WATER LEVEL

Manual Name: TSB1

Manual Title: TECHNICAL SPECIFICATION BASES UNIT 1 MANUAL

TEXT 3.9.7 0 11/15/2002
Title: REFUELING OPERATIONS RESIDUAL HEAT REMOVAL (RHR) - HIGH WATER LEVEL

TEXT 3.9.8 0 11/15/2002
Title: REFUELING OPERATIONS RESIDUAL HEAT REMOVAL (RHR) - LOW WATER LEVEL

TEXT 3.10.1 0 11/15/2002
Title: SPECIAL OPERATIONS INSERVICE LEAK AND HYDROSTATIC TESTING OPERATION

TEXT 3.10.2 0 11/15/2002
Title: SPECIAL OPERATIONS REACTOR MODE SWITCH INTERLOCK TESTING

TEXT 3.10.3 0 11/15/2002
Title: SPECIAL OPERATIONS SINGLE CONTROL ROD WITHDRAWAL - HOT SHUTDOWN

TEXT 3.10.4 0 11/15/2002
Title: SPECIAL OPERATIONS SINGLE CONTROL ROD WITHDRAWAL - COLD SHUTDOWN

TEXT 3.10.5 0 11/15/2002
Title: SPECIAL OPERATIONS SINGLE CONTROL ROD DRIVE (CRD) REMOVAL - REFUELING

TEXT 3.10.6 0 11/15/2002
Title: SPECIAL OPERATIONS MULTIPLE CONTROL ROD WITHDRAWAL - REFUELING

TEXT 3.10.7 0 11/15/2002
Title: SPECIAL OPERATIONS CONTROL ROD TESTING - OPERATING

TEXT 3.10.8 1 04/12/2006
Title: SPECIAL OPERATIONS SHUTDOWN MARGIN (SDM) TEST - REFUELING

B2.0 SAFETY LIMITS (SLs) B2.0-1
B2.1.1 Reactor Core SLs B2.0-1
B2.1.2 Reactor Coolant System (RCS) Pressure SL B2.0-7

B3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY B3.0-1
B3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY TS/B3.0-10

B3.1 REACTIVITY CONTROL SYSTEMS B3.1-1
B3.1.1 Shutdown Margin (SDM) B3.1-1
B3.1.2 Reactivity Anomalies B3.1-8
B3.1.3 Control Rod OPERABILITY B3.1-13
B3.1.4 Control Rod Scram Times B3.1-22
B3.1.5 Control Rod Scram Accumulators B3.1-29
B3.1.6 Rod Pattern Control TS/B3.1-34
B3.1.7 Standby Liquid Control (SLC) System B3.1-39
B3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves B3.1-47

B3.2 POWER DISTRIBUTION LIMITS TS/B3.2-1
B3.2.1 Average Planar Linear Heat Generation Rate (APLHGR) TS/B3.2-1
B3.2.2 Minimum Critical Power Ratio (MCPR) TS/B3.2-5
B3.2.3 Linear Heat Generation Rate (LHGR) B3.2-10
B3.2.4 Average Power Range Monitor (APRM) Gain and Setpoints B3.2-14

B3.3 INSTRUMENTATION TS/B3.3-1
B3.3.1.1 Reactor Protection System (RPS) Instrumentation TS/B3.3-1
B3.3.1.2 Source Range Monitor (SRM) Instrumentation TS/B3.3-35

B3.3.2.1 Control Rod Block Instrumentation TS/B3.3-44
B3.3.2.2 Feedwater – Main Turbine High Water Level Trip Instrumentation B3.3-55
B3.3.3.1 Post Accident Monitoring (PAM) Instrumentation TS/B3.3-64
B3.3.3.2 Remote Shutdown System B3.3-76
B3.3.4.1 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation B3.3-81
B3.3.4.2 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation B3.3-92
B3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation B3.3-101
B3.3.5.2 Reactor Core Isolation Cooling (RCIC) System Instrumentation B3.3-135
B3.3.6.1 Primary Containment Isolation Instrumentation B3.3-147
B3.3.6.2 Secondary Containment Isolation Instrumentation TS/B3.3-180
B3.3.7.1 Control Room Emergency Outside Air Supply (CREOAS) System Instrumentation B3.3-192

(continued)

B3.3 INSTRUMENTATION (continued)
B3.3.8.1 Loss of Power (LOP) Instrumentation TS/B3.3-205
B3.3.8.2 Reactor Protection System (RPS) Electric Power Monitoring B3.3-213

B3.4 REACTOR COOLANT SYSTEM (RCS) B3.4-1
B3.4.1 Recirculation Loops Operating B3.4-1
B3.4.2 Jet Pumps B3.4-10
B3.4.3 Safety/Relief Valves (S/RVs) TS/B3.4-15
B3.4.4 RCS Operational LEAKAGE B3.4-19
B3.4.5 RCS Pressure Isolation Valve (PIV) Leakage B3.4-24
B3.4.6 RCS Leakage Detection Instrumentation B3.4-30
B3.4.7 RCS Specific Activity B3.4-35
B3.4.8 Residual Heat Removal (RHR) Shutdown Cooling System – Hot Shutdown B3.4-39
B3.4.9 Residual Heat Removal (RHR) Shutdown Cooling System – Cold Shutdown B3.4-44
B3.4.10 RCS Pressure and Temperature (P/T) Limits TS/B3.4-49
B3.4.11 Reactor Steam Dome Pressure TS/B3.4-58

B3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM B3.5-1
B3.5.1 ECCS – Operating B3.5-1
B3.5.2 ECCS – Shutdown B3.5-19
B3.5.3 RCIC System TS/B3.5-25

B3.6 CONTAINMENT SYSTEMS TS/B3.6-1
B3.6.1.1 Primary Containment TS/B3.6-1
B3.6.1.2 Primary Containment Air Lock B3.6-7
B3.6.1.3 Primary Containment Isolation Valves (PCIVs) TS/B3.6-15
B3.6.1.4 Containment Pressure B3.6-41
B3.6.1.5 Drywell Air Temperature TS/B3.6-44
B3.6.1.6 Suppression Chamber-to-Drywell Vacuum Breakers TS/B3.6-47
B3.6.2.1 Suppression Pool Average Temperature B3.6-53
B3.6.2.2 Suppression Pool Water Level B3.6-59
B3.6.2.3 Residual Heat Removal (RHR) Suppression Pool Cooling B3.6-62
B3.6.2.4 Residual Heat Removal (RHR) Suppression Pool Spray B3.6-66
B3.6.3.1 Primary Containment Hydrogen Recombiners B3.6-70
B3.6.3.2 Drywell Air Flow System B3.6-76
B3.6.3.3 Primary Containment Oxygen Concentration B3.6-81
B3.6.4.1 Secondary Containment TS/B3.6-84
B3.6.4.2 Secondary Containment Isolation Valves (SCIVs) TS/B3.6-91
B3.6.4.3 Standby Gas Treatment (SGT) System TS/B3.6-101

(continued)

B3.7 PLANT SYSTEMS TS/B3.7-1
B3.7.1 Residual Heat Removal Service Water (RHRSW) System and the Ultimate Heat Sink (UHS) TS/B3.7-1
B3.7.2 Emergency Service Water (ESW) System TS/B3.7-7
B3.7.3 Control Room Emergency Outside Air Supply (CREOAS) System TS/B3.7-12
B3.7.4 Control Room Floor Cooling System TS/B3.7-19
B3.7.5 Main Condenser Offgas B3.7-24
B3.7.6 Main Turbine Bypass System TS/B3.7-27
B3.7.7 Spent Fuel Storage Pool Water Level B3.7-31

B3.8 ELECTRICAL POWER SYSTEM TS/B3.8-1
B3.8.1 AC Sources – Operating TS/B3.8-1
B3.8.2 AC Sources – Shutdown B3.8-38
B3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air B3.8-45
B3.8.4 DC Sources – Operating TS/B3.8-54
B3.8.5 DC Sources – Shutdown B3.8-66
B3.8.6 Battery Cell Parameters B3.8-71
B3.8.7 Distribution Systems – Operating B3.8-78
B3.8.8 Distribution Systems – Shutdown B3.8-86

B3.9 REFUELING OPERATIONS TS/B3.9-1
B3.9.1 Refueling Equipment Interlocks TS/B3.9-1
B3.9.2 Refuel Position One-Rod-Out Interlock B3.9-5
B3.9.3 Control Rod Position B3.9-9
B3.9.4 Control Rod Position Indication B3.9-12
B3.9.5 Control Rod OPERABILITY – Refueling B3.9-16
B3.9.6 Reactor Pressure Vessel (RPV) Water Level B3.9-19
B3.9.7 Residual Heat Removal (RHR) – High Water Level B3.9-22
B3.9.8 Residual Heat Removal (RHR) – Low Water Level B3.9-26

B3.10 SPECIAL OPERATIONS TS/B3.10-1
B3.10.1 Inservice Leak and Hydrostatic Testing Operation TS/B3.10-1
B3.10.2 Reactor Mode Switch Interlock Testing B3.10-6
B3.10.3 Single Control Rod Withdrawal – Hot Shutdown B3.10-11
B3.10.4 Single Control Rod Withdrawal – Cold Shutdown B3.10-16
B3.10.5 Single Control Rod Drive (CRD) Removal – Refueling B3.10-21
B3.10.6 Multiple Control Rod Withdrawal – Refueling B3.10-26
B3.10.7 Control Rod Testing – Operating B3.10-29
B3.10.8 SHUTDOWN MARGIN (SDM) Test – Refueling B3.10-33

TSB1 Tex: TOC
3/18/05

Section	Title	Revision
TOC	Table of Contents	8
B 2.0	SAFETY LIMITS BASES	
	Page B 2.0-1	0
	Page TS / B 2.0-2	2
	Page TS / B 2.0-3	3
	Pages TS/ B 2.0-4 and TS / B 2.0-5	2
	Page TS / B 2.0-6	1
	Pages B 2.0-7 through B 2.0-9	0
B 3.0	LCO AND SR APPLICABILITY BASES	
	Pages B 3.0-1 through B 3.0-4	0
	Pages TS / B 3.0-5 through TS / B 3.0-7	1
	Pages TS / B 3.0-8 through TS / B 3.0-9	2
	Pages TS / B 3.0-10 through TS / B 3.0-12	1
	Pages TS / B 3.0-13 through TS / B 3.0-15	2
	Pages TS / B 3.0-16 and TS / B 3.0-17	0
B 3.1	REACTIVITY CONTROL BASES	
	Pages B 3.1-1 through B 3.1-5	0
	Pages TS / B 3.1-6 and TS / B 3.1-7	1
	Pages B 3.1-8 through B 3.1-13	0
	Page TS / B 3.1-14	1
	Pages B 3.1-15 through B 3.1-22	0
	Page TS / B 3.1-23	1
	Pages B 3.1-24 through B 3.1-27	0
	Page TS / B 3.1-28	1
	Page TS / B 3.1-29	1
	Pages B 3.1-30 through B 3.1-33	0
	Pages TS / B 3.3-34 through TS / B 3.3-36	1
	Page TS / B 3.1-37	2
	Page TS / B 3.1-38	1
	Pages B 3.1-39 through B 3.1-44	0
	Page TS / B 3.1-45	1
	Pages B 3.1-46 and B 3.1-47	0
	Pages TS / B 3.1-48 and TS / B 3.1-49	1
	Page B 3.1-50	0
	Page TS / B 3.1-51	1
B 3.2	POWER DISTRIBUTION LIMITS BASES	
	Page TS / B 3.2-1	1
	Page TS / B 3.2-2	2
	Page TS / B 3.2-3	1
	Page TS / B 3.2-4	2
	Pages TS / B 3.2-5 and TS / B 3.2-6	1

Section	Title	Revision
	Page B 3.2-7	0
	Pages TS / B 3.2-8 through TS / B 3.2-10	1
	Page TS / B 3.2-11	2
	Page B 3.2-12	0
	Page TS / B 3.2-13	2
	Pages B 3.2-14 and B 3.2-15	0
	Page TS / B 3.2-16	2
	Page B 3.2-17	0
	Page TS / B 3.2-18	1
	Page TS / B 3.2-19	4
B 3.3	INSTRUMENTATION	
	Pages TS / B 3.3-1 through TS / B 3.3-4	1
	Page TS / B 3.3-5	2
	Page TS / B 3.3-6	1
	Page TS / B 3.3-7	3
	Page TS / B 3.3-7a	0
	Pages TS / B 3.3-8 through TS / B 3.3-12	3
	Pages TS / B 3.3-12a through TS / B 3.3-12c	0
	Page TS / B 3.3-13	1
	Page TS / B 3.3-14	3
	Pages TS / B 3.3-15 and TS / B 3.3-16	1
	Pages TS / B 3.3-17 and TS / B 3.3-18	3
	Pages TS / B 3.3-19	1
	Pages TS / B 3.3-20 through TS / B 3.3-22	2
	Page TS / B 3.3-22a	0
	Pages TS / B 3.3-23 and TS / B 3.3-24	2
	Pages TS / B 3.3-24a and TS / B 3.3-24b	0
	Pages TS / B 3.3-25 and TS / B 3.3-26	2
	Page TS / B 3.3-27	1
	Pages TS / B 3.3-28 through TS / B 3.3-30	3
	Page TS / B 3.3-30a	0
	Page TS / B 3.3-31	3
	Page TS / B 3.3-32	5
	Pages TS / B 3.3-32a and TS / B 3.3-32b	0
	Page TS / B 3.3-33	5
	Page TS / B 3.3-33a	0
	Page TS / B 3.3-34	1
	Pages TS / B 3.3-35 and TS / B 3.3-36	2
	Pages TS / B 3.3-37 through TS / B 3.3-43	1
	Page TS / B 3.3-44	3

Section	Title	Revision
	Pages TS / B 3.3-45 through TS / B 3.3-49	2
	Page TS / B 3.3-50	3
	Page Ts / B 3.3-51	2
	Pages TS / B 3.3-52 and TS / B 3.3-53	1
	Page TS / B 3.3-54	3
	Pages B 3.3-55 through B 3.3-63	0
	Pages TS / B 3.3-64 and TS / B 3.3-65	2
	Page TS / B 3.3-66	4
	Page TS / B 3.3-67	3
	Pages TS / B 3.3-68 and TS / B 3.3-69	4
	Pages TS / B 3.3-70 and TS / B 3.3-71	3
	Pages TS / 3.3-72 through TS / 3.3-75	2
	Page TS / B 3.3-75a	4
	Page TS / B 3.3-75b	5
	Page TS / B 3.3-75c	4
	Pages B 3.3-76 through 3.3-77	0
	Page TS / B 3.3-78	1
	Pages B 3.3-79 through B 3.3-89	0
	Page TS / B 3.3-90	1
	Page B 3.3-91	0
	Page TS / B 3.3-92 through TS / B 3.3-100	1
	Pages B 3.3-101 through B 3.3-103	0
	Page TS / B 3.3-104	1
	Pages B 3.3-105 and B 3.3-106	0
	Page TS / B 3.3-107	1
	Page B 3.3-108	0
	Page TS / B 3.3-109	1
	Pages B 3.3-110 and B 3.3-111	0
	Pages TS / B 3.3-112 and TS / B 3.3-112a	1
	Pages TS / B 3.3-113 and TS / B 3.3-114	1
	Page TS / B 3.3-115	1
	Page TS / B 3.3-116	2
	Page TS / B 3.3-117	1
	Pages B 3.3-118 through B 3.3-122	0
	Pages TS / B 3.3-123 and TS / B 3.3-124	1
	Page TS / B 3.3-124a	0
	Page B 3.3-125	0
	Page TS / B 3.3-126	1
	Page TS / B 3.3-127	1
	Pages B 3.3-128 through B 3.3-130	0
	Page TS / B 3.3-131	1
	Pages B 3.3-132 through B 3.3-137	0

Section	Title	Revision
	Page TS / B 3.3-138	1
	Pages B 3.3-139 through B 3.3-149	0
	Page TS / B 3.3-150 through TS / B 3.3-162	1
	Page TS / B 3.3-163	2
	Pages TS / B 3.3-164 through TS / B 3.3-177	1
	Pages TS / B 3.3-178 and TS / B 3.3-179	2
	Page TS / B 3.3-179a	2
	Page TS / B 3.3-179b	0
	Page TS / B 3.3-179c	0
	Page TS / B 3.3-180	1
	Page TS / B 3.3-181	2
	Pages TS / B 3.3-182 through TS / B 3.3-186	1
	Pages TS / B 3.3-187 and TS / B 3.3-188	2
	Pages TS / B 3.3-189 through TS / B 3.3-191	1
	Pages B 3.3-192 through B 3.3-204	0
	Page TS / B 3.3-205	1
	Pages B 3.3-206 through B 3.3-219	0
B 3.4	REACTOR COOLANT SYSTEM BASES	
	Pages B 3.4-1 and B 3.4-2	0
	Page TS / B 3.4-3 and Page TS / B 3.4-4	4
	Pages TS / B 3.4-5 through TS / B 3.4-9	2
	Pages B 3.4-10 through B 3.4-14	0
	Page TS / B 3.4-15	1
	Pages TS / B 3.4-16 through TS / B 3.4-18	2
	Pages B 3.4-19 through B 3.4-27	0
	Pages TS / B 3.4-28 and TS / B 3.4-29	1
	Pages B 3.4-30 and B 3.4-31	0
	Page TS / B 3.4-32	1
	Pages B 3.4-33 through B 3.4-36	0
	Page TS / B 3.4-37	1
	Pages B 3.4-38 through B 3.4-40	0
	Page TS / B 3.4-41	1
	Pages B 3.4-42 through B 3.4-48	0
	Page TS / B 3.4-49	2
	Page TS / B 3.4-50	1
	Page TS / B 3.4-51	2
	Pages TS / B 3.4-52 and TS / B 3.4-53	1
	Page TS / B 3.4-54	2
	Page TS / B 3.4-55	2
	Page TS / B 3.4-56	1
	Page TS / B 3.4-57	2
	Pages TS / B 3.4-58 through TS / B 3.4-60	1

Section	Title	Revision
B 3.5	ECCS AND RCIC BASES	
	Pages B 3.5-1 and B 3.5-2	0
	Page TS / B 3.5-3	2
	Page TS / B 3.5-4	1
	Page TS / B 3.5-5	2
	Page TS / B 3.5-6	1
	Pages B 3.5-7 through B 3.5-10	0
	Page TS / B 3.5-11	1
	Page TS / B 3.5-12	0
	Page TS / B 3.5-13	1
	Page TS / B 3.5-14 and TS / B 3.5-15	0
	Pages TS / B 3.5-16 through TS / B 3.5-18	1
	Pages B 3.5-19 through B 3.5-24	0
	Page TS / B 3.5-25	1
	Pages TS / B 3.5-26 and TS / B 3.5-27	1
	Pages B 3.5-28 through B 3.5-31	0
B 3.6	CONTAINMENT SYSTEMS BASES	
	Page TS / B 3.6-1	2
	Page TS / B 3.6-1a	3
	Pages TS / B 3.6-2 through TS / B 3.6-5	2
	Page TS / B 3.6-6	3
	Pages TS / B 3.6-6a and TS / B 3.6-6b	2
	Page TS / B 3.6-6c	0
	Pages B 3.6-7 through B 3.6-14	0
	Page TS / B 3.6-15	2
	Pages TS / B 3.6-15a and TS / B 3.6-15b	0
	Page B 3.6-16	0
	Page TS / B 3.6-17	1
	Page TS / B 3.6-17a	0
	Pages TS / B 3.6-18 and TS / B 3.6-19	0
	Page TS / B 3.6-20	1
	Page TS / B 3.6-21	2
	Page TS / B 3.6-22	1
	Page TS / B 3.6-22a	0
	Page TS / B 3.6-23	1
	Pages TS / B 3.6-24 through TS / B 3.6-25	0
	Page TS / B 3.6-26	1
	Page TS / B 3.6-27	2
	Page TS / B 3.6-28	5
	Page TS / B 3.6-29	1
	Page TS / B 3.6-30	1
	Page TS / B 3.6-31	3

Section	Title	Revision
	Page B 3.6-32	0
	Page TS / B 3.6-33	1
	Pages B 3.6-34 and B 3.6-35	0
	Page TS / B 3.6-36	1
	Page B 3.6-37	0
	Page TS / B 3.6-38	1
	Page B 3.6-39	0
	Page TS / B 3.6-40	4
	Pages B 3.6-41 through B 3.6-43	3
	Pages TS / B 3.6-44 and TS / B 3.6-45	1
	Page TS / B 3.6-46	2
	Pages TS / B 3.6-47 through TS / B 3.6-51	1
	Page TS / B 3.6-52	2
	Pages B 3.6-53 through B 3.6-63	0
	Pages TS / B 3.6-64 and TS / B 3.6-65	1
	Pages B 3.6-66 through B 3.6-72	0
	Page TS / B 3.6-73	1
	Pages B 3.6-74 through B 3.6-77	0
	Page TS / B 3.6-78	1
	Pages B 3.6-79 through B 3.3.6-83	0
	Page TS / B 3.6-84	3
	Page TS / B 3.6-85	2
	Page TS / B 3.6-86	3
	Page TS / B 3.6-87	1
	Pages TS / B 3.6-88 and TS / B 3.6-88a	2
	Page TS / B 3.6-89	3
	Page TS / B 3.6-90	2
	Page TS / B 3.6-91	3
	Pages TS / B 3.6-92 through TS / B 3.6-96	1
	Page TS / B 3.6-97	2
	Pages TS / B 3.6-98 and TS / B 3.6-99	1
	Page TS / B 3.6-100	2
	Pages TS / B 3.6-101 and TS / B 3.6-102	1
	Pages TS / B 3.6-103 and TS / B 3.6-104	2
	Page TS / B 3.6-105	3
	Pages TS / B 3.6-106 and TS / B 3.6-107	2
B 3.7	PLANT SYSTEMS BASES	
	Pages TS / B 3.7-1 through TS / B 3.7-6	2
	Page TS / B 3.7-6a	2
	Pages TS / B 3.3.7-6b and TS / B 3.7-6c	0
	Page TS / B 3.7-7	2
	Pages TS / B 3.7-8 through TS / B 3.7-11	1

Section	Title	Revision
	Pages TS / B 3.7-12 and TS / B 3.7-13	1
	Pages TS / B 3.7-14 through TS / B 3.7-18	2
	Page TS / B 3.7-18a	0
	Pages TS / B 3.7-19 through TS / B 3.7-23	1
	Pages B 3.7-24 through B 3.7-26	0
	Pages TS / B 3.7-27 through TS / B 3.7-29	4
	Page TS / B 3.7-30	2
	Pages B 3.7-31 through B 3.7-33	0
B 3.8	ELECTRICAL POWER SYSTEMS BASES	
	Pages TS / B 3.8-1 through TS / B 3.8-4	2
	Page TS / B 3.8-5	4
	Page TS / B 3.8-6	3
	Pages TS / B 3.8-7 through TS/B 3.8-8	2
	Page TS / B 3.8-9	4
	Page TS / B 3.8-10	3
	Pages TS / B 3.8-11 and TS / B 3.8-17	2
	Page TS / B 3.8-18	3
	Pages TS / B 3.8-19 through TS / B 3.8-21	2
	Pages TS / B 3.8-22 and TS / B 3.8-23	3
	Pages TS / B 3.8-24 through TS / B 3.8-37	2
	Pages B 3.8-38 through B 3.8-53	0
	Pages TS / B 3.8-54 through TS / B 3.8-61	1
	Page TS / B 3.8-62	2
	Page TS / B 3.8-63	2
	Page TS / B 3.8-64	1
	Page TS / B 3.8-65	2
	Pages B 3.8-66 through B 3.8-80	0
	Page TS / B 3.8-81	1
	Pages B 3.8-82 through B 3.8-90	0
B 3.9	REFUELING OPERATIONS BASES	
	Pages TS / B 3.9-1 and TS / B 3.9-1a	1
	Pages TS / B 3.9-2 through TS / B 3.9-4	1
	Pages B 3.9-5 through B 3.9-30	0
B 3.10	SPECIAL OPERATIONS BASES	
	Page TS / B 3.10-1	1
	Pages B 3.10-2 through B 3.10-31	0
	Page TS / B 3.10-32	1
	Page B 3.10-33	0
	Page TS / B 3.10-34	1
	Pages B 3.10-35 and B 3.10-36	0
	Pages TS / 3.10-37 and TS / B 3.10-38	1

TSB1 text LOES
3/22/06

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 Average Power Range Monitor (APRM) Gain and Setpoints

BASES

BACKGROUND

The OPERABILITY of the APRMs and their setpoints is an initial condition of all safety analyses that assume rod insertion upon reactor scram. Applicable GDCs are GDC 10, "Reactor Design," GDC 13, "Instrumentation and Control," GDC 20, "Protection System Functions," and GDC 23, "Protection against Anticipated Operation Occurrences" (Ref. 1). This LCO is provided to require the APRM gain or APRM flow biased scram setpoints to be adjusted when operating under conditions of excessive power peaking to maintain acceptable margin to the fuel transient mechanical design limit (i.e., Protection Against Power Transient (PAPT) limit).

The condition of excessive power peaking is determined by the ratio of the actual power peaking to the limiting power peaking at RTP. This ratio is equal to the ratio of the core limiting MFLPD to the Fraction of RTP (FRTP), where FRTP is the measured THERMAL POWER divided by the RTP. Excessive power peaking exists when:

$$\frac{\text{MFLPD}}{\text{FRTP}} > 1,$$

indicating that MFLPD is not decreasing proportionately to the overall power reduction, or conversely, that power peaking is increasing. To maintain margins similar to those at RTP conditions, the excessive power peaking is compensated by a gain adjustment on the APRMs or adjustment of the APRM setpoints. Either of these adjustments has effectively the same result as maintaining MFLPD less than or equal to FRTP to ensure the PAPT limits are not violated under steady state or transient conditions.

The normally selected APRM setpoints position the scram above the upper bound of the normal power/flow operating region that has been considered in the design of the fuel rods. The setpoints are flow biased with a slope that approximates the upper flow control line, such that an approximately constant margin is maintained between the flow biased trip level and the upper operating boundary for core flows in excess of about 45% of rated core flow. In the range of infrequent operations below 45% of rated core flow,

(continued)

BASES

BACKGROUND (continued)

the margin to scram is reduced because of the nonlinear core flow versus drive flow relationship. The normally selected APRM setpoints are supported by the analyses that concentrate on events initiated from rated conditions. Design experience has shown that minimum deviations occur within expected margins to operating limits (APLHGR, LHGR and MCPR), at rated conditions for normal power distributions. However, at other than rated conditions, control rod patterns can be established that significantly reduce the margin to thermal limits. Therefore, the flow biased APRM scram setpoints may be reduced during operation when the combination of THERMAL POWER and MFLPD indicates an excessive power peaking distribution.

The APRM neutron flux signal is also conditioned to more closely follow the fuel cladding heat flux during power transients. The APRM neutron flux signal is a measure of the core thermal power during steady state operation. During power transients, the APRM signal leads the actual core thermal power response because of the fuel thermal time constant. Therefore, on power increase transients, the APRM signal provides a conservatively high measure of core thermal power. By passing the APRM signal through an electronic filter with a time constant approximately equal to, that of the fuel thermal time constant, an APRM transient response that more closely follows actual fuel cladding heat flux is obtained. The delayed response of the filtered APRM signal allows the flow biased APRM scram levels to be positioned closer to the upper bound of the normal power and flow range, without unnecessarily causing reactor scrams during short duration neutron flux spikes. These spikes can be caused by insignificant transients such as performance of main steam line valve surveillances or momentary flow increases of only several percent.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the APRM gain or setpoint adjustments are that acceptable margins be maintained to the fuel transient mechanical design limit (PAPT).

FSAR safety analyses (Refs. 2 and 3) concentrate on the rated power condition for which the minimum expected margin to the operating limits (APLHGR, LHGR and MCPR) occurs. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

(MCPR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limit the initial margins to these operating limits at rated conditions so that specified acceptable fuel design limits are met during transients initiated from rated conditions. At initial power levels less than rated levels, the margin degradation of either the LHGR or the MCPR during a transient can be greater than at the rated condition event. This greater margin degradation during the transient is primarily offset by the larger initial margin to limits at the lower than rated power levels. However, power distributions can be hypothesized that would result in reduced margins to the pre-transient operating limit. When combined with the increased severity of certain transients at other than rated conditions, the SLs could be approached. At substantially reduced power levels, highly peaked power distributions could be obtained that could reduce thermal margins to the minimum levels required for transient events. To prevent or mitigate such situations, the MCPR margin degradation at reduced power and flow is factored into the power and flow dependent MCPR limits (LCO 3.2.2). For LHGR (Ref. 4), either the APRM gain is adjusted upward by the ratio of the core limiting MFLPD to the FRTP, or the flow biased APRM scram level is reduced by the ratio of FRTP to the core limiting MFLPD. The adjustment in the APRM gain can be performed provided it is during power ascension up to 90% of RATED THERMAL POWER, that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER, the required gain adjustment increment does not exceed 10% of RATED THERMAL POWER, and a notice of the adjustment is posted on the reactor control panel. Either of these adjustments effectively counters the increased severity of some events at other than rated conditions by proportionally increasing the APRM gain or proportionally lowering the flow biased APRM scram setpoints, dependent on the increased peaking that may be encountered.

The APRM gain and setpoints satisfy Criteria 2 and 3 of the NRC Policy Statement (Ref. 5).

(continued)

BASES (continued)

LCO

Meeting any one of the following conditions ensures acceptable operating margin to the transient mechanical design limit (PAPT) for events described above:

a. Limiting excess power peaking;

b. Reducing the APRM flow biased neutron flux upscale scram setpoints by multiplying the APRM setpoints by the ratio of FRTP and the core limiting value of MFLPD; or

c. Increasing APRM gains to cause the APRM to read greater than 100 times MFLPD (in %). This condition is to account for the reduction in margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

MFLPD is the ratio of the limiting LHGR to the LHGR limit for APRM setpoints for the specific bundle type. As power is reduced, if the design power distribution is maintained, MFLPD is reduced in proportion to the reduction in power. However, if power peaking increases above the design value, the MFLPD is not reduced in proportion to the reduction in power. Under these conditions, the APRM gain is adjusted upward or the APRM flow biased scram setpoints are reduced accordingly. When the reactor is operating with peaking less than the design value, it is not necessary to modify the APRM flow biased scram setpoints. Adjusting APRM gain or setpoints is equivalent to MFLPD less than or equal to FRTP, as stated in the LCO.

For compliance with LCO Item b (APRM setpoint adjustment) or Item c (APRM gain adjustment), only APRMs required to be OPERABLE per LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," are required to be adjusted. In addition, each APRM may be allowed to have its gain or setpoints adjusted independently of other APRMs that are having their gain or setpoints adjusted.

APPLICABILITY

The MFLPD limit, APRM gain adjustment, and APRM flow biased scram and associated setdowns are provided to ensure that the fuel transient mechanical design limit (PAPT) is not violated during design basis transients. As discussed in the Bases for LCO 3.2.1, LCO 3.2.2, and LCO 3.2.3,

(continued)

BASES

APPLICABILITY (continued)

sufficient margin to these limits exists below 25% RTP and, therefore, these requirements are only necessary when the reactor is operating at ≥ 25% RTP.

ACTIONS

A.1

If the APRM gain or setpoints are not within limits while the MFLPD has exceeded FRTP, the margin to the fuel transient mechanical design limit (PAPT) may be reduced. Therefore, prompt action should be taken to restore the MFLPD to within its required limit or make acceptable APRM adjustments such that the plant is operating within the assumed margin of the safety analyses.

The 6 hour Completion Time is normally sufficient to restore either the MFLPD to within limits or the APRM gain or setpoints to within limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LCO not met.

The APRM setpoints include the APRM Simulated Thermal Power – High RPS scram setpoint, LCO 3.3.1.1 "RPS Instrumentation," Function 2.b, and APRM Simulated Thermal Power – High rod block setpoint, Technical Requirements Manual (TRM) TRO 3.1.3 "Control Rod Block Instrumentation", Function 1.b.

B.1

If MFLPD cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER is reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2

The MFLPD is required to be calculated and compared to FRTP or APRM gain or setpoints to ensure that the reactor

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2 (continued)

is operating within the assumptions of the safety analysis. These SRs are only required to determine the MFLPD and, assuming MFLPD is greater than FRTP, the appropriate gain or setpoint, and is not intended to be a CHANNEL FUNCTIONAL TEST for the APRM gain or flow biased neutron flux scram functions. The 24 hour Frequency of SR 3.2.4.1 is chosen to coincide with the determination of other thermal limits, specifically those for the LHGR (LCO 3.2. 3). The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 24 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels and because the MFLPD must be calculated prior to exceeding 50% RTP unless performed in the previous 24 hours. When MFLPD is greater than FRTP, SR 3.2.4.2 must be performed. The 12 hour Frequency of SR 3.2.4.2 requires a more frequent verification when MFLPD is greater than the fraction of rated thermal power (FRTP) because more rapid changes in power distribution are typically expected.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 13, GDC 20, and GDC 23.

2. FSAR, Section 4.

3. FSAR, Section 15.

4. ANF-89-98(P)(A) Revision 1 and Revision 1 Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995.

5. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

B 3.3 INSTRUMENTATION

B 3.3.1.1 Reactor Protection System (RPS) Instrumentation

BASES

BACKGROUND

The RPS initiates a reactor scram when one or more monitored parameters exceed their specified limits, to preserve the integrity of the fuel cladding and the Reactor Coolant System (RCS) and minimize the energy that must be absorbed following a loss of coolant accident (LOCA). This can be accomplished either automatically or manually.

The protection and monitoring functions of the RPS have been designed to ensure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance. The LSSS are defined in this Specification as the Allowable Values, which, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits, including Safety Limits (SLs) during Design Basis Accidents (DBAs).

The RPS, as shown in the FSAR, Figure 7.2-1 (Ref. 1), includes sensors, relays, bypass circuits, and switches that are necessary to cause initiation of a reactor scram. Functional diversity is provided by monitoring a wide range of dependent and independent parameters. The input parameters to the scram logic are from instrumentation that monitors reactor vessel water level, reactor vessel pressure, neutron flux, main steam line isolation valve position, turbine control valve (TCV) fast closure trip oil pressure, turbine stop valve (TSV) position, drywell pressure, and scram discharge volume (SDV) water level, as well as reactor mode switch in shutdown position and manual scram signals. There are at least four redundant sensor input signals from each of these parameters (with the exception of the reactor mode switch in shutdown scram signal). When the setpoint is reached, the channel sensor actuates, which then outputs an RPS trip signal to the trip logic. Table B 3.3.1.1-1 summarizes the diversity of sensors capable of initiating scrams during anticipated operating transients typically analyzed.

The RPS is comprised of two independent trip systems (A and B) with two logic channels in each trip system (logic

(continued)

BASES

BACKGROUND (continued)

channels A1 and A2, B1 and B2) as shown in Reference 1. The outputs of the logic channels in a trip system are combined in a one-out-of-two logic so that either channel can trip the associated trip system. The tripping of both trip systems will produce a reactor scram. This logic arrangement is referred to as a one-out-of-two taken twice logic. Each trip system can be reset by use of a reset switch. If a full scram occurs (both trip systems trip), a relay prevents reset of the trip systems for 10 seconds after the full scram signal is received. This 10 second delay on reset ensures that the scram function will be completed.

Two AC powered scram pilot solenoids are located in the hydraulic control unit for each control rod drive (CRD). Each scram pilot valve is operated with the solenoids normally energized. The scram pilot valves control the air supply to the scram inlet and outlet valves for the associated CRD. When either scram pilot valve solenoid is energized, air pressure holds the scram valves closed and, therefore, both scram pilot valve solenoids must be de-energized to cause a control rod to scram. The scram valves control the supply and discharge paths for the CRD water during a scram. One of the scram pilot valve solenoids for each CRD is controlled by trip system A, and the other solenoid is controlled by trip system B. Any trip of trip system A in conjunction with any trip in trip system B results in de-energizing both solenoids, air bleeding off, scram valves opening, and control rod scram.

The DC powered backup scram valves, which energize on a scram signal to depressurize the scram air header, are also controlled by the RPS. Additionally, the RPS System controls the SDV vent and drain valves such that when both trip systems trip, the SDV vent and drain valves close to isolate the SDV.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The actions of the RPS are assumed in the safety analyses of References 3, 4, 5 and 6. The RPS initiates a reactor scram before the monitored parameter values reach the Allowable Values, specified by the setpoint methodology and listed in Table 3.3.1.1-1 to preserve the integrity of the fuel cladding, the reactor coolant pressure boundary (RCPB), and

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

the containment by minimizing the energy that must be absorbed following a LOCA.

RPS instrumentation satisfies Criterion 3 of the NRC Policy Statement. (Ref. 2)

Functions not specifically credited in the accident analysis are retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The OPERABILITY of the RPS is dependent on the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.1.1-1. Each Function must have a required number of OPERABLE channels per RPS trip system, with their setpoints within the specified Allowable Value, where appropriate. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Each channel must also respond within its assumed response time.

Allowable Values are specified for each RPS Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the actual setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter reaches the setpoint, the associated device changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances,

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

instrument drift and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The OPERABILITY of scram pilot valves and associated solenoids, backup scram valves, and SDV valves, described in the Background section, are not addressed by this LCO.

The individual Functions are required to be OPERABLE in the MODES specified in the table, which may require an RPS trip to mitigate the consequences of a design basis accident or transient. To ensure a reliable scram function, a combination of Functions are required in each MODE to provide primary and diverse initiation signals.

The RPS is required to be OPERABLE in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies. Control rods withdrawn from a core cell containing no fuel assemblies do not affect the reactivity of the core and, therefore, are not required to have the capability to scram. Provided all other control rods remain inserted, the RPS function is not required. In this condition, the required SDM (LCO 3.1.1) and refuel position one-rod-out interlock (LCO 3.9.2) ensure that no event requiring RPS will occur. During normal operation in MODES 3 and 4, all control rods are fully inserted and the Reactor Mode Switch Shutdown Position control rod withdrawal block (LCO 3.3.2.1) does not allow any control rod to be withdrawn. Under these conditions, the RPS function is not required to be OPERABLE. The exception to this is Special Operations (LCO 3.10.3 and LCO 3.10.4) which ensure compliance with appropriate requirements.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

Intermediate Range Monitor (IRM)

1.a. Intermediate Range Monitor Neutron Flux—High

The IRMs monitor neutron flux levels from the upper range of the source range monitor (SRM) to the lower range of the average power range monitors (APRMs). The IRMs are capable of generating trip signals that can be used to prevent fuel

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a. Intermediate Range Monitor Neutron Flux-High (continued)

damage resulting from abnormal operating transients in the intermediate power range. In this power range, the most significant source of reactivity change is due to control rod withdrawal. The IRM provides diverse protection for the rod worth minimizer (RWM), which monitors and controls the movement of control rods at low power. The RWM prevents the withdrawal of an out of sequence control rod during startup that could result in an unacceptable neutron flux excursion (Ref. 5). The IRM provides mitigation of the neutron flux excursion. To demonstrate the capability of the IRM System to mitigate control rod withdrawal events, generic analyses have been performed (Ref. 3) to evaluate the consequences of control rod withdrawal events during startup that are mitigated only by the IRM. This analysis, which assumes that one IRM channel in each trip system is bypassed, demonstrates that the IRMs provide protection against local control rod withdrawal errors and results in peak fuel energy depositions below the 170 cal/gm fuel failure threshold criterion.

The IRMs are also capable of limiting other reactivity excursions during startup, such as cold water injection events, although no credit is specifically assumed.

The IRM System is divided into two trip systems, with four IRM channels inputting to each trip system. The analysis of Reference 3 assumes that one channel in each trip system is bypassed. Therefore, six channels with three channels in each trip system are required for IRM OPERABILITY to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This trip is active in each of the 10 ranges of the IRM, which must be selected by the operator to maintain the neutron flux within the monitored level of an IRM range.

The analysis of Reference 3 has adequate conservatism to permit an IRM Allowable Value of 122 divisions of a 125 division scale.

The Intermediate Range Monitor Neutron Flux—High Function must be OPERABLE during MODE 2 when control rods may be withdrawn and the potential for criticality exists. In

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a. Intermediate Range Monitor Neutron Flux—High (continued)

MODE 5, when a cell with fuel has its control rod withdrawn, the IRMs provide monitoring for and protection against unexpected reactivity excursions. In MODE 1, the APRM System and the RWM provide protection against control rod withdrawal error events and the IRMs are not required. In addition, the Function is automatically bypassed when the Reactor Mode Switch is in the Run position.

1.b. Intermediate Range Monitor—Inop

This trip signal provides assurance that a minimum number of IRMs are OPERABLE. Anytime an IRM mode switch is moved to any position other than "Operate," the detector voltage drops below a preset level, or when a module is not plugged in, an inoperative trip signal will be received by the RPS unless the IRM is bypassed. Since only one IRM in each trip system may be bypassed, only one IRM in each RPS trip system may be inoperable without resulting in an RPS trip signal.

This Function was not specifically credited in the accident analysis but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

Six channels of Intermediate Range Monitor—Inop with three channels in each trip system are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal.

Since this Function is not assumed in the safety analysis, there is no Allowable Value for this Function.

This Function is required to be OPERABLE when the Intermediate Range Monitor Neutron Flux—High Function is required.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Average Power Range Monitor (APRM)

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases. The APRM channels receive input signals from the local power range monitors (LPRMs) within the reactor core to provide an indication of the power distribution and local power changes. The APRM channels average these LPRM signals to provide a continuous indication of average reactor power from a few percent to greater than RTP. Each APRM channel also includes an Oscillation Power Range Monitor (OPRM) Upscale Function which monitors small groups of LPRM signals to detect thermal-hydraulic instabilities.

The APRM trip System is divided into four APRM channels and four 2-out-of-4 Voter channels. Each APRM channel provides inputs to each of the four voter channels. The four voter channels are divided into two groups of two each with each group of two providing inputs to one RPS trip system. The system is designed to allow one APRM channel, but no voter channels, to be bypassed. A trip from any one unbypassed APRM will result in a "half-trip" in all four of the voter channels, but no trip inputs to either RPS trip system.

APRM trip Functions 2.a, 2.b, 2.c, and 2.d are voted independently from OPRM Trip Function 2.f. Therefore, any Function 2.a, 2.b, 2.c, or 2.d trip from any two unbypassed APRM channels will result in a full trip in each of the four voter channels, which in turn results in two trip inputs into each RPS trip system logic channel (A1, A2, B1, and B2), thus resulting in a full scram signal. Similarly, a Function 2.f trip from any two unbypassed APRM channels will result in a full trip from each of the four voter channels.

Three of the four APRM channels and all four of the voter channels are required to be OPERABLE to ensure that no single failure will preclude a scram on a valid signal. In addition, to provide adequate coverage of the entire core consistent with the design bases for the APRM Functions 2.a, 2.b, and 2.c, at least [20] LPRM inputs with at least three LPRM inputs from each of the four axial levels at which the LPRMs are located must be OPERABLE for each APRM channel, with no more than [9], LPRM detectors declared inoperable since the most recent APRM gain calibration. Per Reference 23, the minimum input requirement for an APRM channel with 43 LPRM inputs is determined given that the total number of LPRM outputs used as inputs to an APRM channel that may be bypassed shall not exceed twenty-three (23). Hence, (20) LPRM inputs

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Average Power Range Monitor (APRM) (continued)

needed to be operable. For the OPRM Trip Function 2.f, each LPRM in an APRM channel is further associated in a pattern of OPRM "cells," as described in References 17 and 18. Each OPRM cell is capable of producing a channel trip signal.

2.a. Average Power Range Monitor Neutron Flux—High (Setdown)

For operation at low power (i.e., MODE 2), the Average Power Range Monitor Neutron Flux—High (Setdown) Function is capable of generating a trip signal that prevents fuel damage resulting from abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux—High (Setdown) Function will provide a secondary scram to the Intermediate Range Monitor Neutron Flux—High Function because of the relative setpoints. With the IRMs at Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux—High (Setdown) Function will provide the primary trip signal for a corewide increase in power.

The Average Power Range Monitor Neutron Flux – High (Setdown) Function together with the IRM – High Function provide mitigation for the control rod withdrawal event during startup (Section 15.4.1 of Ref. 5). Also, the Function indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed 25% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < 25% RTP.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Average Power Range Monitor Neutron Flux—High (Setdown) (continued)

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 25% RTP.

The Average Power Range Monitor Neutron Flux—High (Setdown) Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists. In MODE 1, the Average Power
Range Monitor Neutron Flux—High Function provides protection against reactivity transients and the RWM protects against control rod withdrawal error events.

2.b. Average Power Range Monitor Simulated Thermal Power—High

The Average Power Range Monitor Simulated Thermal Power—High Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. The trip level is varied as a function of recirculation drive flow (i.e., at lower core flows, the setpoint is reduced proportional to the reduction in power experienced as core flow is reduced with a fixed control rod pattern) but is clamped at an upper limit that is always lower than the Average Power Range Monitor Neutron Flux—High Function Allowable Value. The Average Power Range Monitor Simulated Thermal Power—High Function is not credited in any plant Safety Analyses. The Average Power Range Monitor Simulated Thermal Power – High Function is set above the APRM Rod Block to provide defense in depth to the APRM Neutron Flux – High for transients where THERMAL POWER increases slowly (such as loss of feedwater heating event). During these events, the THERMAL POWER increase does not significantly lag the neutron flux response and, because of a lower trip setpoint, will initiate a scram before the high neutron flux scram. For rapid neutron flux increase events, the THERMAL POWER lags the neutron flux and the Average Power Range Monitor Neutron Flux—High Function will provide a scram signal before the Average

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor Simulated Thermal Power—High (continued)

Power Range Monitor Simulated Thermal Power—High Function setpoint is exceeded.

The Average Power Range Monitor Simulated Thermal Power – High Function uses a trip level generated based on recirculation loop drive flow (W) representative of total core flow. Each APRM channel uses one total recirculation drive flow signal. The total recirculation drive flow signal is generated by the flow processing logic, part of the APRM channel, by summing the flow calculated from two flow transmitter signal inputs, one from each of the two recirculation drive flow loops. The flow processing logic OPERABILITY is part of the APRM channel OPERABILITY requirements for this Function.

The adequacy of drive flow as a representation of core flow is ensured through drive flow alignment, accomplished by SR 3.3.1.1.20.

A note is included, applicable when the plant is in single recirculation loop operation per LCO 3.4.1, which requires reducing by ΔW the recirculation flow value used in the APRM Simulated Thermal Power – High Allowable Value equation. The Average Power Range Monitor Scram Function varies as a function of recirculation loop drive flow (W). ΔW is defined as the difference in indicated drive flow (in percent of drive flow, which produces rated core flow) between two-loop and single-loop operation at the same core flow. The value of ΔW is established to conservatively bound the inaccuracy created in the core flow/drive flow correlation due to back flow in the jet pumps associated with the inactive recirculation loop. This adjusted Allowable Value thus maintains thermal margins essentially unchanged from those for two-loop operation.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor Simulated Thermal Power—High (continued)

The THERMAL POWER time constant of < 7 seconds is based on the fuel heat transfer dynamics and provides a signal proportional to the THERMAL POWER. The simulated thermal time constant is part of filtering logic in the APRM that simulates the relationship between neutron flux and core thermal power.

The Average Power Range Monitor Simulated Thermal Power—High Function is required to be OPERABLE in MODE 1 when there is the possibility of generating excessive THERMAL POWER and potentially exceeding the SL applicable to high pressure and core flow conditions (MCPR SL). During MODES 2 and 5, other IRM and APRM Functions provide protection for fuel cladding integrity.

2.c. Average Power Range Monitor Neutron Flux—High

The Average Power Range Monitor Neutron Flux—High Function is capable of generating a trip signal to prevent fuel damage or excessive RCS pressure. For the overpressurization protection analysis of Reference 4, the Average Power Range Monitor Neutron Flux—High Function is assumed to terminate the main steam isolation valve (MSIV) closure event and, along with the safety/relief valves (S/RVs), limit the peak reactor pressure vessel (RPV) pressure to less than the ASME Code limits. The control rod drop accident (CRDA) analysis (Ref. 5) takes credit for the Average Power Range Monitor Neutron Flux—High Function to terminate the CRDA.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.c. Average Power Range Monitor Neutron Flux—High (continued)

The CRDA analysis assumes that reactor scram occurs on Average Power Range Monitor Neutron Flux - High Function.

The Average Power Range Monitor Neutron Flux—High Function is required to be OPERABLE in MODE 1 where the potential consequences of the analyzed transients could result in the SLs (e.g., MCPR and RCS pressure) being exceeded. Although the Average Power Range Monitor Neutron Flux—High Function is assumed in the CRDA analysis, which is applicable in MODE 2, the Average Power Range Monitor Neutron Flux—High (Setdown) Function conservatively bounds the assumed trip and, together with the assumed IRM trips, provides adequate protection. Therefore, the Average Power Range Monitor Neutron Flux—High Function is not required in MODE 2.

2.d. Average Power Range Monitor—Inop

Three of the four APRM channels are required to be OPERABLE for each of the APRM Functions. This Function (Inop) provides assurance that the minimum number of APRM channels are OPERABLE.

For any APRM channel, any time its mode switch is not in the "Operate" position, an APRM module required to issue a trip is unplugged, or the automatic self-test system detects a critical fault with the APRM channel, an Inop trip is sent to all four voter channels. Inop trips from two or more unbypassed APRM channels result in a trip output from each of the four voter channels to its associated trip system.

This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.d. Average Power Range Monitor—Inop (continued)

There is no Allowable Value for this Function.

This Function is required to be OPERABLE in the MODES where the APRM Functions are required.

2.e. 2-out-of-4 Voter

The 2-out-of-4 Voter Function provides the interface between the APRM Functions, including the OPRM Trip Function, and the final RPS trip system logic. As such, it is required to be OPERABLE in the MODES where the APRM Functions are required and is necessary to support the safety analysis applicable to each of those Functions. Therefore, the 2-out-of-4 Voter Function is required to be OPERABLE in MODES 1 and 2.

All four voter channels are required to be OPERABLE. Each voter channel includes self-diagnostic functions. If any voter channel detects a critical fault in its own processing, a trip is issued from that voter channel to the associated RPS trip system.

The Two-out-of-Four Logic Module includes both the 2-out-of-4 Voter hardware and the APRM Interface hardware. The 2-out-of-4 Voter Function 2.e votes APRM Functions 2.a, 2.b, 2.c, and 2.d independently of Function 2.f. This voting is accomplished by the 2-out-of-4 Voter hardware in the Two-out-of-Four Logic Module. The voter includes separate outputs to RPS for the two independently voted sets of Functions, each of which is redundant (four total outputs). The analysis in Reference 15 took credit for this redundancy in the justification of the 12-hour Completion Time for Condition A, so the voter Function 2.e must be declared inoperable if any of its functionality is inoperable. The voter Function 2.e does not need to be declared inoperable due to any failure affecting only the APRM Interface hardware portion of the Two-out-of-Four Logic Module.

There is no Allowable Value for this Function.

2.f. Oscillation Power Range Monitor (OPRM) Trip

The OPRM Trip Function provides compliance with GDC 10, "Reactor Design," and GDC 12, "Suppression of Reactor Power Oscillations" thereby providing protection from exceeding the fuel MCPR safety limit (SL) due to anticipated thermal-hydraulic power oscillations.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.f. Oscillation Power Range Monitor (OPRM) Trip (continued)

References 17, 18 and 19 describe three algorithms for detecting thermal-hydraulic instability related neutron flux oscillations: the period based detection algorithm (confirmation count and cell amplitude), the amplitude based algorithm, and the growth rate algorithm. All three are implemented in the OPRM Trip Function, but the safety analysis takes credit only for the period based detection algorithm. The remaining algorithms provide defense in depth and additional protection against unanticipated oscillations. OPRM Trip Function OPERABILITY for Technical Specification purposes is based only on the period based detection algorithm.

The OPRM Trip Function receives input signals from the local power range monitors (LPRMs) within the reactor core, which are combined into "cells" for evaluation by the OPRM algorithms. Each channel is capable of detecting thermal-hydraulic instabilities, by detecting the related neutron flux oscillations, and issuing a trip signal before the MCPR SL is exceeded. Three of the four channels are required to be OPERABLE.

The OPRM Trip is automatically enabled (bypass removed) when THERMAL POWER is ≥ 30% RTP, as indicated by the APRM Simulated Thermal Power, and reactor core flow is ≤ the value defined in the COLR, as indicated by APRM measured recirculation drive flow. This is the operating region where actual thermal-hydraulic instability and related neutron flux oscillations are expected to occur. Reference 21 includes additional discussion of OPRM Trip enable region limits.

These setpoints, which are sometimes referred to as the "auto-bypass" setpoints, establish the boundaries of the OPRM Trip enabled region. The APRM Simulated Thermal Power auto-enable setpoint has 1% deadband while the drive flow setpoint has a 2% deadband. The deadband for these setpoints is established so that it increases the enabled region once the region is entered.

The OPRM Trip Function is required to be OPERABLE when the plant is at ≥ 25% RTP. The 25% RTP level is selected to provide margin in the unlikely event that a reactor power increase transient occurring without operator action while the plant is operating below 30% RTP causes a power increase to or beyond the 30% APRM Simulated Thermal Power OPRM Trip auto-enable setpoint. This OPERABILITY requirement assures that the OPRM Trip auto-enable function will be OPERABLE when required.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.f. Oscillation Power Range Monitor (OPRM) Trip (continued)

An APRM channel is also required to have a minimum number of OPRM cells OPERABLE for the Upscale Function 2.f to be OPERABLE. The OPRM cell operability requirements are documented in the Technical Requirements Manual, TRO 3.3.9, and are established as necessary to support the trip setpoint calculations performed in accordance with methodologies in Reference 19.

An OPRM Trip is issued from an APRM channel when the period based detection algorithm in that channel detects oscillatory changes in the neutron flux, indicated by the combined signals of the LPRM detectors in a cell, with period confirmations and relative cell amplitude exceeding specified setpoints. One or more cells in a channel exceeding the trip conditions will result in a channel OPRM Trip from that channel. An OPRM Trip is also issued from the channel if either the growth rate or amplitude-based algorithms detect oscillatory changes in the neutron flux for one or more cells in that channel. (Note: To facilitate placing the OPRM Trip Function 2.f in one APRM channel in a "tripped" state, if necessary to satisfy a Required Action, the APRM equipment is conservatively designed to force an OPRM Trip output from the APRM channel if an APRM Inop condition occurs, such as when the APRM chassis keylock switch is placed in the Inop position.)

There are three "sets" of OPRM related setpoints or adjustment parameters: a) OPRM Trip auto-enable region setpoints for STP and drive flow; b) period based detection algorithm (PBDA) confirmation count and amplitude setpoints; and c) period based detection algorithm tuning parameters.

The first set, the OPRM Trip auto-enable setpoints, as discussed in the SR 3.3.1.1.19 Bases, are treated as nominal setpoints with no additional margins added. The settings are defined in the Technical Requirements Manual, TRO 3.3.9, and confirmed by SR 3.3.1.1.19. The second set, the OPRM PBDA trip setpoints, are established in accordance with methodologies defined in Reference 19, and are documented in the COLR. There are no allowable values for these setpoints. The third set, the OPRM PBDA "tuning" parameters, are established or adjusted in accordance with and controlled by requirements in the Technical Requirements Manual, TRO 3.3.9.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. Reactor Vessel Steam Dome Pressure—High

An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes the neutron flux and THERMAL POWER transferred to the reactor coolant to increase, which could challenge the integrity of the fuel cladding and the RCPB. This trip Function is assumed in the low power generator load rejection without bypass and the recirculation flow controller failure (increasing) event. However, the Reactor Vessel Steam Dome Pressure—High Function initiates a scram for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power. For the overpressurization protection analysis of Reference 4, reactor scram (the analyses conservatively assume a scram from either the Average Power Range Monitor Neutron Flux—High signal, or the Reactor Vessel Steam Dome Pressure—High signal), along with the S/RVs, limits the peak RPV pressure to less than the ASME Section III Code limits.

High reactor pressure signals are initiated from four pressure instruments that sense reactor pressure. The Reactor Vessel Steam Dome Pressure—High Allowable Value is chosen to provide a sufficient margin to the ASME Section III Code limits during the event.

Four channels of Reactor Vessel Steam Dome Pressure—High Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. The Function is

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. Reactor Vessel Steam Dome Pressure—High (continued)

required to be OPERABLE in MODES 1 and 2 when the RCS is pressurized and the potential for pressure increase exists.

4. Reactor Vessel Water Level–Low, Level 3

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, a reactor scram is initiated at Level 3 to substantially reduce the heat generated in the fuel from fission. The Reactor Vessel Water Level—Low, Level 3 Function is assumed in the analysis of the recirculation line break (Ref. 6). The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the Emergency Core Cooling Systems (ECCS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Vessel Water Level—Low, Level 3 signals are initiated from four level instruments that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

Four channels of Reactor Vessel Water Level—Low, Level 3 Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal.

The Reactor Vessel Water Level—Low, Level 3 Allowable Value is selected to ensure that during normal operation the separator skirts are not uncovered (this protects available recirculation pump net positive suction head (NPSH) from significant carryunder) and, for transients involving loss of all normal feedwater flow, initiation of the low pressure ECCS subsystems at Reactor Vessel Water—Low Low Low, Level 1 will not be required.

The Function is required in MODES 1 and 2 where considerable energy exists in the RCS resulting in the limiting transients and accidents. ECCS initiations at Reactor Vessel Water Level—Low Low, Level 2 and Low Low Low,

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4. Reactor Vessel Water Level–Low, Level 3 (continued)

Level 1 provide sufficient protection for level transients in all other MODES.

5. Main Steam Isolation Valve—Closure

MSIV closure results in loss of the main turbine and the condenser as a heat sink for the nuclear steam supply system and indicates a need to shut down the reactor to reduce heat generation. Therefore, a reactor scram is initiated on a Main Steam Isolation Valve—Closure signal before the MSIVs are completely closed in anticipation of the complete loss of the normal heat sink and subsequent overpressurization transient. However, for the overpressurization protection analysis of Reference 4, the Average Power Range Monitor Neutron Flux—High Function, along with the S/RVs, limits the peak RPV pressure to less than the ASME Code limits. That is, the direct scram on position switches for MSIV closure events is not assumed in the overpressurization analysis. Additionally, MSIV closure is assumed in the transients analyzed in Reference 5 (e.g., low steam line pressure, manual closure of MSIVs, high steam line flow). The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the ECCS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

MSIV closure signals are initiated from position switches located on each of the eight MSIVs. Each MSIV has two position switches; one inputs to RPS trip system A while the other inputs to RPS trip system B. Thus, each RPS trip system receives an input from eight Main Steam Isolation Valve—Closure channels, each consisting of one position switch. The logic for the Main Steam Isolation Valve—Closure Function is arranged such that either the inboard or outboard valve on three or more of the main steam lines must close in order for a scram to occur.

The Main Steam Isolation Valve—Closure Allowable Value is specified to ensure that a scram occurs prior to a significant reduction in steam flow, thereby reducing the severity of the subsequent pressure transient.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5. Main Steam Isolation Valve—Closure (continued)

Sixteen channels (arranged in pairs) of the Main Steam Isolation Valve—Closure Function, with eight channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude the scram from this Function on a valid signal. This Function is only required in MODE 1 since, with the MSIVs open and the heat generation rate high, a pressurization transient can occur if the MSIVs close. In addition, the Function is automatically bypassed when the Reactor Mode Switch is not in the Run position. In MODE 2, the heat generation rate is low enough so that the other diverse RPS functions provide sufficient protection.

6. Drywell Pressure—High

High pressure in the drywell could indicate a break in the RCPB. A reactor scram is initiated to minimize the possibility of fuel damage and to reduce the amount of energy being added to the coolant and the drywell. The Drywell Pressure—High Function is assumed in the analysis of the recirculation line break (Ref. 6). The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of Emergency Core Cooling Systems (ECCS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

High drywell pressure signals are initiated from four pressure instruments that sense drywell pressure. The Allowable Value was selected to be as low as possible and indicative of a LOCA inside primary containment.

Four channels of Drywell Pressure—High Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. The Function is required in MODES 1 and 2 where considerable energy exists in the RCS, resulting in the limiting transients and accidents.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

7.a, 7.b. Scram Discharge Volume Water Level – High

The SDV receives the water displaced by the motion of the CRD pistons during a reactor scram. Should this volume fill to a point where there is insufficient volume to accept the displaced water, control rod insertion would be hindered. Therefore, a reactor scram is initiated while the remaining free volume is still sufficient to accommodate the water from a full core scram. The two types of Scram Discharge Volume Water Level—High Functions are an input to the RPS logic. No credit is taken for a scram initiated from these Functions for any of the design basis accidents or transients analyzed in the FSAR. However, they are retained to ensure the scram function remains OPERABLE.

SDV water level is measured by two diverse methods. The level in each of the two SDVs is measured by two float type level switches and two level transmitters with trip units for a total of eight level signals. The outputs of these devices are arranged so that there is a signal from a level switch and a level transmitter with trip unit to each RPS logic channel. The level measurement instrumentation satisfies the recommendations of Reference 8.

The Allowable Value is chosen low enough to ensure that there is sufficient volume in the SDV to accommodate the water from a full scram.

Four channels of each type of Scram Discharge Volume Water Level—High Function, with two channels of each type in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from these Functions on a valid signal. These Functions are required in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn. At all other times, this Function may be bypassed.

8. Turbine Stop Valve—Closure

Closure of the TSVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated at the start of TSV closure in anticipation of

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

8. Turbine Stop Valve—Closure (continued)

the transients that would result from the closure of these valves. The Turbine Stop Valve—Closure Function is the primary scram signal for the turbine trip event analyzed in Reference 5. For this event, the reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the End of Cycle Recirculation Pump Trip (EOC-RPT) System, ensures that the MCPR SL is not exceeded. Turbine Stop Valve—Closure signals are initiated from position switches located on each of the four TSVs. Two independent position switches are associated with each stop valve. One of the two switches provides input to RPS trip system A; the other, to RPS trip system B. Thus, each RPS trip system receives an input from four Turbine Stop Valve—Closure channels, each consisting of one position switch. The logic for the Turbine Stop Valve—Closure Function is such that three or more TSVs must be closed to produce a scram. This Function must be enabled at THERMAL POWER ≥ 30% RTP. This is accomplished automatically by pressure instruments sensing turbine first stage pressure. Because an increase in the main turbine bypass flow can affect this function non-conservatively, THERMAL POWER is derived from first stage pressure. The main turbine bypass valves must not cause the trip Function to be bypassed when THERMAL POWER is ≥ 30% RTP.

The Turbine Stop Valve—Closure Allowable Value is selected to be high enough to detect imminent TSV closure, thereby reducing the severity of the subsequent pressure transient.

Eight channels (arranged in pairs) of Turbine Stop Valve—Closure Function, with four channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function if any three TSVs should close. This Function is required, consistent with analysis assumptions, whenever THERMAL POWER is ≥ 30% RTP. This Function is not required when THERMAL POWER is < 30% RTP since the Reactor Vessel Steam Dome Pressure—High and the Average Power Range Monitor Neutron Flux—High Functions are adequate to maintain the necessary safety margins.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCC, and APPLICABILITY (continued)

9. Turbine Control Valve Fast Closure, Trip Oil Pressure—Low

Fast closure of the TCVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated on TCV fast closure in anticipation of the transients that would result from the closure of these valves. The Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Function is the primary scram signal for the generator load rejection event analyzed in Reference 5. For this event, the reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the EOC-RPT System, ensures that the MCPR SL is not exceeded.

Turbine Control Valve Fast Closure, Trip Oil Pressure—Low signals are initiated by the electrohydraulic control (EHC) fluid pressure at each control valve. One pressure instrument is associated with each control valve, and the signal from each transmitter is assigned to a separate RPS logic channel. This Function must be enabled at THERMAL POWER ≥ 30% RTP. This is accomplished automatically by pressure instruments sensing turbine first stage pressure. Because an increase in the main turbine bypass flow can affect this function non-conservatively, THERMAL POWER is derived from first stage pressure. The main turbine bypass valves must not cause the trip Function to be bypassed when THERMAL POWER is ≥ 30% RTP.

The Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Allowable Value is selected high enough to detect imminent TCV fast closure.

Four channels of Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Function with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is ≥ 30% RTP. This Function is not required when THERMAL POWER is < 30% RTP, since the Reactor Vessel Steam Dome Pressure—High and the Average Power Range Monitor Neutron Flux—High Functions are adequate to maintain the necessary safety margins.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

10. Reactor Mode Switch—Shutdown Position

The Reactor Mode Switch—Shutdown Position Function provides signals, via the manual scram logic channels, to each of the four RPS logic channels, which are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The reactor mode switch is a single switch with four channels, each of which provides input into one of the RPS logic channels.

There is no Allowable Value for this Function, since the channels are mechanically actuated based solely on reactor mode switch position.

Four channels of Reactor Mode Switch—Shutdown Position. Function, with two channels in each trip system, are available and required to be OPERABLE. The Reactor Mode Switch—Shutdown Position Function is required to be OPERABLE in MODES 1 and 2, and MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn.

11. Manual Scram

The Manual Scram push button channels provide signals, via the manual scram logic channels, to each of the four RPS logic channels, which are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

There is one Manual Scram push button channel for each of the four RPS logic channels. In order to cause a scram it is necessary that at least one channel in each trip system be actuated.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

11. Manual Scram (continued)

There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the push buttons.

Four channels of Manual Scram with two channels in each trip system arranged in a one-out-of-two logic are available and required to be OPERABLE in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn.

ACTIONS

A Note has been provided to modify the ACTIONS related to RPS instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable RPS instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable RPS instrumentation channel.

A.1 and A.2

Because of the diversity of sensors available to provide trip signals and the redundancy of the RPS design, an allowable out of service time of 12 hours has been shown to be acceptable (Refs. 9, 15 and 16) to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided the associated Function's inoperable channel is in one trip system and the Function still maintains RPS trip capability (refer to Required Actions B.1, B.2, and C.1 Bases). If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel or the associated trip system must be placed in the tripped

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

condition per Required Actions A.1 and A.2. Placing the inoperable channel in trip (or the associated trip system in trip) would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternatively, if it is not desired to place the channel (or trip system) in trip (e.g., as in the case where placing the inoperable channel in trip would result in a full scram), Condition D must be entered and its Required Action taken.

As noted, Action A.2 is not applicable for APRM Functions 2.a, 2.b, 2.c, 2.d, or 2.f. Inoperability of one required APRM channel affects both trip systems. For that condition, Required Action A.1 must be satisfied, and is the only action (other than restoring OPERABILITY) that will restore capability to accommodate a single failure. Inoperability of more than one required APRM channel of the same trip function results in loss of trip capability and entry into Condition C, as well as entry into Condition A for each channel.

B.1 and B.2

Condition B exists when, for any one or more Functions, at least one required channel is inoperable in each trip system. In this condition provided at least one channel per trip system is OPERABLE, the RPS still maintains trip capability for that Function, but cannot accommodate a single failure in either trip system.

Required Actions B.1 and B.2 limit the time the RPS scram logic, for any Function, would not accommodate single failure in both trip systems (e.g., one-out-of-one and one-out-of-one arrangement for a typical four channel Function). The reduced reliability of this logic arrangement was not evaluated in Reference 9, 15 or 16 for the 12 hour Completion Time. Within the 6 hour allowance, the associated Function will have all required channels OPERABLE or in trip (or any combination) in one trip system.

Completing one of these Required Actions restores RPS to a reliability level equivalent to that evaluated in Reference 9, 15 and 16, which justified a 12 hour allowable out of service time as presented in Condition A. The trip system in the more degraded state should be placed in trip or, alternatively, all the inoperable channels in that trip system should be placed in trip (e.g., a trip system with two inoperable channels could be in a more degraded state than a trip system with four inoperable channels if the two inoperable channels are in the same Function while the four inoperable channels are all in different Functions). The decision of which trip system is in the more degraded state should be based on prudent judgment and take into account current plant conditions (i.e., what MODE the plant is in).

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

If this action would result in a scram, it is permissible to place the other trip system or its inoperable channels in trip.

The 6 hour Completion Time is judged acceptable based on the remaining capability to trip, the diversity of the sensors available to provide the trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse Functions, and the low probability of an event requiring the initiation of a scram.

Alternately, if it is not desired to place the inoperable channels (or one trip system) in trip (e.g., as in the case where placing the inoperable channel or associated trip system in trip would result in a scram), Condition D must be entered and its Required Action taken.

As noted, Condition B is not applicable for APRM Functions 2.a, 2.b, 2.c, 2.d, or 2.f. Inoperability of an APRM channel affects both trip systems and is not associated with a specific trip system as are the APRM 2-out-of-4 Voter (Function 2.e) and other non-APRM channels for which Condition B applies. For an inoperable APRM channel, Required Action A.1 must be satisfied, and is the only action (other than restoring OPERABILITY) that will restore capability to accommodate a single failure. Inoperability of a Function in more than one required APRM channel results in loss of trip capability for that Function and entry into Condition C, as well as entry into Condition A for each channel. Because Conditions A and C provide Required Actions that are appropriate for the inoperability of APRM Functions 2.a, 2.b, 2.c, 2.d, or 2.f, and because these Functions are not associated with specific trip systems as are the APRM 2-out-of-4 Voter and other non-APRM channels, Condition B does not apply.

C.1

Required Action C.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same trip system for the same Function result in the Function not maintaining RPS trip capability. A Function is considered to be maintaining RPS trip capability when sufficient channels are OPERABLE or in trip (or the associated trip system is in trip), such that both trip systems will generate a trip signal from the given Function on a valid signal. For the typical Function with one-out-of-two taken twice logic, this would require both trip systems to have one channel OPERABLE or in trip (or the associated trip system in trip). For Function 5 (Main Steam

(continued)

BASES

ACTIONS

C.1 (continued)

Isolation Valve—Closure), this would require both trip systems to have each channel associated with the MSIVs in three main steam lines (not necessarily the same main steam lines for both trip systems) OPERABLE or in trip (or the associated trip system in trip).

For Function 8 (Turbine Stop Valve—Closure), this would require both trip systems to have three channels, each OPERABLE or in trip (or the associated trip system in trip).

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The

(continued)

BASES

ACTIONS

C.1 (continued)

1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

D.1

Required Action D.1 directs entry into the appropriate Condition referenced in Table 3.3.1.1-1. The applicable Condition specified in the Table is Function and MODE or other specified condition dependent and may change as the Required Action of a previous Condition is completed. Each time an inoperable channel has not met any Required Action of Condition A, B, or C and the associated Completion Time has expired, Condition D will be entered for that channel and provides for transfer to the appropriate subsequent Condition.

E.1, F.1, G.1, and J.1

If the channel(s) is not restored to OPERABLE status or placed in trip (or the associated trip system placed in trip) within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. The allowed Completion Times are reasonable, based on operating experience, to reach the specified condition from full power conditions in an orderly manner and without challenging plant systems. In addition, the Completion Time of Required Actions E.1 and J.1 are consistent with the Completion Time provided in LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)."

H.1

If the channel(s) is not restored to OPERABLE status or placed in trip (or the associated trip system placed in trip) within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by immediately initiating action to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Control rods in core cells containing no fuel assemblies do not affect

(continued)

BASES

ACTIONS

H.1 (continued)

the reactivity of the core and are, therefore, not required to be inserted. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies are fully inserted.

I.1 and I.2

Required Actions I.1 and I.2 are intended to ensure that appropriate actions are taken if more than two inoperable or bypassed OPRM channels result in not maintaining OPRM trip capability.

In the 4-OPRM channel configuration, any 'two' of the OPRM channels out of the total of four and one 2-out-of-4 voter channels in each RPS trip system are required to function for the OPRM safety trip function to be accomplished. Therefore, three OPRM channels assures at least two OPRM channels can provide trip inputs to the 2-out-of-4 voter channels even in the event of a single OPRM channel failure, and the minimum of two 2-out-of-4 voter channels per RPS trip system assures at least one voter channel will be operable per RPS trip system even in the event of a single voter channel failure.

References 15 and 16 justified use of alternate methods to detect and suppress oscillations under limited conditions. The alternate methods are consistent with the guidelines identified in Reference 20. The alternate-methods procedures require increased operator awareness and monitoring for neutron flux oscillations when operating in the region where oscillations are possible. If operator observes indications of oscillation, as described in Reference 20, the operator will take the actions described by procedures, which include manual scram of the reactor. The power/flow map regions where oscillations are possible are developed based on the methodology in Reference 22. The applicable regions are contained in the COLR.

The alternate methods would adequately address detection and mitigation in the event of thermal hydraulic instability oscillations. Based on industry operating experience with actual instability oscillations, the operator would be able to recognize instabilities during this time and take action to suppress them through a manual scram. In addition, the OPRM system may still be available to provide alarms to the operator if the onset of oscillations were to occur.

The 12-hour allowed Completion Time for Required Action I.1 is based on engineering judgment to allow orderly transition to the alternate methods

(continued)

BASES

ACTIONS

I.1 and I.2 (continued)

while limiting the period of time during which no automatic or alternate detect and suppress trip capability is formally in place. Based on the small probability of an instability event occurring at all, the 12 hours is judged to be reasonable.

The 120-day allowed Completion Time, the time that was evaluated in References 15 and 16, is considered adequate because with operation minimized in regions where oscillations may occur and implementation of the alternate methods, the likelihood of an instability event that could not be adequately handled by the alternate methods during this 120-day period was negligibly small.

The primary purpose of Required Actions I.1 and I.2 is to allow an orderly completion, without undue impact on plant operation, of design and verification activities required to correct unanticipated equipment design or functional problems that cause OPRM Trip Function INOPERABILITY in all APRM channels that cannot reasonably be corrected by normal maintenance or repair actions. These Required Actions are not intended and were not evaluated as a routine alternative to returning failed or inoperable equipment to OPERABLE status.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each RPS instrumentation Function are located in the SRs column of Table 3.3.1.1-1.

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the associated Function maintains RPS trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Refs. 9, 15 and 16) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the RPS will trip when necessary.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.1 and SR 3.3.1.1.2

Performance of the CHANNEL CHECK 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 instrument channels could be an indication of excessive instrument drift in one of the channels or 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 3.3.1.1.1 and SR 3.3.1.1.2 (continued)

Agreement criteria which are determined by the plant staff based on an investigation of a combination of the channel instrument uncertainties, may be used to support this parameter comparison and include indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit, and does not necessarily indicate the channel is Inoperable.

The Frequency of once every 12 hours for SR 3.3.1.1.1 is based upon operating experience that demonstrates that channel failure is rare. The Frequency of once every 24 hours for SR 3.3.1.1.2 is based upon operating experience that demonstrates that channel failure is rare and the evaluation in References 15 and 16.

The CHANNEL CHECK supplements less formal checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.1.3

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoints," allows the APRMs to be reading greater than actual THERMAL POWER to compensate for localized power peaking. When this adjustment is made, the requirement for the APRMs to indicate within 2% RTP of calculated power is modified to require the APRMs to indicate within 2% RTP of calculated MFLPD times 100. The Frequency of once per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading between performances of SR 3.3.1.1.8.

A restriction to satisfying this SR when < 25% RTP is provided that requires the SR to be met only at ≥ 25% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < 25% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR, LHGR and APLHGR). At ≥ 25% RTP, the Surveillance is required to have been satisfactorily performed within the last 7 days, in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above 25% if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding 25% RTP. Twelve hours is based on operating experience and in

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.3 (continued)

consideration of providing a reasonable time in which to complete the SR.

SR 3.3.1.1.4

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function.

As noted, SR 3.3.1.1.4 is not required to be performed when entering MODE 2 from MODE 1, since testing of the MODE 2 required IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This allows entry into MODE 2 if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.4 (continued)

performed within 12 hours after entering MODE 2 from MODE 1. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

A Frequency of 7 days provides an acceptable level of system average unavailability over the Frequency interval and is based on reliability analysis (Ref. 9).

SR 3.3.1.1.5

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. A Frequency of 7 days provides an acceptable level of system average availability over the Frequency and is based on the reliability analysis of Reference 9. (The Manual Scram Function's CHANNEL FUNCTIONAL TEST Frequency was credited in the analysis to extend many automatic scram Functions' Frequencies.)

SR 3.3.1.1.6 and SR 3.3.1.1.7

These Surveillances are established to ensure that no gaps in neutron flux indication exist from subcritical to power operation for monitoring core reactivity status.

The overlap between SRMs and IRMs is required to be demonstrated to ensure that reactor power will not be increased into a neutron flux region without adequate indication. The overlap is demonstrated prior to fully withdrawing the SRMs from the core. Demonstrating the overlap prior to fully withdrawing the SRMs from the core is required to ensure the SRMs are on-scale for the overlap demonstration.

The overlap between IRMs and APRMs is of concern when reducing power into the IRM range. On power increases, the system design will prevent further increases (by initiating a rod block) if adequate overlap is not maintained. Overlap between IRMs and APRMs exists when sufficient IRMs and APRMs concurrently have onscale readings such that the transition between MODE 1 and MODE 2 can be made without either APRM downscale rod block, or IRM upscale rod block. Overlap

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.6 and SR 3.3.1.1.7 (continued)

between SRMs and IRMs similarly exists when, prior to fully withdrawing the SRMs from the core, IRMs are above mid-scale on range 1 before SRMs have reached the upscale rod block.

As noted, SR 3.3.1.1.7 is only required to be met during entry into MODE 2 from MODE 1. That is, after the overlap requirement has been met and indication has transitioned to the IRMs, maintaining overlap is not required (APRMs may be reading downscale once in MODE 2).

If overlap for a group of channels is not demonstrated (e.g., IRM/APRM overlap), the reason for the failure of the Surveillance should be determined and the appropriate channel(s) declared inoperable. Only those appropriate channels that are required in the current MODE or condition should be declared inoperable.

A Frequency of 7 days is reasonable based on engineering judgment and the reliability of the IRMs and APRMs.

SR 3.3.1.1.8

LPRM gain settings are determined from the local flux profiles that are either measured by the Traversing Incore Probe (TIP) System at all functional locations or calculated for TIP locations that are not functional. The methodology used to develop the power distribution limits considers the uncertainty for both measured and calculated local flux profiles. This methodology assumes that all the TIP locations are functional for the first LPRM calibration following a refueling outage, and a minimum of 25 functional TIP locations for subsequent LPRM calibrations. The calibrated LPRMs establish the relative local flux profile for appropriate representative input to the APRM System. The 1000 MWD/MT Frequency is based on operating experience with LPRM sensitivity changes.

SR 3.3.1.1.9 and SR 3.3.1.1.14

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.9 and SR 3.3.1.1.14 (continued)

intended function. The 92 day Frequency of SR 3.3.1.1.9 is based on the reliability analysis of Reference 9.

SR 3.3.1.1.9 is modified by a Note that provides a general exception to the definition of CHANNEL FUNCTIONAL TEST. This exception is necessary because the design of instrumentation does not facilitate functional testing of all required contacts of the relay which input into the combinational logic. (Reference 10) Performance of such a test could result in a plant transient or place the plant in an undo risk situation. Therefore, for this SR, the CHANNEL FUNCTIONAL TEST verifies acceptable response by verifying the change of state of the relay which inputs into the combinational logic. The required contacts not tested during the CHANNEL FUNCTIONAL TEST are tested under the LOGIC SYSTEM FUNCTIONAL TEST, SR 3.3.1.1.15. This is acceptable because operating experience shows that the contacts not tested during the CHANNEL FUNCTIONAL TEST normally pass the LOGIC SYSTEM FUNCTIONAL TEST, and the testing methodology minimizes the risk of unplanned transients.

The 24 month Frequency of SR 3.3.1.1.14 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.

SR 3.3.1.1.10, SR 3.3.1.1.11, SR 3.3.1.1.13, and SR 3.3.1.1.18

A CHANNEL CALIBRATION verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

Note 1 for SR 3.3.1.1.18 states that neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 7 day calorimetric calibration (SR 3.3.1.1.3) and the 1000 MWD/MT LPRM

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.10, SR 3.3.1.1.11, SR 3.3.1.1.13 and SR 3.3.1.1.18 (continued)

calibration against the TIPs (SR 3.3.1.1.8).

A Note is provided for SR 3.3.1.1.11 that requires the IRM SRs to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM and IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

A second note is provided for SR 3.3.1.1.18 that requires that the recirculation flow (drive flow) transmitters, which supply the flow signal to the APRMs, be included in the SR for Functions 2.b and 2.f. The APRM Simulated Thermal Power-High Function (Function 2.b) and the OPRM Trip Function (Function 2.f) both require a valid drive flow signal. The APRM Simulated Thermal Power-High Function uses drive flow to vary the trip setpoint. The OPRM Trip Function uses drive flow to automatically enable or bypass the OPRM Trip output to the RPS. A CHANNEL CALIBRATION of the APRM drive flow signal requires both calibrating the drive flow transmitters and the processing hardware in the APRM equipment. SR 3.3.1.1.20 establishes a valid drive flow / core flow relationship. Changes throughout the cycle in the drive flow / core flow relationship due to the changing thermal hydraulic operating conditions of the core are accounted for in the margins included in the bases or analyses used to establish the setpoints for the APRM Simulated Thermal Power-High Function and the OPRM Trip Function.

The Frequency of 184 days for SR 3.3.1.1.11, 92 days for SR 3.3.1.1.12 and 24 months for SR 3.3.1.1.13 and SR 3.3.1.1.18 is based upon the assumptions in the determination of the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.12

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. For the APRM Functions, this test supplements the automatic self-test functions that operate continuously in the APRM and voter channels. The scope of the APRM CHANNEL FUNCTIONAL TEST is that which is necessary to test the hardware. Software controlled functions are tested as part of the initial verification and validation and are only incidentally tested as part of the surveillance testing. Automatic self-test functions check the EPROMs in which the software-controlled logic is defined. Changes in the EPROMs will be detected by the self-test function and alarmed via the APRM trouble alarm. SR 3.3.1.1.1 for the APRM functions includes a step to confirm that the automatic self-test function is still operating.

The APRM CHANNEL FUNCTIONAL TEST covers the APRM channels (including recirculation flow processing -- applicable to Function 2.b and the auto-enable portion of Function 2.f only), the 2-out-of-4 Voter channels, and the interface connections into the RPS trip systems from the voter channels.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The 184-day Frequency of SR 3.3.1.1.12 is based on the reliability analyses of References 15 and 16. (NOTE: The actual voting logic of the 2-out-of-4 Voter Function is tested as part of SR 3.3.1.1.15. The auto-enable setpoints for the OPRM Trip are confirmed by SR 3.3.1.1.19.)

A Note is provided for Function 2.a that requires this SR to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM Function cannot be performed in MODE 1 without utilizing jumpers or lifted leads. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2.

A second Note is provided for Functions 2.b and 2.f that clarifies that the CHANNEL FUNCTIONAL TEST for Functions 2.b and 2.f includes testing of the recirculation flow processing electronics, excluding the flow transmitters.

SR 3.3.1.1.15

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The functional testing of control rods (LCO 3.1.3), and SDV vent

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.15 (continued)

and drain valves (LCO 3.1.8), overlaps this Surveillance to provide complete testing of the assumed safety function.

The LOGIC SYSTEM FUNCTIONAL TEST for APRM Function 2.e simulates APRM and OPRM trip conditions at the 2-out-of-4 Voter channel inputs to check all combinations of two tripped inputs to the 2-out-of-4 logic in the voter channels and APRM-related redundant RPS relays.

The 24 month Frequency is based on the need to perform portions of this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.

SR 3.3.1.1.16

This SR ensures that scrams initiated from the Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ 30% RTP. This is performed by a Functional check that ensures the scram feature is not bypassed at ≥ 30% RTP. Because main turbine bypass flow can affect this function nonconservatively (THERMAL POWER is derived from turbine first stage pressure), the opening of the main turbine bypass valves must not cause the trip Function to be bypassed when Thermal Power is ≥ 30% RTP.

If any bypass channel's trip function is nonconservative (i.e., the Functions are bypassed at ≥ 30% RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition, this SR is met and the channel is considered OPERABLE.

The Frequency of 24 months is based on engineering judgment and reliability of the components.

SR 3.3.1.1.17

This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. This test may be performed in one

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.17 (continued)

measurement or in overlapping segments, with verification that all components are tested. The RPS RESPONSE TIME acceptance criteria are included in Reference 11.

RPS RESPONSE TIME for the APRM 2-out-of-4 Voter Function (2.e) includes the APRM Flux Trip output relays and the OPRM Trip output relays of the voter and the associated RPS relays and contactors. (Note: The digital portion of the APRM, OPRM and 2-out-of-4 Voter channels are excluded from RPS RESPONSE TIME testing because self-testing and calibration checks the time base of the digital electronics. Confirmation of the time base is adequate to assure required response times are met. Neutron detectors are excluded from RPS RESPONSE TIME testing because the principles of detector operation virtually ensure an instantaneous response time. See References 12 and 13).

RPS RESPONSE TIME tests are conducted on an 24 month STAGGERED TEST BASIS. Note 3 requires STAGGERED TEST BASIS Frequency to be determined based on 4 channels per trip system, in lieu of the 8 channels specified in Table 3.3.1.1-1 for the MSIV Closure Function because channels are arranged in pairs.

This Frequency is based on the logic interrelationships of the various channels required to produce an RPS scram signal. The 24 month Frequency is consistent with the typical industry refueling cycle and is based upon plant operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

SR 3.3.1.1.17 for Function 2.e confirms the response time of that function, and also confirms the response time of components to Function 2.e and other RPS functions. (Reference 14)

Note 3 allows the STAGGERED TEST BASIS Frequency for Function 2.e to be determined based on 8 channels rather than the 4 actual 2-out-of-4 Voter channels. The redundant outputs from the 2-out-of-4 Voter channel (2 for APRM trips and 2 for OPRM trips) are considered part of the same channel, but the OPRM and APRM outputs are considered to be separate channels for application of SR 3.3.1.1.17, so N = 8. The note further requires that testing of OPRM and APRM outputs from a 2-out-of-4 Voter be alternated. In addition to these commitments, References 15 and 16 require that the testing of inputs to each RPS Trip System alternate.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.17 (continued)

Combining these frequency requirements, an acceptable test sequence is one that:

a. Tests each RPS Trip System interface every other cycle,

b. Alternates the testing of APRM and OPRM outputs from any specific 2-out-of-4 Voter Channel

c. Alternates between divisions at least every other test cycle.

The testing sequence shown in the table below is one sequence that satisfies these requirements.

Function 2.e Testing Sequence for SR 3.3.1.1.17

24-Month Cycle	Voter Output Tested	"Staggering"					
		Voter A1 Output	Voter A2 Output	Voter B1 Output	Voter B2 Output	RPS Trip System	Division
1st	OPRM A1	OPRM				A	1
2nd	APRM B1			APRM		B	1
3rd	OPRM A2		OPRM			A	2
4th	APRM B2				APRM	B	2
5th	APRM A1	APRM				A	1
6th	OPRM B1			OPRM		B	1
7th	APRM A2		APRM			A	2
8th	OPRM B2				OPRM	B	2

After 8 cycles, the sequence repeats.

Each test of an OPRM or APRM output tests each of the redundant outputs from the 2-out-of-4 Voter channel for that Function and each of the corresponding relays in the RPS. Consequently, each of the RPS relays is tested every fourth cycle. The RPS relay testing frequency is twice the frequency justified by References 15 and 16.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.19

This surveillance involves confirming the OPRM Trip auto-enable setpoints. The auto-enable setpoint values are considered to be nominal values as discussed in Reference 21. This surveillance ensures that the OPRM Trip is enabled (not bypassed) for the correct values of APRM Simulated Thermal Power and recirculation drive flow. Other surveillances ensure that the APRM Simulated Thermal Power and recirculation drive flow properly correlate with THERMAL POWER (SR 3.3.1.1.2) and core flow (SR 3.3.1.1.20), respectively.

If any auto-enable setpoint is nonconservative (i.e., the OPRM Trip is bypassed when APRM Simulated Thermal Power ≥ 30% and recirculation drive flow ≤ value equivalent to the core flow value defined in the COLR, then the affected channel is considered inoperable for the OPRM Trip Function. Alternatively, the OPRM Trip auto-enable setpoint(s) may be adjusted to place the channel in a conservative condition (not bypassed). If the OPRM Trip is placed in the not-bypassed condition, this SR is met, and the channel is considered OPERABLE.

For purposes of this surveillance, consistent with Reference 21, the conversion from core flow values defined in the COLR to drive flow values used for this SR can be conservatively determined by a linear scaling assuming that 100% drive flow corresponds to 100 Mlb/hr core flow, with no adjustment made for expected deviations between core flow and drive flow below 100%.

The Frequency of 24 months is based on engineering judgment and reliability of the components.

SR 3.3.1.1.20

The APRM Simulated Thermal Power-High Function (Function 2.b) uses drive flow to vary the trip setpoint. The OPRM Trip Function (Function 2.f) uses drive flow to automatically enable or bypass the OPRM Trip output to RPS. Both of these Functions use drive flow as a representation of reactor core flow. SR 3.3.1.1.18 ensures that the drive flow transmitters and processing electronics are calibrated. This SR adjusts the recirculation drive flow scaling factors in each APRM channel to provide the appropriate drive flow/core flow alignment.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.20

The Frequency of 24 months considers that any change in the core flow to drive flow functional relationship during power operation would be gradual and the maintenance of the Recirculation System and core components that may impact the relationship is expected to be performed during refueling outages. This frequency also considers the period after reaching plant equilibrium conditions necessary to perform the test, engineering judgment of the time required to collect and analyze the necessary flow data, and engineering judgment of the time required to enter and check the applicable scaling factors in each of the APRM channels. This timeframe is acceptable based on the relatively small alignment errors expected, and the margins already included in the APRM Simulated Thermal Power – High and OPRM Trip Function trip – enable setpoints.

REFERENCES

1. FSAR, Figure 7.2-1.
2. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).
3. NEDO-23842, "Continuous Control Rod Withdrawal in the Startup Range," April 18, 1978.
4. FSAR, Section 5.2.2.
5. FSAR, Chapter 15.
6. FSAR, Section 6.3.3.

(continued)

BASES

REFERENCES (continued)

7. Not used.

8. P. Check (NRC) letter to G. Lainas (NRC), "BWR Scram Discharge System Safety Evaluation," December 1, 1980.

9. NEDO-30851-P-A , "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988.

10. NRC Inspection and Enforcement Manual, Part 9900: Technical Guidance, Standard Technical Specification 1.0 Definitions, Issue date 12/08/86.

11. FSAR, Table 7.3-28.

12. NEDO-32291A "System Analyses for Elimination of Selected Response Time Testing Requirements," October 1995.

13. NRC Safety Evaluation Report related to Amendment No. 171 for License No. NPF 14 and Amendment No. 144 for License No. NPF 22.

14. NEDO-32291-A Supplement 1 "System Analyses for the Elimination of Selected Response Time Testing Requirements," October 1999.

15. NEDC-32410P-A, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," October 1995.

16. NEDC-32410P-A Supplement 1, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," November 1997.

17. NEDO-31960-A, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," November 1995.

18. NEDO-31960-A, Supplement 1, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," November 1995.

19. NEDO-32465-A, "BWR Owners' Group Long-Term Stability Detect and Suppress Solutions Licensing Basis Methodology and Reload Applications," August 1996.

BASES

REFERENCES (continued)

20. BWROG Letter BWROG 9479, L. A. England (BWROG) to M. J. Virgilio, "BWR Owners' Group Guidelines for Stability Interim Corrective Action," June 6, 1994.

21. BWROG Letter BWROG 96113, K. P. Donovan (BWROG) to L. E. Phillips (NRC), "Guidelines for Stability Option III 'Enable Region' (TAC M92882)," September 17, 1996.

22. EMF-CC-074(P)(A), Volume 4, "BWR Stability Analysis: Assessment of STAIF with Input from MICROBURN-B2."

23. GE Letter to PPL, GE-2005-EMC426, "Susquehanna 1 & 2 Minimum LPRM Input Requirement for NUMAC APRM 4-Channel Design," April 26, 2005.

Table B 3.3.1.1-1 (page 1 of 1)
RPS Instrumentation Sensor Diversity

Initiation Events	Scram Sensors for Initiating Events						
	RPV Variables			Anticipatory			Fuel
	(a)	(b)	(c)	(d)	(e)	(f)	(g)
MSIV Closure	X		X			X	X
Turbine Trip (w/bypass)	X			X	X		X
Generator Trip (w/bypass)	X			X			X
Pressure Regulator Failure (primary pressure decrease) (MSIV closure trip)	X	X	X			X	X
Pressure Regulator Failure (primary pressure decrease) (Level 8 trip)	X				X		X
Pressure Regulator Failure (primary pressure increase)	X						X
Feedwater Controller Failure (high reactor water level)	X	X			X		X
Feedwater Controller Failure (low reactor water level)	X		X			X	
Loss of Condenser Vacuum	X				X	X	X
Loss of AC Power (loss of transformer)	X		X		X	X	
Loss of AC Power (loss of grid connections)	X		X	X	X	X	X

(a) Reactor Vessel Steam Dome Pressure—High
(b) Reactor Vessel Water Level—High, Level 8
(c) Reactor Vessel Water Level—Low, Level 3
(d) Turbine Control Valve Fast Closure
(e) Turbine Stop Valve—Closure
(f) Main Steam Isolation Valve—Closure
(g) Average Power Range Monitor Neutron Flux—High

B 3.3 INSTRUMENTATION

B 3.3.1.2 Source Range Monitor (SRM) Instrumentation

BASES

BACKGROUND

The SRMs provide the operator with information relative to the neutron flux level at startup and low flux levels in the core. As such, the SRM indication is used by the operator to monitor the approach to criticality and determine when criticality is achieved. The SRMs are not fully withdrawn from the core until the SRM to intermediate range monitor (IRM) overlap is demonstrated (as required by SR 3.3.1.1.6), when the SRMs are normally fully withdrawn from the core.

The SRM subsystem of the Neutron Monitoring System (NMS) consists of four channels. Each of the SRM channels can be bypassed, but only one at any given time, by the operation of a bypass switch. Each channel includes one detector that can be physically positioned in the core. Each detector assembly consists of a miniature fission chamber with associated cabling, signal conditioning equipment, and electronics associated with the various SRM functions. The signal conditioning equipment converts the current pulses from the fission chamber to analog DC currents that correspond to the count rate. Each channel also includes indication, alarm, and control rod blocks. However, this LCO specifies OPERABILITY requirements only for the monitoring and indication functions of the SRMs.

During refueling, shutdown, and low power operations, the primary indication of neutron flux levels is provided by the SRMs or special movable detectors connected to the normal SRM circuits. The SRMs provide monitoring of reactivity changes during fuel or control rod movement and give the control room operator early indication of unexpected subcritical multiplication that could be indicative of an approach to criticality.

APPLICABLE SAFETY ANALYSES

Prevention and mitigation of prompt reactivity excursions during refueling and low power operation is provided by LCO 3.9.1, "Refueling Equipment Interlocks"; LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"; LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"; IRM Neutron Flux—High and Average Power Range Monitor (APRM) Neutron Flux—High

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

(Setdown) Functions; and LCO 3.3.2.1, "Control Rod Block Instrumentation."

The SRMs have no safety function and are not assumed to function during any FSAR design basis accident or transient analysis. However, the SRMs provide the only on-scale monitoring of neutron flux levels during startup and refueling. Therefore, they are being retained in Technical Specifications.

LCO

During startup in MODE 2, three of the four SRM channels are required to be OPERABLE to monitor the reactor flux level prior to and during control rod withdrawal, subcritical multiplication and reactor criticality, and neutron flux level and reactor period until the flux level is sufficient to maintain the IRMs on Range 3 or above. All but one of the channels are required in order to provide a representation of the overall core response during those periods when reactivity changes are occurring throughout the core.

In MODES 3 and 4, with the reactor shut down, two SRM channels provide redundant monitoring of flux levels in the core.

In MODE 5, during a spiral offload or reload, an SRM outside the fueled region will no longer be required to be OPERABLE, since it is not capable of monitoring neutron flux in the fueled region of the core. Fueled region is a continuous area with fuel. Thus, CORE ALTERATIONS are allowed in a quadrant with no OPERABLE SRM in an adjacent quadrant provided the Table 3.3.1.2-1, footnote (b), requirement that the bundles being spiral reloaded or spiral offloaded are all in a single fueled region containing at least one OPERABLE SRM is met. Spiral reloading and offloading encompass reloading or offloading a cell on the edge of a continuous fueled region (the cell can be reloaded or offloaded in any sequence).

In nonspiral routine operations, two SRMs are required to be OPERABLE to provide redundant monitoring of reactivity changes occurring in the reactor core. Because of the local nature of reactivity changes during refueling, adequate coverage is provided by requiring one SRM to be OPERABLE in

(continued)

BASES

LCO (continued)

the quadrant of the reactor core where CORE ALTERATIONS are being performed, and the other SRM to be OPERABLE in an adjacent quadrant containing fuel. These requirements ensure that the reactivity of the core will be continuously monitored during CORE ALTERATIONS.

Special movable detectors, according to footnote (c) of Table 3.3.1.2-1, may be used during CORE ALTERATIONS in place of the normal SRM nuclear detectors. These special detectors must be connected to the normal SRM circuits in the NMS, such that the applicable neutron flux indication can be generated. These special detectors provide more flexibility in monitoring reactivity changes during fuel loading, since they can be positioned anywhere within the core during refueling. They must still meet the location requirements of SR 3.3.1.2.2 and all other required SRs for SRMs.

For an SRM channel to be considered OPERABLE, it must be providing neutron flux monitoring indication.

APPLICABILITY

The SRMs are required to be OPERABLE in MODES 2, 3, 4, and 5 prior to the IRMs being on scale on Range 3 to provide for neutron monitoring. In MODE 1, the APRMs provide adequate monitoring of reactivity changes in the core; therefore, the SRMs are not required. In MODE 2, with IRMs on Range 3 or above, the IRMs provide adequate monitoring and the SRMs are not required.

ACTIONS

A.1 and B.1

In MODE 2, with the IRMs on Range 2 or below, SRMs provide the means of monitoring core reactivity and criticality. With any number of the required SRMs inoperable, the ability to monitor neutron flux is degraded. Therefore, a limited time is allowed to restore the inoperable channels to OPERABLE status.

Provided at least one SRM remains OPERABLE, Required Action A.1 allows 4 hours to restore the required SRMs to OPERABLE status. This time is reasonable because there is adequate capability remaining to monitor the core, there is

(continued)

BASES

ACTIONS

A.1 and B.1 (continued)

limited risk of an event during this time, and there is sufficient time to take corrective actions to restore the required SRMs to OPERABLE status or to establish alternate IRM monitoring capability. During this time, control rod withdrawal and power increase is not precluded by this Required Action. Having the ability to monitor the core with at least one SRM, proceeding to IRM Range 3 or greater (with overlap required by SR 3.3.1.1.6), and thereby exiting the Applicability of this LCO, is acceptable for ensuring adequate core monitoring and allowing continued operation.

With three required SRMs inoperable, Required Action B.1 allows no positive changes in reactivity (control rod withdrawal must be immediately suspended) due to inability to monitor the changes. Required Action A.1 still applies and allows 4 hours to restore monitoring capability prior to requiring control rod insertion. This allowance is based on the limited risk of an event during this time, provided that no control rod withdrawals are allowed, and the desire to concentrate efforts on repair, rather than to immediately shut down, with no SRMs OPERABLE.

C.1

In MODE 2, if the required number of SRMs is not restored to OPERABLE status within the allowed Completion Time, the reactor shall be placed in MODE 3. With all control rods fully inserted, the core is in its least reactive state with the most margin to criticality. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

D.1 and D.2

With one or more required SRMs inoperable in MODE 3 or 4, the neutron flux monitoring capability is degraded or nonexistent. The requirement to fully insert all insertable control rods ensures that the reactor will be at its minimum reactivity level while no neutron monitoring capability is available. Placing the reactor mode switch in the shutdown position prevents subsequent control rod withdrawal by

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

maintaining a control rod block. The allowed Completion Time of 1 hour is sufficient to accomplish the Required Action, and takes into account the low probability of an event requiring the SRM occurring during this interval.

E.1 and E.2

With one or more required SRM inoperable in MODE 5, the ability to detect local reactivity changes in the core during refueling is degraded. CORE ALTERATIONS must be immediately suspended and action must be immediately initiated to insert all insertable control rods in core cells containing one or more fuel assemblies. Suspending CORE ALTERATIONS prevents the two most probable causes of reactivity changes, fuel loading and control rod withdrawal, from occurring. Inserting all insertable control rods ensures that the reactor will be at its minimum reactivity given that fuel is present in the core. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe, conservative position.

Action (once required to be initiated) to insert control rods must continue until all insertable rods in core cells containing one or more fuel assemblies are inserted.

SURVEILLANCE REQUIREMENTS

The SRs for each SRM Applicable MODE or other specified conditions are found in the SRs column of Table 3.3.1.2-1.

SR 3.3.1.2.1 and SR 3.3.1.2.3

Performance of the CHANNEL CHECK 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 another channel. 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 something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2.1 and SR 3.3.1.2.3 (continued)

is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria which are determined by the plant staff based on an investigation of a combination of the channel instrument uncertainties, may be used to support this parameter comparison and include indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit, and does not necessarily indicate the channel is Inoperable.

The Frequency of once every 12 hours for SR 3.3.1.2.1 is based on operating experience that demonstrates channel failure is rare. While in MODES 3 and 4, reactivity changes are not expected; therefore, the 12 hour Frequency is relaxed to 24 hours for SR 3.3.1.2.3. The CHANNEL CHECK supplements less formal checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.2.2

To provide adequate coverage of potential reactivity changes in the core, a maximum of two SRMs are required to be OPERABLE. One SRM is required to be OPERABLE in the quadrant where CORE ALTERATIONS are being performed, and the other OPERABLE SRM must be in an adjacent quadrant containing fuel. However, in accordance with Table 3.3.1.2-1, only one SRM is required during a spiral reload until the fueled region is large enough to encompass a second installed SRM. Note 1 states that the SR is required to be met only during CORE ALTERATIONS. It is not required to be met at other times in MODE 5 since core reactivity changes are not occurring. This Surveillance consists of a review of plant logs to ensure that SRMs required to be OPERABLE for given CORE ALTERATIONS are, in fact, OPERABLE. In the event that only one SRM is required to be OPERABLE, per Table 3.3.1.2-1, footnote (b), only the a. portion of this SR is required. Note 2 clarifies that more than one of the three requirements can be met by the same OPERABLE SRM. The 12 hour Frequency is based upon operating experience and supplements operational controls over refueling activities

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2.2 (continued)

that include steps to ensure that the SRMs required by the LCO are in the proper quadrant.

SR 3.3.1.2.4

This Surveillance consists of a verification of the SRM instrument readout to ensure that the SRM reading is greater than a specified minimum count rate, which ensures that the detectors are indicating count rates indicative of neutron flux levels within the core. The signal-to-noise ratio shown in Figure 3.3.1.2-1 is the SRM count rate at which there is a 95% probability that the SRM signal indicates the presence of neutrons and only a 5% probability that the SRM signal is a result of noise (Ref. 1). With few fuel assemblies loaded, the SRMs will not have a high enough count rate to satisfy the SR. Therefore, allowances are made for loading sufficient "source" material, in the form of irradiated fuel assemblies, to establish the minimum count rate.

To accomplish this, the SR is modified by a Note that states that the count rate is not required to be met on an SRM that has less than or equal to four fuel assemblies adjacent to the SRM and no other fuel assemblies are in the associated core quadrant. With four or less fuel assemblies loaded around each SRM and no other fuel assemblies in the associated core quadrant, even with a control rod withdrawn, the configuration will not be critical. The signal to noise ratio is only required to be determined every 7 or 31 days per the requirements of SR 3.3.1.2.5 or 3.3.1.2.6.

The Frequency is based upon channel redundancy and other information available in the control room, and ensures that the required channels are frequently monitored while core reactivity changes are occurring. When no reactivity changes are in progress, the Frequency is relaxed from 12 hours to 24 hours.

SR 3.3.1.2.5 and SR 3.3.1.2.6

Performance of a CHANNEL FUNCTIONAL TEST demonstrates the associated channel will function properly. SR 3.3.1.2.5 is

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2.5 and SR 3.3.1.2.6 (continued)

required in MODE 5, and the 7 day Frequency ensures that the channels are OPERABLE while core reactivity changes could be in progress. This Frequency is reasonable, based on operating experience and on other Surveillances (such as a CHANNEL CHECK), that ensure proper functioning between CHANNEL FUNCTIONAL TESTS.

SR 3.3.1.2.6 is required in MODE 2 with IRMs on Range 2 or below, and in MODES 3 and 4. Since core reactivity changes do not normally take place, the Frequency has been extended from 7 days to 31 days. The 31 day Frequency is based on operating experience and on other Surveillances (such as CHANNEL CHECK) that ensure proper functioning between CHANNEL FUNCTIONAL TESTS.

Verification of the signal to noise ratio also ensures that the detectors are inserted to an acceptable operating level. In a fully withdrawn condition, the detectors are sufficiently removed from the fueled region of the core to essentially eliminate neutrons from reaching the detector. Any count rate obtained while the detectors are fully withdrawn is assumed to be "noise" only.

The Note to the Surveillance allows the Surveillance to be delayed until entry into the specified condition of the Applicability (THERMAL POWER decreased to IRM Range 2 or below). The SR must be performed within 12 hours after IRMs are on Range 2 or below. The allowance to enter the Applicability with the 31 day Frequency not met is reasonable, based on the limited time of 12 hours allowed after entering the Applicability and the inability to perform the Surveillance while at higher power levels. Although the Surveillance could be performed while on IRM Range 3, the plant would not be expected to maintain steady state operation at this power level. In this event, the 12 hour Frequency is reasonable, based on the SRMs being otherwise verified to be OPERABLE (i.e., satisfactorily performing the CHANNEL CHECK) and the time required to perform the Surveillances.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.2.7

Performance of a CHANNEL CALIBRATION at a Frequency of 24 months verifies the performance of the SRM detectors and associated circuitry. The Frequency considers the plant conditions required to perform the test, the ease of performing the test, and the likelihood of a change in the system or component status. The neutron detectors are excluded from the CHANNEL CALIBRATION because they cannot readily be adjusted. The detectors are fission chambers that are designed to have a relatively constant sensitivity over the range and with an accuracy specified for a fixed useful life.

Note 2 to the Surveillance allows the Surveillance to be delayed until entry into the specified condition of the Applicability. The SR must be performed in MODE 2 within 12 hours of entering MODE 2 with IRMs on Range 2 or below. The allowance to enter the Applicability with the 24 month Frequency not met is reasonable, based on the limited time of 12 hours allowed after entering the Applicability and the inability to perform the Surveillance while at higher power levels. Although the Surveillance could be performed while on IRM Range 3, the plant would not be expected to maintain steady state operation at this power level. In this event, the 12 hour Frequency is reasonable, based on the SRMs being otherwise verified to be OPERABLE (i.e., satisfactorily performing the CHANNEL CHECK) and the time required to perform the Surveillances.

REFERENCES

1. General Electric Service Information Letter (SIL) 478 "SRM Minimum Count Rate" dated December 16, 1988.

(continued)

B 3.3 INSTRUMENTATION

B 3.3.2.1 Control Rod Block Instrumentation

BASES

BACKGROUND

Control rods provide the primary means for control of reactivity changes. Control rod block instrumentation includes channel sensors, logic circuitry, switches, and relays that are designed to ensure that specified fuel design limits are not exceeded for postulated transients and accidents. During high power operation, the rod block monitor (RBM) provides protection for control rod withdrawal error events. During low power operations, control rod blocks from the rod worth minimizer (RWM) enforce specific control rod sequences designed to mitigate the consequences of the control rod drop accident (CRDA). During shutdown conditions, control rod blocks from the Reactor Mode Switch—Shutdown Position Function ensure that all control rods remain inserted to prevent inadvertent criticalities.

The purpose of the RBM is to limit control rod withdrawal if localized neutron flux exceeds a predetermined setpoint during control rod manipulations. The RBM supplies a trip signal to the Reactor Manual Control System (RMCS) to appropriately inhibit control rod withdrawal during power operation above the low power range setpoint. The RBM has two channels, either of which can initiate a control rod block when the channel output exceeds the control rod block setpoint. One RBM channel inputs into one RMCS rod block circuit and the other RBM channel inputs into the second RMCS rod block circuit. The RBM channel signal is generated by averaging a set of local power range monitor (LPRM) signals at various core heights surrounding the control rod being withdrawn. An APRM flux signal from one of the four redundant average power range monitor (APRM) channels supplies a reference signal for one of the RBM channels and an APRM flux signal from another of the APRM channels supplies the reference signal to the second RBM channel. This reference signal is used to enable the RBM. If the APRM is indicating less than the low power range setpoint, the RBM is automatically bypassed. The RBM is also automatically bypassed if a peripheral control rod is selected (Ref. 2).

The purpose of the RWM is to control rod patterns during startup, such that only specified control rod sequences and relative positions are allowed over the operating range from all control rods inserted to 10% RTP. The sequences effectively limit the potential amount and rate of reactivity increase during a CRDA. Prescribed control rod sequences are stored in the RWM, which will initiate control rod withdrawal and insert blocks when the actual sequence deviates beyond allowances from the stored sequence. The RWM determines the actual sequence

(continued)

BASES

BACKGROUND (continued)

based position indication for each control rod. The RWM also uses steam flow signals to determine when the reactor power is above the preset power level at which the RWM is automatically bypassed (Ref. 1). The RWM is a single channel system that provides input into RMCS rod block channel 2.

The function of the individual rod sequence steps (banking steps) is to minimize the potential reactivity increase from postulated CRDA at low power levels. However, if the possibility for a control rod to drop can be eliminated, then banking steps at low power levels are not needed to ensure the applicable event limits can not be exceeded. The rods may be inserted without the need to stop at intermediate positions since the possibility of a CRDA is eliminated by the confirmation that withdrawn control rods are coupled.

To eliminate the possibility of a CRDA, administrative controls require that any partially inserted control rods, which have not been confirmed to be coupled since their last withdrawal, be fully inserted prior to reaching the THERMAL POWER of ≤10% RTP. If a control rod has been checked for coupling at notch 48 and the rod has not been moved inward, this rod is in contact with it's drive and is not required to be fully inserted prior to reaching the THERMAL POWER of ≤10% RTP. However, if it cannot be confirmed that the control rod has been moved inward, then that rod shall be fully inserted prior to reaching the THERMAL POWER of ≤10% RTP. The remaining control rods may then be inserted without the need to stop at intermediate positions since the possibility of a CRDA has been eliminated.

With the reactor mode switch in the shutdown position, a control rod withdrawal block is applied to all control rods to ensure that the shutdown condition is maintained. This Function prevents inadvertent criticality as the result of a control rod withdrawal during MODE 3 or 4, or during MODE 5 when the reactor mode switch is required to be in the shutdown position. The reactor mode switch has two channels, each inputting into a separate RMCS rod block circuit. A rod block in either RMCS circuit will provide a control rod block to all control rods.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor

The RBM is designed to limit control rod withdrawal if localized neutron flux exceeds a predetermined setpoint. The RBM was originally designed to

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

prevent fuel damage during a Rod Withdrawal Error (RWE) event while operating in the power range in a normal mode of operation. FSAR 15.4.2 (Ref. 10) (Rod Withdrawal Error - At Power) originally took credit for the RBM automatically actuating to stop control rod motion and preventing fuel damage during an RWE event at power. However, current reload analyses do not take credit for the RBM system. The Allowable Values are chosen as a function of power level to not exceed the APRM scram setpoints.

The RBM function satisfies Criterion 4 of the NRC Policy Statement (Ref. 7).

Two channels of the RBM are required to be OPERABLE, with their setpoints within the appropriate Allowable Value for the associated power range, to ensure that no single instrument failure can preclude a rod block for this Function. The actual setpoints are calibrated consistent with applicable setpoint methodology.

Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Values between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor power), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The RBM will function when operating greater than 30% RTP. Below this power level, the RBM is not required to be OPERABLE.

2. Rod Worth Minimizer

The RWM enforces the banked position withdrawal sequence (BPWS) to ensure that the initial conditions of the CRDA analysis are not violated.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The analytical methods and assumptions used in evaluating the CRDA are summarized in References 2, 3, 4, and 5. The BPWS requires that control rods be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions. Requirements that the control rod sequence is in compliance with the BPWS are specified in LCO 3.1.6, "Rod Pattern Control."

When performing a shutdown of the plant, an optional BPWS control rod sequence (Ref. 7) may be used if the coupling of each withdrawn control rod has been confirmed. The rods may be inserted without the need to stop at intermediate positions. When using the Reference 11 control rod insertion sequence for shutdown, the rod worth minimizer may be reprogrammed to enforce the requirements of the improved BPWS control rod insertion, or may be bypassed and the improved BPWS shutdown sequence implemented under the controls in Condition D.

The RWM Function satisfies Criterion 3 of the NRC Policy Statement. (Ref. 7)

Since the RWM is designed to act as a backup to operator control of the rod sequences, only one channel of the RWM is available and required to be OPERABLE (Ref. 6). Special circumstances provided for in the Required Action of LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.6 may necessitate bypassing the RWM to allow continued operation with inoperable control rods, or to allow correction of a control rod pattern not in compliance with the BPWS. The RWM may be bypassed as required by these conditions, but then it must be considered inoperable and the Required Actions of this LCO followed.

Compliance with the BPWS, and therefore OPERABILITY of the RWM, is required in MODES 1 and 2 when THERMAL POWER is < 10% RTP. When THERMAL POWER is > 10% RTP, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA (Refs. 4 and 6). In MODES 3 and 4, all control rods are required to be inserted into the core (except as provided in 3.10 "Special Operations"); therefore, a CRDA cannot occur. In MODE 5, since only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will be subcritical.

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY (continued)

3. Reactor Mode Switch—Shutdown Position

During MODES 3 and 4, and during MODE 5 when the reactor mode switch is required to be in the shutdown position, the core is assumed to be subcritical; therefore, no positive reactivity insertion events are analyzed. The Reactor Mode Switch—Shutdown Position control rod withdrawal block ensures that the reactor remains subcritical by blocking control rod withdrawal, thereby preserving the assumptions of the safety analysis.

The Reactor Mode Switch—Shutdown Position Function satisfies Criterion 3 of the NRC Policy Statement. (Ref. 7)

Two channels are required to be OPERABLE to ensure that no single channel failure, will preclude a rod block when required. There is no Allowable Value for this Function since the channels are mechanically actuated based solely on reactor mode switch position.

During shutdown conditions (MODE 3, 4, or 5), no positive reactivity insertion events are analyzed because assumptions are that control rod withdrawal blocks are provided to prevent criticality. Therefore, when the reactor mode switch is in the shutdown position, the control rod withdrawal block is required to be OPERABLE. During MODE 5 with the reactor mode switch in the refueling position, the refuel position one-rod-out interlock (LCO 3.9.2) provides the required control rod withdrawal blocks.

ACTIONS

A.1

With one RBM channel inoperable, the remaining OPERABLE channel is adequate to perform the control rod block function; however, overall reliability is reduced because a single failure in the remaining OPERABLE channel can result in no control rod block capability for the RBM. For this reason, Required Action A.1 requires restoration of the inoperable channel to OPERABLE status. The Completion Time of 5 days is based on the low probability of an event occurring coincident with a failure in the remaining OPERABLE channel.

(continued)

BASES

ACTIONS (continued)

B.1

If Required Action A.1 is not met and the associated Completion Time has expired, the inoperable channel must be placed in trip within 48 hours. If both RBM channels are inoperable, the RBM is not capable of performing its intended function; thus, one channel must also be placed in trip. This initiates a control rod withdrawal block, thereby ensuring that the RBM function is met.

The 48 hour Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities and is acceptable because it minimizes risk while allowing time for restoration or tripping of inoperable channels.

C.1, C.2.1.1, C.2.1.2, and C.2.2

With the RWM inoperable during a reactor startup, the operator is still capable of enforcing the prescribed control rod sequence. However, the overall reliability is reduced because a single operator error can result in violating the control rod sequence. Therefore, control rod movement must be immediately suspended except by scram. Alternatively, startup may continue if at least 12 control rods have already been withdrawn, or a reactor startup with an inoperable RWM was not performed in the last calendar year, i.e., the last 12 months. Required Actions C.2.1.1 and C.2.1.2 require verification of these conditions by review of plant logs and control room indications. A reactor startup with an inoperable RWM is defined as rod withdrawal during startup when the RWM is required to be OPERABLE. Once Required Action C.2.1.1 or C.2.1.2 is satisfactorily completed, control rod withdrawal may proceed in accordance with the restrictions imposed by Required Action C.2.2. Required Action C.2.2 allows for the RWM Function to be performed manually and requires a double check of compliance with the prescribed rod sequence by a second licensed operator (Reactor Operator or Senior Reactor Operator) or other qualified member of the technical staff. The RWM may be bypassed under these conditions to allow continued operations. In addition, Required Actions of LCO 3.1.3 and LCO 3.1.6 may require bypassing the RWM, during which time the RWM must be considered inoperable with Condition C entered and its Required Actions taken.

(continued)

BASES

ACTIONS (continued)

D.1

With the RWM inoperable during a reactor shutdown, the operator is still capable of enforcing the prescribed control rod sequence. Required Action D.1 allows for the RWM Function to be performed manually and requires a double check of compliance with the prescribed rod sequence by a second licensed operator (Reactor Operator or Senior Reactor Operator) or other qualified member of the technical staff. The RWM may be bypassed under these conditions to allow the reactor shutdown to continue.

E.1 and E.2

With one Reactor Mode Switch—Shutdown Position control rod withdrawal block channel inoperable, the remaining OPERABLE channel is adequate to perform the control rod withdrawal block function. However, since the Required Actions are consistent with the normal action of an OPERABLE Reactor Mode Switch—Shutdown Position Function (i.e., maintaining all control rods inserted), there is no distinction between having one or two channels inoperable.

In both cases (one or both channels inoperable), suspending all control rod withdrawal and initiating action to fully insert all insertable control rods in core cells containing one or more fuel assemblies will ensure that the core is subcritical with adequate SDM ensured by LCO 3.1.1. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and are therefore not required to be inserted. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies are fully inserted.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each Control Rod Block instrumentation Function are found in the SRs column of Table 3.3.2.1-1.

The Surveillances are modified by a Note to indicate that when an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Refs. 9, 12 and 13).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that a control rod block will be initiated when necessary.

SR 3.3.2.1.1

A CHANNEL FUNCTIONAL TEST is performed for each RBM channel to ensure that the entire channel will perform the intended function. It includes the Reactor Manual Control Multiplexing System input. The Frequency of 184 days is based on reliability analyses (Refs. 8, 12 and 13).

SR 3.3.2.1.2 and SR 3.3.2.1.3

A CHANNEL FUNCTIONAL TEST is performed for the RWM to ensure that the entire system will perform the intended function. The CHANNEL FUNCTIONAL TEST for the RWM is performed by attempting to withdraw a control rod not in compliance with the prescribed sequence and verifying a control rod block occurs and by verifying proper indication of the selection error of at least one out-of-sequence control rod. As noted in the SRs, SR 3.3.2.1.2 is not required to be performed until 1 hour after any control rod is withdrawn in MODE 2. As noted, SR 3.3.2.1.3 is not required to be performed until 1 hour after THERMAL POWER is ≤ 10% RTP in MODE 1. This allows entry into MODE 2 for SR 3.3.2.1.2, and entry into MODE 1 when THERMAL POWER is ≤ 10% RTP for SR 3.3.2.1.3, to perform the required Surveillance if the 92 day Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs. The Frequencies are based on reliability analysis (Ref. 8).

SR 3.3.2.1.4

The RBM trips are automatically bypassed when power is below a specified value and a peripheral control rod is not selected. The power Allowable Value must be verified periodically to not be bypassed when ≥ 30% RTP. This is performed by a Functional check. If any RBM bypass setpoint is non-conservative, then the affected RBM channel is

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1.4 (continued)

considered inoperable. Alternatively, the RBM channel can be placed in the conservative condition (i.e., enabling the RBM trip). If placed in this condition, the SR is met and the RBM channel is not considered inoperable. As noted neutron detectors are excluded from the Surveillance because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal.

Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.8. The 24 month Frequency is based on the need to perform the surveillance during a plant start-up.

SR 3.3.2.1.5

The RWM is automatically bypassed when power is above a specified value. The power level is determined from steam flow signals. The automatic bypass setpoint must be verified periodically to be not bypassed ≤ 10% RTP. This is performed by a Functional check. If the RWM low power setpoint is nonconservative, then the RWM is considered inoperable. Alternately, the low power setpoint channel can be placed in the conservative condition (nonbypass). If placed in the nonbypassed condition, the SR is met and the RWM is not considered inoperable. The Frequency is based on the need to perform the Surveillance during a plant start-up.

SR 3.3.2.1.6

A CHANNEL FUNCTIONAL TEST is performed for the Reactor Mode Switch—Shutdown Position Function to ensure that the entire channel will perform the intended function. The CHANNEL FUNCTIONAL TEST for the Reactor Mode Switch—Shutdown Position Function is performed by attempting to withdraw any control rod with the reactor mode switch in the shutdown position and verifying a control rod block occurs.

As noted in the SR, the Surveillance is not required to be performed until 1 hour after the reactor mode switch is in the shutdown position, since testing of this interlock with the reactor mode switch in any other position cannot be performed without using jumpers, lifted leads, or movable

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1.6 (continued)

links. This allows entry into MODES 3 and 4 if the 24 month Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs.

The 24 month Frequency is based on the need to perform portions of this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency.

SR 3.3.2.1.7

CHANNEL CALIBRATION is a test that verifies the channel responds to the measured parameter with the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibration consistent with the plant specific setpoint methodology.

As noted, neutron detectors are excluded from the CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.8.

The Frequency is based upon the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.2.1.8

The RWM will only enforce the proper control rod sequence if the rod sequence is properly input into the RWM computer. This SR ensures that the proper sequence is loaded into the RWM so that it can perform its intended function. The Surveillance is performed once prior to declaring RWM OPERABLE following loading of sequence into RWM, since this is when rod sequence input errors are possible.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 7.7.1.2.8.

2. FSAR, Section 7.6.1.a.5.7

3. NEDE-24011-P-A-9-US, "General Electrical Standard Application for Reload Fuel," Supplement for United States, Section S 2.2.3.1, September 1988.

4. "Modifications to the Requirements for Control Rod Drop Accident Mitigating Systems," BWR Owners' Group, July 1986.

5. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.

6. NRC SER, "Acceptance of Referencing of Licensing Topical Report NEDE-24011-P-A," "General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17," December 27, 1987.

7. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 32193)

8. NEDC-30851-P-A, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988.

9. GENE-770-06-1, "Addendum to Bases for changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation, Technical Specifications," February 1991.

10. FSAR, Section 15.4.2.

11. NEDO 33091-A, Revision 2, "Improved BPWS Control Rod Insertion Process," April 2003.

12. NEDC-32410P-A, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," October 1995.

13. NEDC-32410P-A Supplement 1, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," November 1997.

B 3.3 INSTRUMENTATION

B 3.3.3.1 Post Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

The primary purpose of the PAM instrumentation is to display plant variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for Design Basis Events. The instruments that monitor these variables are designated as Type A, Category I, and non-Type A, Category I, in accordance with Regulatory Guide 1.97 (Ref. 1).

The OPERABILITY of the accident monitoring instrumentation ensures that there is sufficient information available on selected plant parameters to monitor and assess plant status and behavior following an accident. This capability is consistent with the recommendations of Reference 1.

APPLICABLE SAFETY ANALYSES

The PAM instrumentation LCO ensures the OPERABILITY of Regulatory Guide 1.97, Type A variables so that the control room operating staff can:

- Perform the diagnosis specified in the Emergency Operating Procedures (EOPs). These variables are restricted to preplanned actions for the primary success path of Design Basis Accidents (DBAs), (e.g., loss of coolant accident (LOCA)), and
- Take the specified, preplanned, manually controlled actions for which no automatic control is provided, which are required for safety systems to accomplish their safety function.

The PAM instrumentation LCO also ensures OPERABILITY of Category I, non-Type A, variables so that the control room operating staff can:

- Determine whether systems important to safety are performing their intended functions;

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

- Determine the potential for causing a gross breach of the barriers to radioactivity release;
- Determine whether a gross breach of a barrier has occurred; and
- Initiate action necessary to protect the public and for an estimate of the magnitude of any impending threat.

The plant specific Regulatory Guide 1.97 Analysis (Ref. 2 and 3) documents the process that identified Type A and Category I, non-Type A, variables.

Accident monitoring instrumentation that satisfies the definition of Type A in Regulatory Guide 1.97 meets Criterion 3 of the NRC Policy Statement. (Ref. 4) Category I, non-Type A, instrumentation is retained in Technical Specifications (TS) because they are intended to assist operators in minimizing the consequences of accidents. Therefore, these Category I variables are important for reducing public risk.

LCO

LCO 3.3.3.1 requires two OPERABLE channels for all but one Function to ensure that no single failure prevents the operators from being presented with the information necessary to determine the status of the plant and to bring the plant to, and maintain it in, a safe condition following that accident.

Furthermore, provision of two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information.

The exception to the two channel requirement is primary containment isolation valve (PCIV) position. In this case, the important information is the status of the primary containment penetrations. The LCO requires one position indicator for each active PCIV. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve and prior knowledge of passive valve or via system boundary

(continued)

BASES

LCO (continued)

status. If a normally active PCIV is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE.

The following list is a discussion of the specified instrument Functions listed in Table 3.3.3.1-1 in the accompanying LCO. Table B 3.3.3.1-1 provides a listing of the instruments that are used to meet the operability requirements for the specific functions.

1. Reactor Steam Dome Pressure

Reactor steam dome pressure is a Type A, Category 1, variable provided to support monitoring of Reactor Coolant System (RCS) integrity and to verify operation of the Emergency Core Cooling Systems (ECCS). Two independent pressure channels, consisting of three wide range control room indicators and one wide range control room recorder per channel with a range of 0 psig to 1500 psig, monitor pressure. The wide range recorders are the primary method of indication available for use by the operators during an accident, therefore, the PAM Specification deals specifically with this portion of the instrument channel.

2. Reactor Vessel Water Level

Reactor vessel water level is a Type A, Category 1, variable provided to support monitoring of core cooling and to verify operation of the ECCS. A combination of three different level instrument ranges, with two independent channels each, monitor Reactor Vessel Water Level. The extended range instrumentation measures from -150 inches to 180 inches and outputs to three control room level indicators per channel. The wide range instrumentation measures from -150 inches to 60 inches and outputs to one control room recorder and three control room indicators per channel. The fuel zone range instrumentation measures from -310 inches to -110 inches and outputs to a control room recorder (one channel) and a control room indicator (one channel). These three ranges of instruments combine to provide level indication from the bottom of the Core to above the main steam line. The wide range level recorders, the fuel zone level indicator and level recorder, and one inner ring extended range level indicator per channel are the primary method of indication available for use by the operator during an accident, therefore the PAM

(continued)

BASES

LCO

2. Reactor Vessel Water Level (continued)

Specification deals specifically with this portion of the instrument channel.

3. Suppression Chamber Water Level

Suppression chamber water level is a Type A, Category 1, variable provided to detect a breach in the reactor coolant pressure boundary (RCPB). This variable is also used to verify and provide long term surveillance of ECCS function. A combination of two different level instrument ranges, with two independent channels each, monitor Suppression chamber water level. The wide range instrumentation measures from the ECCS suction lines to approximately the top of the chamber and outputs to one control room recorder per channel. The wide range recorders are the primary method of indication available for use by the operator during an accident, therefore the PAM Specification deals specifically with this portion of the instrument channel.

4. Primary Containment Pressure

Primary Containment pressure is a Type A, Category 1, variable provided to detect a breach of the RCPB and to verify ECCS functions that operate to maintain RCS integrity. A combination of two different pressure instrument ranges, with two independent channels each, monitor primary containment pressure. The LOCA range measures from -15 psig to 65 psig and outputs to one control room recorder per channel. The accident range measures from 0 psig to 250 psig and outputs to one control room recorder per channel (same recorders as the LOCA range). The recorders (both ranges) are the primary method of indication available for use by the operator during an accident, therefore the PAM Specification deals specifically with this portion of the instrument channel.

5. Primary Containment High Radiation

Primary containment area radiation (high range) is provided to monitor the potential of significant radiation releases

(continued)

BASES

LCO

5. Primary Containment High Radiation (continued)

and to provide release assessment for use by operators in determining the need to invoke site emergency plans. Two independent channels, which output to one control room recorder per channel with a range of 10^0 to $1X10^8$ R/hr, monitor radiation. The PAM Specification deals specifically with this portion of the instrument channel.

6. Primary Containment Isolation Valve (PCIV) Position

PCIV position is provided for verification of containment integrity. In the case of PCIV position, the important information is the isolation status of the containment penetration. The LCO requires a channel of valve position indication in the control room to be OPERABLE for an active PCIV in a containment penetration flow path, i.e., two total channels of PCIV position indication for a penetration flow path with two active valves. For containment penetrations with only one active PCIV having control room indication, Note (b) requires a single channel of valve position indication to be OPERABLE. This is sufficient to redundantly verify the isolation status of each isolable penetration via indicated status of the active valve, as applicable, and prior knowledge of passive valve or system boundary status. If a penetration flow path is isolated, position indication for the PCIV(s) in the associated penetration flow path is not needed to determine status. Therefore, the position indication for valves in an isolated penetration flow path is not required to be OPERABLE. These valves which require position indication are specified in Table B 3.6.1.3-1. Furthermore, the loss of position indication does not necessarily result in the PCIV being inoperable.

The PCIV position PAM instrumentation consists of position switches unique to PCIVs, associated wiring and control room indicating lamps (not necessarily unique to a PCIV) for active PCIVs (check valves and manual valves are not required to have position indication). Therefore, the PAM Specification deals specifically with these instrument channels.

(continued)

BASES

LCO (continued)

7. Neutron Flux

Wide range neutron flux is a Category I variable provided to verify reactor shutdown. The Neutron Monitoring System Average Power Range Monitors (APRM) provides reliable neutron flux measurement from 0% to 125% of full power. The APRM consists of four channels each with their own chassis powered with redundant power supplies. The APRM sends signals to the analog isolator module which in turn sends individual APRM signals to the recorders used for post accident monitoring. The PAM function for neutron flux is satisfied by having any 2 channels of APRM provided for post accident monitoring. The PAM Specification deals specifically with this portion of the instrument channel.

The Neutron Monitoring System (NMS) was evaluated against the criteria established in General Electric NEDO-31558A to ensure its acceptability for post-accident monitoring. NEDO-31558A provides alternate criteria for the NMS to meet the post-accident monitoring guidance of Regulatory Guide 1.97. Based on the evaluation, the NMS was found to meet the criteria established in NEDO-31558A. The APRM sub-function of the NMS is used to provide the Neutron Flux monitoring identified in TS 3.3.3.1 (Ref. 5 and 6).

8. Containment Hydrogen and Oxygen Analyzers

The drywell and suppression chamber hydrogen and oxygen concentrations are Type A, Category 1, variables. Two independent gas analyzers monitor hydrogen and oxygen concentration to detect unsafe combustible gas levels in primary containment. The analyzers are capable of determining hydrogen concentration in the range of 0 to 30% by volume and oxygen concentration in the range of 0 to 10% by volume, and each provide control room indication and output to a control room recorder. Each gas analyzer must be capable of sampling either the drywell or the suppression chamber. The recorders are the primary method of indication available for use by the operator during an accident, therefore the PAM Specification deals specifically with this portion of the instrument channel. The gas analyzer piping is provided with heat tracing to reduce the buildup of condensation in the system. H2O2 Analyzers can be considered OPERABLE for accident monitoring (TS 3.3.3.1) for up to 100 days with their heat tracing INOPERABLE.

(continued)

BASES

LCO (continued)

9. Drywell Atmosphere Temperature

Drywell atmosphere temperature is a Category I variable provided to verify RCS and containment integrity and to verify the effectiveness of ECCS actions taken to prevent containment breach. Two independent temperature channels, consisting of two control room recorders per channel with a range of 40 to 440 degrees F, monitor temperature. The PAM Specification deals specifically with the inner ring temperature recorder portion of the instrument channel.

10. Suppression Chamber Water Temperature

Suppression Chamber water temperature is a Type A, Category 1, variable provided to detect a condition that could potentially lead to containment breach and to verify the effectiveness of ECCS actions taken to prevent containment breach. The suppression chamber water temperature instrumentation allows operators to detect trends in suppression chamber water temperature in sufficient time to take action to prevent steam quenching vibrations in the suppression pool. Two channels are required to be OPERABLE. Each channel consists of eight sensors of which a minimum of four sensors (one sensor in each quadrant) must be OPERABLE to consider a channel OPERABLE. The outputs for the temperature sensors are displayed on two independent indicators in the control room and recorded on the monitoring units located in the control room on a back panel. The temperature indicators are the primary method of indication available for use by the operator during an accident, therefore the PAM Specification deals specifically with this portion of the instrument channel.

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1 and 2. These variables are related to the diagnosis and preplanned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in MODES 1 and 2. In MODES 3, 4, and 5, plant conditions are such that the likelihood of an event that would require PAM instrumentation is extremely low; therefore, PAM instrumentation is not required to be OPERABLE in these MODES.

(continued)

BASES (continued)

ACTIONS

A note has been provided to modify the ACTIONS related to PAM instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable PAM instrumentation channels provide appropriate compensatory measures for separate Functions. As such, a Note has been provided that allows separate Condition entry for each inoperable PAM Function.

A.1

When one or more Functions have one required channel that is inoperable, the required inoperable channel must be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channels, the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval.

B.1

If a channel has not been restored to OPERABLE status in 30 days, this Required Action specifies initiation of action in accordance with Specification 5.6.7, which requires a written report to be submitted to the NRC. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative actions.

(continued)

BASES

ACTIONS

B.1 (continued)

This action is appropriate in lieu of a shutdown requirement because alternative actions are identified before the written report is submitted to the NRC, and given the likelihood of plant conditions that would require information provided by this instrumentation.

C.1

When one or more Functions have two required channels that are inoperable (i.e., two channels inoperable in the same Function), one channel in the Function should be restored to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation and the availability of alternate means to obtain the required information. Continuous operation with two required channels inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur.

D.1

This Required Action directs entry into the appropriate Condition referenced in Table 3.3.3.1-1. The applicable Condition referenced in the Table is Function dependent. Each time an inoperable channel has not met any Required Action of Condition C, as applicable, and the associated Completion Time has expired, Condition D is entered for that channel and provides for transfer to the appropriate subsequent Condition.

E.1

For the majority of Functions in Table 3.3.3.1-1, if any Required Action and associated Completion Time of Condition C are not met, the plant must be brought to a MODE in which the LCO not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions

(continued)

BASES

ACTIONS

E.1 (continued)

from full power conditions in an orderly manner and without challenging plant systems.

F.1

Since alternate means of monitoring primary containment area radiation have been developed and tested, the Required Action is not to shut down the plant, but rather to follow the directions of Specification 5.6.7. These alternate means will be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allotted time. The report provided to the NRC should discuss the alternate means used, describe the degree to which the alternate means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels.

SURVEILLANCE REQUIREMENTS

The following SRs apply to each PAM instrumentation Function in Table 3.3.3.1-1.

SR 3.3.3.1.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel against 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 instrument channels could be an indication of excessive instrument drift in one of the channels or 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.

Agreement criteria which are determined by the plant staff based on an investigation of a combination of the channel instrument uncertainties, may be used to support this

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1.1 (continued)

parameter comparison and include indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit and does necessarily indicate the channel is Inoperable.

The Frequency of 31 days is based upon plant operating experience, with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function in any 31 day interval is rare. The CHANNEL CHECK supplements less formal checks of channels during normal operational use of those displays associated with the required channels of this LCO.

SR 3.3.3.1.2 and SR 3.3.3.1.3

A CHANNEL CALIBRATION is performed every 92 days for the containment Hydrogen and Oxygen Analyzers or 24 months for the other Functions except for the PCIV Position Function. The PCIV Position Function is adequately demonstrated by the Remote Position Indication performed in accordance with 5.5.6, "Inservice Testing Program". CHANNEL CALIBRATION verifies that the channel responds to measured parameter with the necessary range and accuracy, and does not include alarms.

The CHANNEL CALIBRATION for the Containment High Radiation instruments shall consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr and a one point calibration check of the detector below 10 R/hr with an installed or portable gamma source.

The CHANNEL CALIBRATION for the hydrogen analyzers, use a sample gas containing: a) Nominal zero volume percent hydrogen, balance nitrogen and b) Nominal thirty volume percent hydrogen, balance nitrogen.

The Frequency is based on operating experience and for the 24 month Frequency consistency with the industry refueling cycles.

(continued)

BASES

REFERENCES

1. Regulatory Guide 1.97 Rev. 2, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," February 6, 1985

2. Nuclear Regulatory Commission Letter A. Schwencer to N. Curtis, Emergency Response Capability, Conformance to R.G. 1.97, Rev. 2, dated February 6, 1985.

3. PP&L Letter (PLA-2222), N. Curtis to A. Schwencer, dated May 31, 1984.

4. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 32193)

5. NEDO-31558A, BWROG Topical Report, Position on NRC Reg. Guide 1.97, Revision 3 Requirements for Post Accident Neutron Monitoring System (NMS).

6. Nuclear Regulatory Commission Letter from C. Poslusny to R.G. Byram dated July 3, 1996.

TABLE B 3.3.3.1-1

Post Accident Instruments
(Page 1 of 3)

Instrument/Variable	Element	Transmitter	Recorder	Indicator
1. Reactor Steam Dome Pressure	N/A	PT-14201A	LP/PR-14201A (blue pen)*	PI-14202A PI-14202A1 PI-14204A
	N/A	PT-14201B	LR/PR-14201B (blue pen)*	PI-14202B (left side) PI-14202B1 (left side) PI-14204B (left side)
2. Reactor Vessel Water Level	N/A	LT-14201A (Wide Range)	LR/PR-14201A (red pen)*	LI-14201A (left side) LI-14201A1 (left side) LI-14203A (left side)
	N/A	LT-14201B (Wide Range)	LR/PR-14201B (red pen)*	LI-14201B (left side) LI-14201B1 (left side) LI-14203B (left side)
	N/A	LT-14203A (Extended Range)	N/A	LI-14201A (right side)*(1) LI-14201A1 (right side) LI-14203A (right side)
	N/A	LT-14203B (Extended Range)	N/A	LI-14201B (right side)*(1) LI-14201B1 (right side) LI-14203B (right side)
	N/A	LT-14202A (Fuel Zone Range)	LR-14202 (red pen)*	N/A
	N/A	LT-14202B (Fuel Zone Range)	N/A	LI-14205B*
3. Suppression Chamber Water Level	N/A	LT-15776A (Wide Range)	LR-15776A (red pen)*	N/A
	N/A	LT-15776B (Wide Range)	LR-15776B (red pen)*	N/A
	N/A	LT-15775A (Narrow Range)	LR-15776A (blue pen)	LI-15775A
	N/A	LT-15775B (Narrow Range)	LR-15776B (blue pen)	LI-15775B

TS-Prop/3.3/SA33031A.B1B

TABLE B 3.3.3.1-1

Post Accident Instruments
(Page 2 of 3)

Instrument/Variable	Element	Transmitter	Recorder	Indicator
4. Primary Containment Pressure	N/A	PT-15709A (0 to 250 psig)	UR-15701A (Dark Blue)*	N/A
	N/A	PT-15709B (0 to 250 psig)	UR-15701B (Dark Blue)*	N/A
	N/A	PT-15710A (-15 to 65 psig)	UR-15701A (Red)*	N/A
	N/A	PT-15710B (-15 to 65 psig)	UR-15701B (Red)*	N/A
5. Primary Containment High Radiation	RE-15720A	RITS-15720A	RR-15720A*	N/A
	RE-15720B	RITS-15720B	RR-15720B*	N/A
6. PCIV Position	See Technical Specification Bases Table B 3.6.1.3-1 for PCIV that require position indication to be OPERABLE			
7. Neutron Flux	N/A	APRM-1	NR-C51-1R603A (red pen)*	N/A
	N/A	APRM-2	NR-C51-1R603B (red pen)*	N/A
	N/A	APRM-3	NR-C51-1R603C (red pen)*	N/A
	N/A	APRM-4	NR-C51-1R603D (red pen)*	N/A
8. Containment Oxygen and Hydrogen Analyzer	AE-15745A (Hydrogen)	AIT-15745A	UR-15701A (Blue Violet)* UR-15701A (Violet)*	N/A
	AE-15745B (Hydrogen)	AIT-15745B	UR-15701B (Blue Violet)* UR-15701B (Violet)*	N/A
	AE-15746A (Oxygen)	AIT-15746A	UR-15701A (Orange)*	N/A
	AE-15746B (Oxygen)	AIT-15746B	UR-15701B (Orange)*	N/A

TABLE B 3.3.3.1-1

Post Accident Instruments
(Page 3 of 3)

Instrument/Variable	Element	Transmitter	Recorder	Indicator
9. Drywell Atmosphere Temperature	TE-15790A	TT-15790A	UR-15701A (Brown)* TR-15790A (point # 1)	N/A
	TE-15790B	TT-15790B	UR-15701B (Brown)* TR-15790B (point # 1)	N/A
10. Suppression Chamber Water Temperature	TE-15753 TE-15755 TE-15757 TE-15759 TE-15763 TE-15765 TE-15767 TE-15769	TX-15751	N/A	TIAH-15751* TI-15751
	TE-15752 TE-15754 TE-15758 TE-15760 TE-15762 TE-15766 TE-15768 TE-15770	TX-15752	N/A	TIAH-15752* TI-15752

* Indicates that the instrument (and associated components in the instrument channel) is considered as instrument channel surveillance acceptance criteria.

(1) In the case of the inner ring indicators for extended range level, it is recommended that LI-14201A and LI-14201B be used as acceptance criteria, however LI-14201A1, LI-14201B1, LI-14203A, or LI-14203B may be used in their place provided that surveillance requirements are satisfied. Only one set of these instruments needs to be OPERABLE.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 Recirculation Loops Operating

BASES

BACKGROUND

The Reactor Coolant Recirculation System is designed to provide a forced coolant flow through the core to remove heat from the fuel. The forced coolant flow removes more heat from the fuel than would be possible with just natural circulation. The forced flow, therefore, allows operation at significantly higher power than would otherwise be possible. The recirculation system also controls reactivity over a wide span of reactor power by varying the recirculation flow rate to control the void content of the moderator. The Reactor Coolant Recirculation System consists of two recirculation pump loops external to the reactor vessel. These loops provide the piping path for the driving flow of water to the reactor vessel jet pumps. Each external loop contains one variable speed motor driven recirculation pump, a motor generator (MG) set to control pump speed and associated piping, jet pumps, valves, and instrumentation. The recirculation pump, piping, and valves are part of the reactor coolant pressure boundary and are located inside the drywell structure. The jet pumps are reactor vessel internals.

The recirculated coolant consists of saturated water from the steam separators and dryers that has been subcooled by incoming feedwater. This water passes down the annulus between the reactor vessel wall and the core shroud. A portion of the coolant flows from the vessel, through the two external recirculation loops, and becomes the driving flow for the jet pumps. Each of the two external recirculation loops discharges high pressure flow into an external manifold, from which individual recirculation inlet lines are routed to the jet pump risers within the reactor vessel. The remaining portion of the coolant mixture in the annulus becomes the suction flow for the jet pumps. This flow enters the jet pump at suction inlets and is accelerated by the driving flow. The drive flow and suction flow are mixed in the jet pump throat section. The total flow then passes through the jet pump diffuser section into the area below the core (lower plenum), gaining sufficient head in the process to drive the required flow upward through the core. The subcooled water enters the bottom of the fuel channels and contacts the fuel cladding, where heat

(continued)

BASES

BACKGROUND (continued)

is transferred to the coolant. As it rises, the coolant begins to boil, creating steam voids within the fuel channel that continue until the coolant exits the core. Because of reduced moderation, the steam voiding introduces negative reactivity that must be compensated for to maintain or to increase reactor power. The recirculation flow control allows operators to increase recirculation flow and sweep some of the voids from the fuel channel, overcoming the negative reactivity void effect. Thus, the reason for having variable recirculation flow is to compensate for reactivity effects of boiling over a wide range of power generation without having to move control rods and disturb desirable flux patterns.

Each recirculation loop is manually started from the control room. The MG set provides regulation of individual recirculation loop drive flows. The flow in each loop is manually controlled.

APPLICABLE SAFETY ANALYSES

The operation of the Reactor Coolant Recirculation System is an initial condition assumed in the design basis loss of coolant accident (LOCA) (Ref. 1). During a LOCA caused by a recirculation loop pipe break, the intact loop is assumed to provide coolant flow during the first few seconds of the accident. The initial core flow decrease is rapid because the recirculation pump in the broken loop ceases to pump reactor coolant to the vessel almost immediately. The pump in the intact loop coasts down relatively slowly. This pump coastdown governs the core flow response for the next several seconds until the jet pump suction is uncovered (Ref. 1). The analyses assume that both loops are operating at the same flow prior to the accident. However, the LOCA analysis was reviewed for the case with a flow mismatch between the two loops, with the pipe break assumed to be in the loop with the higher flow. While the flow coastdown and core response are potentially more severe in this assumed case (since the intact loop starts at a lower flow rate and the core response is the same as if both loops were operating at a lower flow rate), a small mismatch has been determined to be acceptable based on engineering judgement. The recirculation system is also assumed to have sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients (Ref. 2), which are analyzed in Chapter 15 of the FSAR.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Plant specific LOCA analyses have been performed assuming only one operating recirculation loop. These analyses have demonstrated that, in the event of a LOCA caused by a pipe break in the operating recirculation loop, the Emergency Core Cooling System response will provide adequate core cooling, provided that the APLHGR limit for SPC ATRIUM™-10 fuel is modified.

The transient analyses of Chapter 15 of the FSAR have also been performed for single recirculation loop operation and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR requirements are modified. During single recirculation loop operation, modification to the Reactor Protection System (RPS) average power range monitor (APRM) instrument setpoints is also required to account for the different relationships between recirculation drive flow and reactor core flow. The APLHGR, LHGR, and MCPR limits for single loop operation are specified in the COLR. The APRM Simulated Thermal Power-High setpoint is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation." In addition, a restriction on recirculation pump speed is incorporated to address reactor vessel internals vibration concerns and assumptions in the event analysis.

Recirculation loops operating satisfies Criterion 2 of the NRC Policy Statement (Ref. 5).

LCO

Two recirculation loops are required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. With the limits specified in SR 3.4.1.1 not met, the recirculation loop with the lower flow must be considered not in operation. With only one recirculation loop in operation, modifications to the required APLGHR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE"), LHGR limits (LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), and APRM Simulated Thermal Power—High setpoint (LCO 3.3.1.1) may be applied to allow continued operation consistent with the safety analysis assumptions. Furthermore, restrictions are placed on recirculation pump speed to ensure the initial assumption of the event analysis are maintained.

(continued)

BASES

LCO (continued)

The LCO is modified by a Note that allows up to 12 hours to establish the required limits and setpoints after a change from two recirculation loops operation to single recirculation loop operation. If the limits and setpoints are not in compliance with the applicable requirements at the end of this period, the ACTIONS required by the applicable specifications must be implemented. This time is provided to stabilize operation with one recirculation loop by: limiting flow in the operating loop, limiting total THERMAL POWER, monitor APRM and local power range monitor (LPRM) neutron flux noise levels; and, fully implementing and confirming the required limit and setpoint modifications.

APPLICABILITY

In MODES 1 and 2, requirements for operation of the Reactor Coolant Recirculation System are necessary since there is considerable energy in the reactor core and the limiting design basis transients and accidents are assumed to occur.

In MODES 3, 4, and 5, the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important.

ACTIONS

A.1

When operating with no recirculation loops operating in MODE 1, the potential for thermal-hydraulic oscillations is greatly increased. Although this transient is protected for expected modes of oscillation by the OPRM system, when OPERABLE per LCO 3.3.1.1, Function 2.f (Reference 3, 4), the prudent response to the natural circulation condition is to preclude potential thermal-hydraulic oscillations by immediately placing the mode switch in the shutdown position.

B.1

Recirculation loop flow must match within required limits when both recirculation loops are in operation. If flow mismatch is not within required limits, matched flow must be restored within 2 hours. If matched flows are not restored, the recirculation loop with lower flow must be declared "not in operation." Should a LOCA occur with recirculation loop flow not matched, the core flow coastdown and resultant core response may not be bounded by the LOCA analyses. Therefore, only a limited time is allowed prior to imposing restrictions associated with single loop operation. Operation with only one recirculation loop satisfies the requirements of the LCO and the initial conditions of the accident sequence.

(continued)

BASES

ACTIONS (continued)

The 2 hour Completion Time is based on the low probability of an accident occurring during this time period, providing a reasonable time to complete the Required Action, and considering that frequent core monitoring by operators allows abrupt changes in core flow conditions to be quickly detected.

These Required Actions do not require tripping the recirculation pump in the lowest flow loop when the mismatch between total jet pump flows of the two loops is greater than the required limits. However, in cases where large flow mismatches occur, low flow or reverse flow can occur in the low flow loop jet pumps, causing vibration of the jet pumps. If zero or reverse flow is detected, the condition should be alleviated by changing recirculation pump speed to re-establish forward flow or by tripping the pump.

C.1

With no recirculation loops in operation while in MODE 2 or if after going to single loop operations the required limits and setpoints cannot be established, the plant must be brought to MODE 3, where the LCO does not apply within 12 hours. In this condition, the recirculation loops are not required to be operating because of the reduced severity of DBAs and minimal dependence on the recirculation loop coastdown characteristics. The allowed Completion Time of 12 hours is reasonable to reach MODE 3 from full power conditions in an orderly manner without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

This SR ensures the recirculation loops are within the allowable limits for mismatch. At low core flow (i.e., < 75 million lbm/hr), the MCPR requirements provide larger margins to the fuel cladding integrity Safety Limit such that the potential adverse effect of early boiling transition during a LOCA is reduced. A larger flow mismatch can therefore be allowed when core flow is < 75 million lbm/hr. The recirculation loop jet pump flow, as used in this Surveillance, is the summation of the flows from all of the jet pumps associated with a single recirculation loop.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1 (continued)

The mismatch is measured in terms of core flow. If the flow mismatch exceeds the specified limits, the loop with the lower flow is considered inoperable. The SR is not required when both loops are not in operation since the mismatch limits are meaningless during single loop or natural circulation operation. The Surveillance must be performed within 24 hours after both loops are in operation. The 24 hour Frequency is consistent with the Surveillance Frequency for jet pump OPERABILITY verification and has been shown by operating experience to be adequate to detect off normal jet pump loop flows in a timely manner.

SR 3.4.1.2

As noted, this SR is only applicable when in single loop operation. This SR ensures the recirculation pump limit is maintained. The 24 hour Frequency is based on operating experience and the operators inherent knowledge of the current reactor status.

REFERENCES

1. FSAR, Section 6.3.3.7.
2. FSAR, Section 5.4.1.4.
3. GE NEDO-31960-A "BWROG Long Term Stability Solutions Licensing Methodology," November, 1995.
4. GE NEDO-31960-A "BWROG Long Term Stability Solutions Licensing Methodology, "Supplement 1," November 1995.
5. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

B 3.10 SPECIAL OPERATIONS

B 3.10.8 SHUTDOWN MARGIN (SDM) Test-Refueling

BASES

BACKGROUND

The purpose of this MODE 5 Special Operations LCO is to permit SDM testing to be performed for those plant configurations in which the reactor pressure vessel (RPV) head is either not in place or the head bolts are not fully tensioned.

LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," requires that adequate SDM be demonstrated following fuel movements or control rod replacement within the RPV. The demonstration must be performed prior to or within 4 hours after criticality is reached. This SDM test may be performed prior to or during the first startup following the refueling. Performing the SDM test prior to startup requires the test to be performed while in MODE 5, with the vessel head bolts less than fully tensioned (and possibly with the vessel head removed). While in MODE 5, the reactor mode switch is required to be in the shutdown or refuel position, where the applicable control rod blocks ensure that the reactor will not become critical. The SDM test requires the reactor mode switch to be in the startup/hot standby position, since more than one control rod will be withdrawn for the purpose of demonstrating adequate SDM. This Special Operations LCO provides the appropriate additional controls to allow withdrawing more than one control rod from a core cell containing one or more fuel assemblies when the reactor vessel head bolts are less than fully tensioned.

APPLICABLE SAFETY ANALYSES

Prevention and mitigation of unacceptable reactivity excursions during control rod withdrawal, with the reactor mode switch in the startup/hot standby position while in MODE 5, is provided by the intermediate range monitor (IRM) neutron flux scram (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), and control rod block instrumentation (LCO 3.3.2.1, "Control Rod Block Instrumentation"). The limiting reactivity excursion during startup conditions while in MODE 5 is the control rod drop accident (CRDA).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. For SDM tests performed within these defined sequences, the analyses of Reference 1 is applicable. However, for some sequences developed for the SDM testing, the control rod patterns assumed in the safety analyses of Reference 1 may not be met. Therefore, special CRDA analyses, performed in accordance with an NRC approved methodology, are required to demonstrate the SDM test sequence will not result in unacceptable consequences should a CRDA occur during the testing. For the purpose of this test, the protection provided by the normally required MODE 5 applicable LCOs, in addition to the requirements of this LCO, will maintain normal test operations as well as postulated accidents within the bounds of the appropriate safety analyses (Ref. 1). In addition to the added requirements for the RWM, APRM, and control rod coupling, the notch out mode is specified for control rod withdrawals that are not in conformance with the BPWS. Requiring the notch out mode limits withdrawal steps to a single notch, which limits inserted reactivity, and allows adequate monitoring of changes in neutron flux, which may occur during the test.

As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of the NRC Policy Statement apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. SDM tests may be performed while in MODE 2, in accordance with Table 1.1-1, without meeting this Special Operations LCO or its ACTIONS. For SDM tests performed while in MODE 5, additional requirements must be met to ensure that adequate protection against potential reactivity excursions is available. To provide additional scram protection, beyond the normally required IRMs, the APRMs are also required to be OPERABLE (LCO 3.3.1.1, Functions 2.a, 2.d, and 2.e) as though the reactor were in MODE 2. Because multiple control rods will be withdrawn and the reactor will potentially become critical, RPS MODE 2 requirements for Functions 2.a, 2.d, and 2.e of Table 3.3.1.1-1

(continued)

BASES

LCO (continued)

must be enforced and the approved control rod withdrawal sequence must be enforced by the RWM (LCO 3.3.2.1, Function 2, MODE 2), or must be verified by a second licensed operator or other qualified member of the technical staff. The SDM may be demonstrated during an in sequence control rod withdrawal, in which the highest worth control rod is analytically determined, or during local criticals, where the highest worth control rod is determined by analysis or testing.

Local critical tests require the withdrawal of control rods in a sequence that is not in conformance with the BPWS. This testing would therefore require bypassing or reprogramming of the rod worth minimizer to allow the withdrawal of rods not in conformance with BPWS, and therefore additional requirements must be met (see LCO 3.10.7, "Control Rod Testing – Operating").

Control rod withdrawals that do not conform to the banked position withdrawal sequence specified in LCO 3.1.6, "Rod Pattern Control," (i.e., out of sequence control rod withdrawals) must be made in the individual notched withdrawal mode to minimize the potential reactivity insertion associated with each movement.

Coupling integrity of withdrawn control rods is required to minimize the probability of a CRDA and ensure proper functioning of the withdrawn control rods, if they are required to scram. Because the reactor vessel head may be removed during these tests, no other CORE ALTERATIONS may be in progress. Furthermore, since the control rod scram function with the RCS at atmospheric pressure relies solely on the CRD accumulator, it is essential that the CRD charging water header remain pressurized. This Special Operations LCO then allows changing the Table 1.1-1 reactor mode switch position requirements to include the startup/hot standby position, such that the SDM tests may be performed while in MODE 5.

APPLICABILITY

These SDM test Special Operations requirements are only applicable if the SDM tests performed in accordance with LCO 3.1.1, "SDM" are to be performed while in MODE 5 with the reactor vessel head removed or the head bolts not fully tensioned. Additional requirements during these tests to

(continued)

BASES

APPLICABILITY (continued)

enforce control rod withdrawal sequences and restrict other CORE ALTERATIONS provide protection against potential reactivity excursions. Operations in all other MODES are unaffected by this LCO.

ACTIONS

A.1

With one or more control rods discovered uncoupled during this Special Operation, a controlled insertion of each uncoupled control rod is required; either to attempt recoupling, or to preclude a control rod drop. This controlled insertion is preferred since, if the control rod fails to follow the drive as it is withdrawn (i.e., is "stuck" in an inserted position), placing the reactor mode switch in the shutdown position per Required Action B.1 could cause substantial secondary damage. If recoupling is not accomplished, operation may continue, provided the control rods are fully inserted within 3 hours and disarmed (electrically or hydraulically) within 4 hours. Inserting a control rod ensures the shutdown and scram capabilities are not adversely affected. The control rod is disarmed to prevent inadvertent withdrawal during subsequent operations. The control rods can be hydraulically disarmed by closing the drive water and exhaust water isolation valves. Electrically the control rods can be disarmed by disconnecting power from all four directional control valve solenoids. Required Action A.1 is modified by a Note that allows the RWM to be bypassed if required to allow insertion of the inoperable control rods and continued operation. LCO 3.3.2.1, "Control Rod Block Instrumentation," Actions provide additional requirements when the RWM is bypassed to ensure compliance with the CRDA analysis.

The allowed Completion Times are reasonable, considering the small number of allowed inoperable control rods, and provide time to insert and disarm the control rods in an orderly manner and without challenging plant systems.

Condition A is modified by a Note allowing separate Condition entry for each uncoupled control rod. This is acceptable since the Required Actions for this Condition provide appropriate compensatory actions for each uncoupled control rod. Complying with the Required Actions may allow for continued operation. Subsequent uncoupled control rods

(continued)

BASES

ACTIONS

A.1 (continued)

are governed by subsequent entry into the Condition and application of the Required Actions.

B.1

With one or more of the requirements of this LCO not met for reasons other than an uncoupled control rod, the testing should be immediately stopped by placing the reactor mode switch in the shutdown or refuel position. This results in a condition that is consistent with the requirements for MODE 5 where the provisions of this Special Operations LCO are no longer required.

SURVEILLANCE REQUIREMENTS

SR 3.10.8.1

Performance of the applicable SRs for LCO 3.3.1.1, Functions 2.a and 2.d will ensure that the reactor is operated within the bounds of the safety analysis.

SR 3.10.8.1, SR 3.10.8.2, and SR 3.10.8.3

LCO 3.3.1.1, Functions 2.a, 2.d and 2.e, made applicable in this Special Operations LCO, are required to have applicable Surveillances met to establish that this Special Operations LCO is being met. However, the control rod withdrawal sequences during the SDM tests may be enforced by the RWM (LCO 3.3.2.1, Function 2, MODE 2 requirements) or by a second licensed operator or other qualified member of the technical staff. As noted, either the applicable SRs for the RWM (LCO 3.3.2.1) must be satisfied according to the applicable Frequencies (SR 3.10.8.2), or the proper movement of control rods must be verified (SR 3.10.8.3). This latter verification (i.e., SR 3.10.8.3) must be performed during control rod movement to prevent deviations from the specified sequence. These surveillances provide adequate assurance that the specified test sequence is being followed.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.10.8.4

Periodic verification of the administrative controls established by this LCO will ensure that the reactor is operated within the bounds of the safety analysis. The 12 hour Frequency is intended to provide appropriate assurance that each operating shift is aware of and verifies compliance with these Special Operations LCO requirements.

SR 3.10.8.5

Coupling verification is performed to ensure the control rod is connected to the control rod drive mechanism and will perform its intended function when necessary. The verification is required to be performed any time a control rod is withdrawn to the "full out" notch position, or prior to declaring the control rod OPERABLE after work on the control rod or CRD System that could affect coupling. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved as well as operating experience related to uncoupling events.

SR 3.10.8.6

CRD charging water header pressure verification is performed to ensure the motive force is available to scram the control rods in the event of a scram signal. A minimum accumulator pressure is specified, below which the capability of the accumulator to perform its intended function becomes degraded and the accumulator is considered inoperable. The minimum accumulator pressure of 940 psig is well below the expected pressure of 1100 psig. The 7 day Frequency has been shown to be acceptable through operating experience and takes into account indications available in the control room.

REFERENCE

1. XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors," Exxon Nuclear Company, March 1983.