MANUAL HARD COPY DISTRIBUTION
DOCUMENT TRANSMITTAL 2005-15008

USER INFORMATION:

~~[illegible]RLACH*ROSE [illegible]~~ EMPL#:028401 CA#: 0363

~~Addres[illegible] [illegible]CSA2~~

~~Phone#[illegible] 254[illegible]94~~

TRANSMITTAL INFORMATION:

TO: [redacted] 03/25/2005

LOCATION: USNRC

FROM: NUCLEAR RECORDS DOCUMENT CONTROL CENTER (NUCSA-2)

THE FOLLOWING CHANGES HAVE OCCURRED TO THE HARDCOPY OR ELECTRONIC MANUAL ASSIGNED TO YOU:

TSB2 - TECHNICAL SPECIFICATIONS BASES UNIT 2 MANUAL

REMOVE MANUAL TABLE OF CONTENTS DATE: 03/22/2005

ADD MANUAL TABLE OF CONTENTS DATE: 03/24/2005

CATEGORY: DOCUMENTS TYPE: TSB2
ID: TEXT 3.1.1
REMOVE: REV:0

ADD: REV: 1

CATEGORY: DOCUMENTS TYPE: TSB2
ID: TEXT 3.1.4
REMOVE: REV:0

A001

ADD: REV: 1

CATEGORY: DOCUMENTS TYPE: TSB2
ID: TEXT 3.1.6
REMOVE: REV:1

ADD: REV: 2

CATEGORY: DOCUMENTS TYPE: TSB2
ID: TEXT 3.10.7
REMOVE: REV:0

ADD: REV: 1

CATEGORY: DOCUMENTS TYPE: TSB2
ID: TEXT 3.10.8
REMOVE: REV:0

ADD: REV: 1

CATEGORY: DOCUMENTS TYPE: TSB2
ID: TEXT 3.2.1
REMOVE: REV:0

ADD: REV: 1

CATEGORY: DOCUMENTS TYPE: TSB2
ID: TEXT 3.2.2
REMOVE: REV:0

ADD: REV: 1

CATEGORY: DOCUMENTS TYPE: TSB2
ID: TEXT LOES
REMOVE: REV:56

ADD: REV:57

UPDATES FOR HARD COPY MANUALS WILL BE DISTRIBUTED WITHIN 5 DAYS IN ACCORDANCE WITH DEPARTMENT PROCEDURES. PLEASE MAKE ALL CHANGES AND ACKNOWLEDGE COMPLETE IN YOUR NIMS INBOX UPON RECEIPT OF HARD COPY. FOR ELECTRONIC MANUAL USERS, ELECTRONICALLY REVIEW THE APPROPRIATE DOCUMENTS AND ACKNOWLEDGE COMPLETE IN YOUR NIMS INBOX.

Manual Name: TSB2

Manual Title: TECHNICAL SPECIFICATIONS BASES UNIT 2 MANUAL

Table Of Contents

Issue Date: 03/24/2005

Manual Name: TSB2

Manual Title: TECHNICAL SPECIFICATIONS BASES UNIT 2 MANUAL

TEXT 3.1.7 0 11/18/2002
Title: REACTIVITY CONTROL SYSTEMS STANDBY LIQUID CONTROL (SLC) SYSTEM

TEXT 3.1.8 0 11/18/2002
Title: REACTIVITY CONTROL SYSTEMS SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN VALVES

TEXT 3.2.1 1 03/24/2005
Title: POWER DISTRIBUTION LIMITS AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

TEXT 3.2.2 1 03/24/2005
Title: POWER DISTRIBUTION LIMITS MINIMUM CRITICAL POWER RATIO (MCPR)

TEXT 3.2.3 0 11/18/2002
Title: POWER DISTRIBUTION LIMITS LINEAR HEAT GENERATION RATE (LHGR)

TEXT 3.2.4 0 11/18/2002
Title: POWER DISTRIBUTION LIMITS AVERAGE POWER RANGE MONITOR (APRM) GAIN AND SETPOINTS

TEXT 3.3.1.1 1 03/01/2005
Title: INSTRUMENTATION REACTOR PROTECTION SYSTEM (RPS) INSTRUMENTATION

TEXT 3.3.1.2 0 11/18/2002
Title: INSTRUMENTATION SOURCE RANGE MONITOR (SRM) INSTRUMENTATION

TEXT 3.3.1.3 0 11/22/2004
Title: OPRM INSTRUMENTATION

TEXT 3.3.2.1 1 02/17/2005
Title: INSTRUMENTATION CONTROL ROD BLOCK INSTRUMENTATION

TEXT 3.3.2.2 0 11/18/2002
Title: INSTRUMENTATION FEEDWATER - MAIN TURBINE HIGH WATER LEVEL TRIP INSTRUMENTATION

TEXT 3.3.3.1 0 11/18/2002
Title: INSTRUMENTATION POST ACCIDENT MONITORING (PAM) INSTRUMENTATION

LDCN 3710

Manual Name: TSB2

Manual Title: TECHNICAL SPECIFICATIONS BASES UNIT 2 MANUAL

TEXT 3.3.3.2 0 11/18/2002
Title: INSTRUMENTATION REMOTE SHUTDOWN SYSTEM

TEXT 3.3.4.1 0 11/18/2002
Title: INSTRUMENTATION END OF CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) INSTRUMENTATION

TEXT 3.3.4.2 0 11/18/2002
Title: INSTRUMENTATION ANTICIPATED TRANSIENT WITHOUT SCRAM RECIRCULATION PUMP TRIP (ATWS-RPT) INSTRUMENTATION

TEXT 3.3.5.1 2 01/24/2005
Title: INSTRUMENTATION EMERGENCY CORE COOLING SYSTEM (ECCS) INSTRUMENTATION

TEXT 3.3.5.2 0 11/18/2002
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 11/18/2002
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 11/18/2002
Title: INSTRUMENTATION REACTOR PROTECTION SYSTEM (RPS) ELECTRIC POWER MONITORING

TEXT 3.4.1 2 11/22/2004
Title: REACTOR COOLANT SYSTEM (RCS) RECIRCULATION LOOPS OPERATING

TEXT 3.4.2 0 11/18/2002
Title: REACTOR COOLANT SYSTEM (RCS) JET PUMPS

Manual Name: TSB2

Manual Title: TECHNICAL SPECIFICATIONS BASES UNIT 2 MANUAL

TEXT 3.4.3 0 11/18/2002

Title: REACTOR COOLANT SYSTEM (RCS) SAFETY/RELIEF VALVES (S/RVS)

TEXT 3.4.4 0 11/18/2002

Title: REACTOR COOLANT SYSTEM (RCS) RCS OPERATIONAL LEAKAGE

TEXT 3.4.5 0 11/18/2002

Title: REACTOR COOLANT SYSTEM (RCS) RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

TEXT 3.4.6 0 11/18/2002

Title: REACTOR COOLANT SYSTEM (RCS) RCS LEAKAGE DETECTION INSTRUMENTATION

TEXT 3.4.7 0 11/18/2002

Title: REACTOR COOLANT SYSTEM (RCS) RCS SPECIFIC ACTIVITY

TEXT 3.4.8 0 11/18/2002

Title: REACTOR COOLANT SYSTEM (RCS) RESIDUAL HEAT REMOVAL (RHR) SHUTDOWN COOLING SYSTEM - HOT SHUTDOWN

TEXT 3.4.9 0 11/18/2002

Title: REACTOR COOLANT SYSTEM (RCS) RESIDUAL HEAT REMOVAL (RHR) SHUTDOWN COOLING SYSTEM - COLD SHUTDOWN

TEXT 3.4.10 0 11/18/2002

Title: REACTOR COOLANT SYSTEM (RCS) RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

TEXT 3.4.11 0 11/18/2002

Title: REACTOR COOLANT SYSTEM (RCS) REACTOR STEAM DOME PRESSURE

TEXT 3.5.1 0 11/18/2002

Title: EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM ECCS - OPERATING

TEXT 3.5.2 0 11/18/2002

Title: EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM ECCS - SHUTDOWN

TEXT 3.5.3 0 11/18/2002

Title: EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM RCIC SYSTEM

Manual Name: TSB2

Manual Title: TECHNICAL SPECIFICATIONS BASES UNIT 2 MANUAL

TEXT 3.6.1.1 0 11/18/2002
Title: CONTAINMENT SYSTEMS PRIMARY CONTAINMENT

TEXT 3.6.1.2 0 11/18/2002
Title: CONTAINMENT SYSTEMS PRIMARY CONTAINMENT AIR LOCK

TEXT 3.6.1.3 0 11/18/2002
Title: CONTAINMENT SYSTEMS PRIMARY CONTAINMENT ISOLATION VALVES (PCIVS)

TEXT 3.6.1.4 0 11/18/2002
Title: CONTAINMENT SYSTEMS CONTAINMENT PRESSURE

TEXT 3.6.1.5 0 11/18/2002
Title: CONTAINMENT SYSTEMS DRYWELL AIR TEMPERATURE

TEXT 3.6.1.6 0 11/18/2002
Title: CONTAINMENT SYSTEMS SUPPRESSION CHAMBER-TO-DRYWELL VACUUM BREAKERS

TEXT 3.6.2.1 0 11/18/2002
Title: CONTAINMENT SYSTEMS SUPPRESSION POOL AVERAGE TEMPERATURE

TEXT 3.6.2.2 0 11/18/2002
Title: CONTAINMENT SYSTEMS SUPPRESSION POOL WATER LEVEL

TEXT 3.6.2.3 0 11/18/2002
Title: CONTAINMENT SYSTEMS RESIDUAL HEAT REMOVAL (RHR) SUPPRESSION POOL COOLING

TEXT 3.6.2.4 0 11/18/2002
Title: CONTAINMENT SYSTEMS RESIDUAL HEAT REMOVAL (RHR) SUPPRESSION POOL SPRAY

TEXT 3.6.3.1 0 11/18/2002
Title: CONTAINMENT SYSTEMS PRIMARY CONTAINMENT HYDROGEN RECOMBINERS

TEXT 3.6.3.2 0 11/18/2002
Title: CONTAINMENT SYSTEMS DRYWELL AIR FLOW SYSTEM

Manual Name: TSB2

Manual Title: TECHNICAL SPECIFICATIONS BASES UNIT 2 MANUAL

TEXT 3.6.3.3	0	11/18/2002
Title: CONTAINMENT SYSTEMS PRIMARY CONTAINMENT OXYGEN CONCENTRATION		
TEXT 3.6.4.1	2	03/01/2005
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	2	11/09/2004
Title: CONTAINMENT SYSTEMS STANDBY GAS TREATMENT (SGT) SYSTEM		
TEXT 3.7.1	0	11/18/2002
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	11/18/2002
Title: PLANT SYSTEMS CONTROL ROOM EMERGENCY OUTSIDE AIR SUPPLY (CREOAS) SYSTEM		
TEXT 3.7.4	0	11/18/2002
Title: PLANT SYSTEMS CONTROL ROOM FLOOR COOLING SYSTEM		
TEXT 3.7.5	0	11/18/2002
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	11/18/2002
Title: PLANT SYSTEMS SPENT FUEL STORAGE POOL WATER LEVEL		
TEXT 3.8.1	1	10/17/2003
Title: ELECTRICAL POWER SYSTEMS AC SOURCES - OPERATING		

Manual Name: TSB2

Manual Title: TECHNICAL SPECIFICATIONS BASES UNIT 2 MANUAL

TEXT 3.8.2	0	11/18/2002
Title: ELECTRICAL POWER SYSTEMS AC SOURCES - SHUTDOWN		
TEXT 3.8.3	0	11/18/2002
Title: ELECTRICAL POWER SYSTEMS DIESEL FUEL OIL, LUBE OIL, AND STARTING AIR		
TEXT 3.8.4	0	11/18/2002
Title: ELECTRICAL POWER SYSTEMS DC SOURCES - OPERATING		
TEXT 3.8.5	0	11/18/2002
Title: ELECTRICAL POWER SYSTEMS DC SOURCES - SHUTDOWN		
TEXT 3.8.6	0	11/18/2002
Title: ELECTRICAL POWER SYSTEMS BATTERY CELL PARAMETERS		
TEXT 3.8.7	0	11/18/2002
Title: ELECTRICAL POWER SYSTEMS DISTRIBUTION SYSTEMS - OPERATING		
TEXT 3.8.8	0	11/18/2002
Title: ELECTRICAL POWER SYSTEMS DISTRIBUTION SYSTEMS - SHUTDOWN		
TEXT 3.9.1	0	11/18/2002
Title: REFUELING OPERATIONS REFUELING EQUIPMENT INTERLOCKS		
TEXT 3.9.2	0	11/18/2002
Title: REFUELING OPERATIONS REFUEL POSITION ONE-ROD-OUT INTERLOCK		
TEXT 3.9.3	0	11/18/2002
Title: REFUELING OPERATIONS CONTROL ROD POSITION		
TEXT 3.9.4	0	11/18/2002
Title: REFUELING OPERATIONS CONTROL ROD POSITION INDICATION		
TEXT 3.9.5	0	11/18/2002
Title: REFUELING OPERATIONS CONTROL ROD OPERABILITY - REFUELING		

Manual Name: TSB2

Manual Title: TECHNICAL SPECIFICATIONS BASES UNIT 2 MANUAL

TEXT 3.9.7 0 11/18/2002
Title: REFUELING OPERATIONS RESIDUAL HEAT REMOVAL (RHR) - HIGH WATER LEVEL

TEXT 3.9.8 0 11/18/2002
Title: REFUELING OPERATIONS RESIDUAL HEAT REMOVAL (RHR) - LOW WATER LEVEL

TEXT 3.10.1 0 11/18/2002
Title: SPECIAL OPERATIONS INSERVICE LEAK AND HYDROSTATIC TESTING OPERATION

TEXT 3.10.2 0 11/18/2002
Title: SPECIAL OPERATIONS REACTOR MODE SWITCH INTERLOCK TESTING

TEXT 3.10.3 0 11/18/2002
Title: SPECIAL OPERATIONS SINGLE CONTROL ROD WITHDRAWAL - HOT SHUTDOWN

TEXT 3.10.4 0 11/18/2002
Title: SPECIAL OPERATIONS SINGLE CONTROL ROD WITHDRAWAL - COLD SHUTDOWN

TEXT 3.10.5 0 11/18/2002
Title: SPECIAL OPERATIONS SINGLE CONTROL ROD DRIVE (CRD) REMOVAL - REFUELING

TEXT 3.10.6 0 11/18/2002
Title: SPECIAL OPERATIONS MULTIPLE CONTROL ROD WITHDRAWAL - REFUELING

TEXT 3.10.7 1 03/24/2005
Title: SPECIAL OPERATIONS CONTROL ROD TESTING - OPERATING

TEXT 3.10.8 1 03/24/2005
Title: SPECIAL OPERATIONS SHUTDOWN MARGIN (SDM) TEST - REFUELING

Section	Title	Revision
TOC	Table of Contents	6
B 2.0	SAFETY LIMITS BASES	
	Page TS / B 2.0-1	1
	Page TS / B 2.0-2	2
	Page TS / B 2.0-3	3
	Page TS / B 2.0-4	4
	Page TS / B 2.0-5	1
	Pages B 2.0-6 through B 2.0-8	0
B 3.0	LCO AND SR APPLICABILITY BASES	
	Pages B 3.0-1 through B 3.0-7	0
	Pages TS / B 3.0-8 and TS / B 3.0-9	1
	Pages B 3.0-10 through B 3.0-12	0
	Pages TS / B 3.0-13 through TS / B 3.0-15	1
B 3.1	REACTIVITY CONTROL BASES	
	Pages B 3.1-1 through B 3.1-4	0
	Page TS / B 3.1-5	1
	Pages TS / B 3.1-6 and TS / B 3.1-7	2
	Pages B 3.1-8 through B 3.1-27	0
	Page TS / B 3.1-28	2
	Pages B 3.1-29 through B 3.1-33	0
	Pages TS / B 3.1.34 through TS / B 3.1-36	1
	Page TS / B 3.1-37	2
	Page TS / B 3.1-38	2
	Pages B 3.1-39 through B 3.1-51	0
B 3.2	POWER DISTRIBUTION LIMITS BASES	
	Pages TS / B 3.2-1 and TS / B 3.2-2	1
	Page TS / B 3.2-3	2
	Page TS / B 3.2-4	1
	Pages TS / B 3.2-5 and TS / B 3.2-6	3
	Page TS / B 3.2-7	2
	Pages TS / B 3.2-8 and TS / B 3.2-9	3
	Pages TS / B 3.2-10 through TS / B 3.2-19	1
B 3.3	INSTRUMENTATION	
	Pages TS / B 3.3-1 through TS / B 3.3-7	1
	Pages TS / B 3.3-8 through TS / B 3.3-11	2
	Pages TS / B 3.3-12 through TS / B 3.3-27	1
	Pages TS / B 3.3-28 through TS / B 3.3-30	2
	Page TS / B 3.3-31	1
	Pages TS / B 3.3-32 and TS / B 3.3-33	2
	Pages TS / B 3.3-34 through TS / B 3.3-43	1
	Pages TS / B 3.3-43a though TS / B 3.3-43i	0
	Pages TS / B 3.3-44 through TS / B 3.3-54	2

Section	Title	Revision
	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
	Page TS / B 3.3-68	4
	Pages TS / B 3.3-69 and TS / B 3.3-70	3
	Pages TS / B 3.3-71 through TS / B 3.3-75	2
	Page TS / B 3.3-75a	4
	Pages TS / B 3.3-75b through TS / B 3.3-75c	3
	Pages B 3.3-76 through B 3.3-91	0
	Pages TS / B 3.3-92 through TS / B 3.3-103	1
	Page TS / B 3.3-104	2
	Pages TS / B 3.3-105 and TS / B 3.3-106	1
	Page TS / B 3.3-107	2
	Page TS / B 3.3-108	1
	Page TS / B 3.3-109	2
	Pages TS / B 3.3-110 through TS / B 3.3-115	1
	Pages TS / B 3.3-116 through TS / B 3.3-118	2
	Pages TS / B 3.3-119 through TS / B 3.3-120	1
	Pages TS / B 3.3-121 and TS / B 3.3-122	2
	Page TS / B 3.3-123	1
	Page TS / B 3.3-124	2
	Page TS / B 3.3-124a	0
	Page TS / B 3.3-125	1
	Page TS / B 3.3-126	2
	Page TS / B 3.3-127	3
	Page TS / B 3.3-128	2
	Pages TS / B 3.3-129 through TS / B 3.3-131	1
	Page TS / B 3.3-132	2
	Pages TS / B 3.3-133 and TS / B 3.3-134	1
	Pages B 3.3-135 through B 3.3-137	0
	Page TS / B 3.3-138	1
	Pages B 3.3-139 through B 3.3-149	0
	Pages 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
	Page TS / B 3.3-178	2
	Page TS / B 3.3-179	3
	Page TS / B 3.3-179a	2
	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 3.3-205	0
	Page TS / B 3.3-206	1
	Pages B 3.3-207 through B 3.3-220	0

Section	Title	Revision
B 3.4	REACTOR COOLANT SYSTEM BASES	
	Pages TS / B 3.4-1 and TS / B 3.4-2	1
	Pages TS / B 3.4-3 through TS / B 3.4-9	3
	Pages B 3.4-10 through B 3.4-14	0
	Page TS / B 3.4-15	1
	Pages TS / B 3.4-16 and TS / B 3.4-17	2
	Page TS / B 3.4-18	1
	Pages B 3.4-19 through B 3.4-28	0
	Page TS / B 3.4-29	1
	Pages B 3.3-30 through B 3.3-48	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
	Pages TS / B 3.4-54 and TS / B 3.4-55	2
	Pages TS / B 3.4-56 through TS / B 3.4-60	1
B 3.5	ECCS AND RCIC BASES	
	Pages TS / B 3.5-1 and TS / B 3.5-2	1
	Page TS / B 3.5-3	2
	Pages TS / B 3.5-4 through TS / B 3.5-10	1
	Page TS / B 3.5-11	2
	Pages TS / B 3.5-12 through TS / B 3.5-14	1
	Pages TS / B 3.5-15 through TS / B.3.5-17	2
	Page TS / B 3.18	1
	Pages B 3.5-19 through B 3.5-24	0
	Page TS / B 3.5-25	1
	Pages B 3.5-26 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	3
	Pages TS / B 3.6-15a and TS / B 3.6-15b	0
	Page TS / B 3.6-16	1
	Page TS / B 3.6-17	2
	Page TS / B 3.6-17a	0
	Pages TS / B 3.6-18 and TS / B 3.6-19	1
	Page TS / B 3.6-20	2
	Page TS / B 3.6-21	3
	Pages TS / B 3.6-21a and TS / B 3.6-21b	0

Section	Title	Revision
	Pages TS / B 3.6-22 and TS / B 3.6-23	2
	Pages TS / B 3.6-24 through TS / B 3.6-26	1
	Page TS / B 3.6-27	3
	Page TS / B 3.6-28	6
	Page TS / B 3.6-29	3
	Page TS / B 3.6-29a	0
	Page TS / B 3.6-30	2
	Page TS / B 3.6-31	3
	Pages TS / B 3.6-32 through TS / B 3.6-34	1
	Pages TS / B 3.6-35 through TS / B 3.6-37	2
	Page TS / B 3.6-38	1
	Page TS / B 3.6-39	4
	Pages B 3.6-40 through B 3.6-42	0
	Pages TS / B 3.6-43 through TS / B 3.6-50	1
	Page TS / B 3.6-51	2
	Pages B 3.6-52 through B 3.6-62	0
	Page TS / B 3.6-63	1
	Pages B 3.6-64 through B 3.6-82	0
	Page TS / B 3.6-83	3
	Pages TS / B 3.6-84 and TS / B 3.6-85	2
	Pages TS / B 3.6-86 through TS / B 3.6-87a	1
	Page TS / B 3.6-88	3
	Page TS / B 3.6-89	1
	Page TS / B 3.3-90	3
	Pages TS / B 3.6-91 through TS / B 3.6-95	1
	Page TS / B 3.6-96	2
	Pages TS / B 3.6-97 and TS / B 3.6-98	1
	Page TS / B 3.6-99	2
	Page TS / B 3.6-99a	0
	Pages TS / B 3.6-100 and TS / B 3.6-101	1
	Pages TS / B 3.6-102 through TS / B 3.6-104	2
	Pages TS / B 3.6-105 and TS / B 3.6-106	1
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 3.7-6b and TS / B 3.7-6c	0
	Page TS / B 3.7-7	2
	Page TS / B 3.7-8	1
	Pages B 3.7-9 through B 3.7-11	0
	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 3.7-26	1
	Pages B 3.7-24 through B 3.7-26	0
	Pages TS / B 3.7-27 through TS / B 3.7-29	2
	Page TS / B 3.7-30	1

TSB2 LOES
2/28/05

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

SDM requirements are specified to ensure:

a. The reactor can be made subcritical from all operating conditions and transients and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits; and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

These requirements are satisfied by the control rods, as described in GDC 26 (Ref. 1), which can compensate for the reactivity effects of the fuel and water temperature changes experienced during all operating conditions.

APPLICABLE SAFETY ANALYSES

The control rod drop accident (CRDA) analysis (Refs. 2 and 3) assumes the core is subcritical with the highest worth control rod withdrawn. Typically, the first control rod withdrawn has a very high reactivity worth and, should the core be critical during the withdrawal of the first control rod, the consequences of a CRDA could exceed the fuel damage limits for a CRDA (see Bases for LCO 3.1.6, "Rod Pattern Control"). Also, SDM is assumed as an initial condition for the control rod removal error during refueling and fuel assembly insertion error during refueling accidents (Ref. 4). The analysis of these reactivity insertion events assumes the refueling interlocks are OPERABLE when the reactor is in the refueling mode of operation. These interlocks prevent the withdrawal of more than one control rod from the core during refueling. (Special consideration and requirements for multiple control rod withdrawal during refueling are covered in Special Operations LCO 3.10.6, "Multiple Control Rod Withdrawal—Refueling.") The analysis assumes this condition is acceptable since the core will be

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	shut down with the highest worth control rod withdrawn, if adequate SDM has been demonstrated. Prevention or mitigation of reactivity insertion events is necessary to limit energy deposition in the fuel to prevent significant fuel damage, which could result in undue release of radioactivity. Adequate SDM ensures inadvertent criticalities and potential CRDAs involving high worth control rods (namely the first control rod withdrawn) will not cause significant fuel damage. SDM satisfies Criterion 2 of the NRC Policy Statement (Ref. 5).
LCO	The specified SDM limit accounts for the uncertainty in the demonstration of SDM by testing. Separate SDM limits are provided for testing where the highest worth control rod is determined analytically or by measurement. This is due to the reduced uncertainty in the SDM test when the highest worth control rod is determined by measurement. When SDM is demonstrated by calculations not associated with a test (e.g., to confirm SDM during the fuel loading sequence), additional margin is included to account for uncertainties in the calculation. To ensure adequate SDM during the design process, a design margin is included to account for uncertainties in the design calculations (Ref. 6).
APPLICABILITY	In MODES 1 and 2, SDM must be provided because subcriticality with the highest worth control rod withdrawn is assumed in the CRDA analysis (Ref. 2). In MODES 3 and 4, SDM is required to ensure the reactor will be held subcritical with margin for a single withdrawn control rod. SDM is required in MODE 5 to prevent an open vessel, inadvertent criticality during the withdrawal of a single control rod from a core cell containing one or more fuel assemblies or a fuel assembly insertion error (Ref. 4).

(continued)

BASES (continued)

ACTIONS

A.1

With SDM not within the limits of the LCO in MODE 1 or 2, SDM must be restored within 6 hours. Failure to meet the specified SDM may be caused by a control rod that cannot be inserted. The allowed Completion Time of 6 hours is acceptable, considering that the reactor can still be shut down, assuming no failures of additional control rods to insert, and the low probability of an event occurring during this interval.

B.1

If the SDM cannot be restored, the plant must be brought to MODE 3 in 12 hours, to prevent the potential for further reductions in available SDM (e.g., additional stuck control rods). 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.

C.1

With SDM not within limits in MODE 3, the operator must immediately initiate action to fully insert all insertable control rods. Action must continue until all insertable control rods are fully inserted. This action results in the least reactive condition for the core.

D.1, D.2, D.3, and D.4

With SDM not within limits in MODE 4, the operator must immediately initiate action to fully insert all insertable control rods. Action must continue until all insertable control rods are fully inserted. This action results in the least reactive condition for the core. Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at least one Standby Gas Treatment (SGT) subsystem is OPERABLE; and secondary containment isolation capability (i.e., at least one secondary containment isolation valve and associated instrumentation are OPERABLE, or other acceptable

(continued)

BASES

ACTIONS

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

administrative controls to assure isolation capability) in each secondary containment penetration flow path not isolated and required to be isolated to mitigate radioactivity releases. This may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the surveillances needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, SRs may need to be performed to restore the component to OPERABLE status. Actions must continue until all required components are OPERABLE.

E.1, E.2, E.3, E.4, and E.5

With SDM not within limits in MODE 5, the operator must immediately suspend CORE ALTERATIONS that could reduce SDM (e.g., insertion of fuel in the core or the withdrawal of control rods). Suspension of these activities shall not preclude inserting control rods or removing fuel from the core to reduce the total reactivity.

Action must also be immediately initiated to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies have been fully inserted. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and therefore do not have to be inserted.

Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at least one SGT subsystem is OPERABLE; and secondary containment isolation capability (i.e., at least one secondary containment isolation valve and associated instrumentation are OPERABLE, or other acceptable administrative controls to assure isolation capability) in each associated penetration flow path not isolated that is

(continued)

BASES

ACTIONS

E.1, E.2, E.3, E.4, and E.5 (continued)

assumed to be isolated to mitigate radioactivity releases. This may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the Surveillances as needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, SRs may need to be performed to restore the component to OPERABLE status. Action must continue until all required components are OPERABLE.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

SDM must be verified to be within limits to ensure that the reactor can be made subcritical from any initial operating condition. Adequate SDM is demonstrated by testing before or during the first startup after fuel movement, control rod replacement, or shuffling within the reactor pressure vessel. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from another core location. Since core reactivity will vary during the cycle as a function of fuel depletion and poison burnup, the beginning of cycle (BOC) test must also account for changes in core reactivity during the cycle. Therefore, to obtain the SDM, the initial measured value must be increased by an adder, "R", which is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated BOC core reactivity. If the value of "R" is zero (that is, BOC is the most reactive point in the cycle), no correction to the BOC measured value is required (Ref. 6). For the SDM demonstrations that rely solely on calculation of the highest worth control rod, additional margin (0.10% $\Delta k/k$) must be added to the SDM limit of 0.28% $\Delta k/k$ to account for uncertainties in the calculation.

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

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 BPWS. This testing would therefore require re-programming or bypassing of the rod worth minimizer to allow the withdrawal of control rods not in conformance with BPWS, and therefore additional requirements must be met (see LCO 3.10.7, "Control Rod Testing—Operating").

The Frequency of 4 hours after reaching criticality is allowed to provide a reasonable amount of time to perform the required calculations and have appropriate verification.

During MODE 5, adequate SDM is required to ensure that the reactor does not reach criticality during control rod withdrawals. An evaluation of each planned in-vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures that the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload/reload sequences inherently satisfy the SR, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.

2. FSAR, Section 15.

3. 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.

4. FSAR, Section 15.4.1.1.

(continued)

BASES

REFERENCES (continued)

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

6. FSAR, Section 4.3.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Control Rod Scram Times

BASES

BACKGROUND

The scram function of the Control Rod Drive (CRD) System controls reactivity changes during abnormal operational transients to ensure that specified acceptable fuel design limits are not exceeded (Ref. 1). The control rods are scrammed by positive means using hydraulic pressure exerted on the CRD piston.

When a scram signal is initiated, control air is vented from the scram valves, allowing them to open by spring action. Opening the exhaust valve reduces the pressure above the main drive piston to atmospheric pressure, and opening the inlet valve applies the accumulator or reactor pressure to the bottom of the piston. Since the notches in the index tube are tapered on the lower edge, the collet fingers are forced open by cam action, allowing the index tube to move upward without restriction because of the high differential pressure across the piston. As the drive moves upward and the accumulator pressure reduces below the reactor pressure, a ball check valve opens, letting the reactor pressure complete the scram action. If the reactor pressure is low, such as during startup, the accumulator will fully insert the control rod in the required time without assistance from reactor pressure.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the control rod scram function are presented in References 2, 3, and 4. The Design Basis Accident (DBA) and transient analyses assume that all of the control rods scram at a specified insertion rate. The resulting negative scram reactivity forms the basis for the determination of plant thermal limits (e.g., the MCPR). Other distributions of scram times (e.g., several control rods scramming slower than the average time with several control rods scramming faster than the average time) can also provide sufficient scram reactivity. Surveillance of each individual control rod's scram time ensures the scram reactivity assumed in the DBA and transient analyses can be met.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The scram function of the CRD System protects the MCPR Safety Limit (SL) (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), which ensure that no fuel damage will occur if these limits are not exceeded. Above 800 psig, the scram function is designed to insert negative reactivity at a rate fast enough to prevent the actual MCPR from becoming less than the MCPR SL, during the analyzed limiting power transient. Below 800 psig, the scram function is assumed to perform during the control rod drop accident and, therefore, also provides protection against violating fuel damage limits during reactivity insertion accidents (Ref. 5) (see Bases for LCO 3.1.6, "Rod Pattern Control"). For the reactor vessel overpressure protection analysis, the scram function, along with the safety/relief valves, ensure that the peak vessel pressure is maintained within the applicable ASME Code limits.

Control rod scram times satisfy Criterion 3 of the NRC Policy Statement (Ref. 6).

LCO

The scram times specified in Table 3.1.4-1 (in the accompanying LCO) are required to ensure that the scram reactivity assumed in the DBA and transient analysis is met (Ref. 7). To account for single failures and "slow" scramming control rods, the scram times specified in Table 3.1.4-1 are faster than those assumed in the design basis analysis. The scram times have a margin that allows up to approximately 7% of the control rods (e.g., 185 x 7% $\approx$ 13) to have scram times exceeding the specified limits (i.e., "slow" control rods) including a single stuck control rod (as allowed by LCO 3.1.3, "Control Rod OPERABILITY") and an additional control rod failing to scram per the single failure criterion. The scram times are specified as a function of reactor steam dome pressure to account for the pressure dependence of the scram times. The scram times are specified relative to measurements based on reed switch positions, which provide the control rod position indication. The reed switch closes ("pickup") when the index tube passes a specific location and then opens ("dropout") as the index tube travels upward. Verification of the specified scram times in Table 3.1.4-1 is

(continued)

BASES

LCO (continued)

accomplished through measurement of the "dropout" times. To ensure that local scram reactivity rates are maintained within acceptable limits, no more than one "slow" control rod may occupy a face or diagonally adjacent location to any other "slow" or stuck control rod.

Table 3.1.4-1 is modified by two Notes which state that control rods with scram times not within the limits of the table are considered "slow" and that control rods with scram times > 7 seconds are considered inoperable as required by SR 3.1.3.4.

This LCO applies only to OPERABLE control rods since inoperable control rods will be inserted and disarmed (LCO 3.1.3). Slow scramming control rods may be conservatively declared inoperable and not accounted for as "slow" control rods.

APPLICABILITY

In MODES 1 and 2, a scram is assumed to function during transients and accidents analyzed for these plant conditions. These events are assumed to occur during startup and power operation; therefore, the scram function of the control rods is required during these MODES. In MODES 3 and 4, the control rods are not able to be withdrawn (except as permitted by LCO 3.10.3 and LCO 3.10.4) since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod scram capability during these conditions. Scram requirements in MODE 5 are contained in LCO 3.9.5, "Control Rod OPERABILITY—Refueling."

ACTIONS

A.1

When the requirements of this LCO are not met, the rate of negative reactivity insertion during a scram may not be within the assumptions of the safety analyses. Therefore, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. 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.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

The four SRs of this LCO are modified by a Note stating that during a single control rod scram time surveillance, the CRD pumps shall be isolated from the associated scram accumulator. With the CRD pump isolated, (i.e., charging valve closed) the influence of the CRD pump head does not affect the single control rod scram times. During a full core scram, the CRD pump head would be seen by all control rods and would have a negligible effect on the scram insertion times.

SR 3.1.4.1

The scram reactivity used in DBA and transient analyses is based on an assumed control rod scram time. Measurement of the scram times with reactor steam dome pressure ≥ 800 psig demonstrates acceptable scram times for the transients analyzed in References 3 and 4.

Maximum scram insertion times occur at a reactor steam dome pressure of approximately 800 psig because of the competing effects of reactor steam dome pressure and stored accumulator energy. Therefore, demonstration of adequate scram times at reactor steam dome pressure ≥ 800 psig ensures that the measured scram times will be within the specified limits at higher pressures. Limits are specified as a function of reactor pressure to account for the sensitivity of the scram insertion times with pressure and to allow a range of pressures over which scram time testing can be performed. To ensure that scram time testing is performed within a reasonable time following fuel movement within the reactor pressure vessel after a shutdown ≥ 120 days or longer, control rods are required to be tested before exceeding 40% RTP following the shutdown. In the event fuel movement is limited to selected core cells, it is the intent of this SR that only those CRDs associated with the core cells affected by the fuel movement are required to be scram time tested. However, if the reactor remains shutdown ≥ 120 days, all control rods are required to be scram time tested. This Frequency is acceptable considering the additional surveillances performed for control rod OPERABILITY, the frequent verification of adequate accumulator pressure, and the required testing of control rods affected by work on control rods or the CRD System.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.2

Additional testing of a sample of control rods is required to verify the continued performance of the scram function during the cycle. A representative sample contains at least 10% of the control rods. The sample remains representative if no more than 20% of the control rods in the sample tested are determined to be "slow." With more than 20% of the sample declared to be "slow" per the criteria in Table 3.1.4-1, additional control rods are tested until this 20% criterion (e.g., 20% of the entire sample size) is satisfied, or until the total number of "slow" control rods (throughout the core, from all surveillances) exceeds the LCO limit. For planned testing, the control rods selected for the sample should be different for each test. Data from inadvertent scrams should be used whenever possible to avoid unnecessary testing at power, even if the control rods with data may have been previously tested in a sample. The 120 day Frequency is based on operating experience that has shown control rod scram times do not significantly change over an operating cycle. This Frequency is also reasonable based on the additional Surveillances done on the CRDs at more frequent intervals in accordance with LCO 3.1.3 and LCO 3.1.5, "Control Rod Scram Accumulators."

SR 3.1.4.3

When work that could affect the scram insertion time is performed on a control rod or the CRD System, testing must be done to demonstrate that each affected control rod retains adequate scram performance over the range of applicable reactor pressures from zero to the maximum permissible pressure. The scram testing must be performed once before declaring the control rod OPERABLE. The required scram time testing must demonstrate the affected control rod is still within acceptable limits. The limits for reactor pressures < 800 psig are established based on a high probability of meeting the acceptance criteria at reactor pressures ≥ 800 psig. Limits for ≥ 800 psig are found in Table 3.1.4-1. If testing demonstrates the affected control rod does not meet these limits, but is within the 7-second limit of Table 3.1.4-1, Note 2, the control rod can be declared OPERABLE and "slow."

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.3 (continued)

Specific examples of work that could affect the scram times are (but are not limited to) the following: removal of any CRD for maintenance or modification; replacement of a control rod; and maintenance or modification of a scram solenoid pilot valve, scram valve, accumulator, isolation valve or check valve in the piping required for scram.

The Frequency of once prior to declaring the affected control rod OPERABLE is acceptable because of the capability to test the control rod over a range of operating conditions and the more frequent surveillances on other aspects of control rod OPERABILITY.

SR 3.1.4.4

When work that could affect the scram insertion time is performed on a control rod or CRD System, testing must be done to demonstrate each affected control rod is still within the limits of Table 3.1.4-1 with the reactor steam dome pressure ≥ 800 psig. Where work has been performed at high reactor pressure, the requirements of SR 3.1.4.3 and SR 3.1.4.4 can be satisfied with one test. For a control rod affected by work performed while shut down, however, a zero pressure and high pressure test may be required. This testing ensures that, prior to withdrawing the control rod for continued operation, the control rod scram performance is acceptable for operating reactor pressure conditions. Alternatively, a control rod scram test during hydrostatic pressure testing could also satisfy both criteria.

The Frequency of once prior to exceeding 40% RTP is acceptable because of the capability to test the control rod over a range of operating conditions and the more frequent surveillances on other aspects of control rod OPERABILITY.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. FSAR, Section 4.3.2.
3. FSAR, Section 4.6.

(continued)

BASES

REFERENCES (continued)

4. FSAR, Section 15.0

5. FSAR, Section 15.4.9.

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

7. Letter from R.F. Janecek (BWROG) to R.W. Starostecki (NRC), "BWR Owners Group Revised Reactivity Control System Technical Specifications," BWROG-8754, September 17, 1987.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Rod Pattern Control

BASES

BACKGROUND

Control rod patterns during startup conditions are controlled by the operator and the rod worth minimizer (RWM) (LCO 3.3.2.1, "Control Rod Block Instrumentation"), so that only specified control rod sequences and relative positions are allowed over the operating range of all control rods inserted to 10% RTP. The sequences limit the potential amount of reactivity addition that could occur in the event of a Control Rod Drop Accident (CRDA).

This Specification assures that the control rod patterns are consistent with the assumptions of the CRDA analyses of References 1 and 2.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the CRDA are summarized in References 1 and 2. CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analysis. The RWM (LCO 3.3.2.1) provides backup to operator control of the withdrawal sequences to ensure that the initial conditions of the CRDA analysis are not violated.

Prevention or mitigation of positive reactivity insertion events is necessary to limit the energy deposition in the fuel, thereby preventing significant fuel damage which could result in the undue release of radioactivity. Since the failure consequences for UO_2 have been shown to be insignificant below fuel energy depositions of 300 cal/gm (Ref. 3), the fuel damage limit of 280 cal/gm provides a margin of safety from significant core damage which would result in release of radioactivity (Refs. 4 and 5). Generic evaluations (Ref. 1 & 6) of a design basis CRDA have shown that the maximum reactor pressure will be less than the required ASME Code limits (Ref.7). The offsite doses are calculated each cycle using the methodology in reference 1 to demonstrate that the calculated offsite doses will be well within the required limits (Ref. 5). Control rod patterns analyzed in Reference 1 follow the banked position withdrawal sequence (BPWS). The BPWS is applicable from the condition of all control rods fully inserted to 10% RTP (Ref. 2). For the BPWS, the control rods are required to be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

(e.g., between notches 08 and 12). The banked positions are established to minimize the maximum incremental control rod worth without being overly restrictive during normal plant operation. For each reload cycle the CRDA is analyzed to demonstrate that the 280 cal/gm fuel damage limit will not be violated during a CRDA while following the BPWS mode of operation for control rod patterns. These analyses consider the effects of fully inserted inoperable and OPERABLE control rods not withdrawn in the normal sequence of BPWS, but are still in compliance with the BPWS requirements regarding out of sequence control rods. These requirements allow a limited number (i.e., eight) and distribution of fully inserted inoperable control rods.

When performing a shutdown of the plant, an optional BPWS control rod sequence (Ref. 9) may be used provided that all withdrawn control rods have been confirmed to be coupled prior to reaching THERMAL POWER of ≤10% RTP. 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. When using the Reference 9 control rod sequence for shutdown, the RWM 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 LCO 3.3.2.1, Condition D controls.

In order to use the Reference 9 BPWS shutdown process, an extra check is required in order to consider a control rod to be "confirmed" to be coupled. This extra check ensures that no Single Operator Error can result in an incorrect coupling check. For purposes of this shutdown process, the method for confirming that control rods are coupled varies depending on the position of the control rod in the core. Details on this coupling confirmation requirement are provided in Reference 9, which requires that any partially inserted control rods, which have not been confirmed to be coupled since their last withdrawal, be fully inserted prior to reaching THERMAL POWER of ≤10% RTP. If a control rod has been checked for coupling at notch 48 and the rod has since only been moved inward, this rod is in contact with it's drive and is not required to be fully inserted prior to reaching 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. This extra check may be performed as an administrative check, by examining logs, previous

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	surveillance's or other information. If the requirements for use of the BPWS control rod insertion process contained in Reference 9 are followed, the plant is considered to be in compliance with the BPWS requirements, as required by LOC 3.1.6. Rod pattern control satisfies Criterion 3 of the NRC Policy Statement (Ref. 8).
LCO	Compliance with the prescribed control rod sequences minimizes the potential consequences of a CRDA by limiting the initial conditions to those consistent with the BPWS. This LCO only applies to OPERABLE control rods. For inoperable control rods required to be inserted, separate requirements are specified in LCO 3.1.3, "Control Rod OPERABILITY," consistent with the allowances for inoperable control rods in the BPWS.
APPLICABILITY	In MODES 1 and 2, when THERMAL POWER is ≤ 10% RTP, the CRDA is a Design Basis Accident and, therefore, compliance with the assumptions of the safety analysis is required. When THERMAL POWER is > 10% RTP, there is no credible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA (Ref. 2). In MODES 3, 4, and 5, since the reactor is shut down and 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 remain subcritical with a single control rod withdrawn.
ACTIONS	<u>A.1 and A.2</u> With one or more OPERABLE control rods not in compliance with the prescribed control rod sequence, actions may be taken to either correct the control rod pattern or declare the associated control rods inoperable within 8 hours. Noncompliance with the prescribed sequence may be the result of "double notching," drifting from a control rod drive cooling water transient, leaking scram valves, or a power reduction to ≤ 10% RTP before establishing the correct control rod pattern. The number of OPERABLE control rods not in compliance with the prescribed sequence is limited to eight, to prevent the operator from attempting to correct a control rod pattern that significantly deviates from the prescribed sequence. When the control

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

rod pattern is not in compliance with the prescribed sequence, all control rod movement should be stopped except for moves needed to correct the rod pattern, or scram if warranted.

Required Action A.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.2.1 requires verification of control rod movement by a qualified member of the technical staff. This ensures that the control rods will be moved to the correct position. A control rod not in compliance with the prescribed sequence is not considered inoperable except as required by Required Action A.2. OPERABILITY of control rods is determined by compliance with LCO 3.1.3, "Control Rod OPERABILITY," LCO 3.1.4, "Control Rod Scram Times," and LCO 3.1.5, "Control Rod Scram Accumulators." The allowed Completion Time of 8 hours is reasonable, considering the restrictions on the number of allowed out of sequence control rods and the low probability of a CRDA occurring during the time the control rods are out of sequence.

B.1 and B.2

If nine or more OPERABLE control rods are out of sequence, the control rod pattern significantly deviates from the prescribed sequence. Control rod withdrawal should be suspended immediately to prevent the potential for further deviation from the prescribed sequence. Control rod insertion to correct control rods withdrawn beyond their allowed position is allowed since, in general, insertion of control rods has less impact on control rod worth than withdrawals have. Required Action B.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.2.1 requires verification of control rod movement by a qualified member of the technical staff.

When nine or more OPERABLE control rods are not in compliance with BPWS, the reactor mode switch must be placed in the shutdown position within 1 hour. With the mode switch in shutdown, the reactor is shut down, and as such, does not meet the applicability requirements of this LCO. The allowed Completion Time of 1 hour is reasonable to allow insertion of control rods to restore compliance, and is appropriate relative to the low probability of a CRDA occurring with the control rods out of sequence.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

The control rod pattern is verified to be in compliance with the BPWS at a 24 hour Frequency to ensure the assumptions of the CRDA analyses are met. The 24 hour Frequency was developed considering that the primary check on compliance with the BPWS is performed by the RWM (LCO 3.3.2.1), which provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at ≤ 10% RTP.

REFERENCES

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.
2. "Modifications to the Requirements for Control Rod Drop Accident Mitigating System," BWR Owners Group, July 1986.
3. NUREG-0979, Section 4.2.1.3.2, April 1983.
4. NUREG-0800, Section 15.4.9, Revision 2, July 1981.
5. 10 CFR 100.11.
6. NEDO-21778-A, "Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors," December 1978.
7. ASME, Boiler and Pressure Vessel Code.
8. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).
9. NEDO 33091-A, Revision 2, "Improved BPWS Control Rod Insertion Process," April 2003.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that limits specified in 10 CFR 50.46 are not exceeded during the postulated design basis loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES

SPC performed LOCA calculations for the SPC ATRIUM™-10 fuel design. The analytical methods and assumptions used in evaluating the fuel design limits from 10 CFR 50.46 are presented in References 3, 4, 5, and 6 for the SPC analysis. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs) that determine the APLHGR Limits are presented in References 3 through 9.

LOCA analyses are performed to ensure that the APLHGR limits are adequate to meet the Peak Cladding Temperature (PCT), maximum cladding oxidation, and maximum hydrogen generation limits of 10 CFR 50.46. The analyses are performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. A complete discussion of the analysis codes are provided in References 3, 4, 5, and 6 for the SPC analysis. The PCT following a postulated LOCA is a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within the assembly.

APLHGR limits are developed as a function of fuel type and exposure. The SPC analysis is valid for full cores of ATRIUM™-10 fuel. The SPC LOCA analyses also consider several alternate operating modes in the development of the APLHGR limits (e.g. Extended Load Line Limit Analysis (ELLA), Suppression Pool Cooling Mode, and Single Loop Operation (SLO)). LOCA analyses were performed for the regions of the power/flow map bounded by the 100% rod line and the APRM rod block line (i.e., the ELLA region). The ELLA region is analyzed to determine whether an APLHGR multiplier as a function of core flow is required. The results of the analysis demonstrate the PCTs are within the 10 CFR 50.46 limit, and that APLHGR multipliers as a function of core flow are not required.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SPC LOCA analyses consider the delay in Low Pressure Coolant Injection (LPCI) availability when the unit is operating in the Suppression Pool Cooling Mode. The delay in LPCI availability is due to the time required to realign valves from the Suppression Pool Cooling Mode to the LPCI mode. The results of the analyses demonstrate that the PCTs are within the 10 CFR 50.46 limit.

Finally, the SPC LOCA analyses were performed for Single-Loop Operation. The results of the SPC analysis for ATRIUM™-10 fuel shows that an APLHGR limit which is 0.8 times the two-loop APLHGR limit meets the 10 CFR 50.46 acceptance criteria, and that the PCT is less than the limiting two-loop PCT.

The APLHGR satisfies Criterion 2 of the NRC Policy Statement (Ref. 10).

LCO

The APLHGR limits specified in the COLR are the result of the DBA analyses.

APPLICABILITY

The APLHGR limits are primarily derived from LOCA analyses that are assumed to occur at high power levels. Design calculations and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. At THERMAL POWER levels < 25% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

ACTIONS

A.1

If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the DBA may not be met. Therefore, prompt action should be taken to restore the APLHGR(s) to within the required limits such that the plant operates within analyzed conditions. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limits and is acceptable based on the low probability of a DBA occurring simultaneously with the APLHGR out of specification.

(continued)

BASES

ACTIONS (continued)

B.1

If the APLHGR 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 must be 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 3.2.1.1

APLHGRs are required to be initially calculated within 24 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. Additionally, APLHGRs must be calculated prior to exceeding 50% RTP unless performed in the previous 24 hours. APLHGRs are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. 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 APLHGRs must be calculated prior to exceeding 50% RTP.

REFERENCES

1. Not Used
2. Not Used
3. EMF-2361(P)(A), "EXEM BWR-2000 ECCS Evaluation Model," Framatome ANP.
4. ANF-CC-33(P)(A) Supplement 2, "HUXY: A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option," January 1991.
5. XN-CC-33(P)(A) Revision 1, "HUXY: A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option Users Manual," November 1975.

(continued)

BASES

REFERENCES (continued)

6. XN-NF-80-19(P)(A), Volumes 2, 2A, 2B, and 2C "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," September 1982.

7. FSAR, Chapter 4.

8. FSAR, Chapter 6.

9. FSAR, Chapter 15.

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

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). The operating limit MCPR is established to ensure that no fuel damage results during anticipated operational occurrences (AOOs). Although fuel damage does not necessarily occur if a fuel rod actually experienced boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented in References 2 through 10. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained.

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency These analyses may also consider other

(continued)

APPLICABLE SAFETY ANALYSES (continued)

combinations of plant conditions (i.e., control rod scram speed, bypass valve performance, EOC-RPT, cycle exposure, etc.). Flow dependent MCPR limits are determined by analysis of slow flow runout transients.

The MCPR satisfies Criterion 2 of the NRC Policy Statement (Ref. 11).

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The operating limit MCPR is determined by the larger of the flow dependent MCPR and power dependent MCPR limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Studies of the variation of limiting transient behavior have been performed over the range of power and flow conditions. These studies encompass the range of key actual plant parameter values important to typically limiting transients. The results of these studies demonstrate that a margin is expected between performance and the MCPR requirements, and that margins increase as power is reduced to 25% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 25% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.

ACTIONS

A.1

If any MCPR is outside the required limits, an assumption regarding an initial condition of the design basis transient analyses may not be met. Therefore, prompt action should be taken to restore the MCPR(s) to within the required limits such that the plant remains operating within

(continued)

ACTIONS

A.1 (continued)

analyzed conditions. The 2 hour Completion Time is normally sufficient to restore the MCPR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the MCPR out of specification.

B.1

If the MCPR 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 must be 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 3.2.2.1

The MCPR is required to be initially calculated within 24 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. Additionally, MCPR must be calculated prior to exceeding 50% RTP unless performed in the previous 24 hours. MCPR is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. 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 MCPR must be calculated prior to exceeding 50% RTP.

SR 3.2.2.2

Because the transient analysis takes credit for conservatism in the scram time performance, it must be demonstrated that the specific scram time is consistent with those used in the transient analysis. SR 3.2.2.2 compares the average measured scram times to the assumed scram times documented in the COLR. The COLR contains a table of scram times based on the LCO 3.1.4, "Control Rod Scram Times" and the realistic scram times, both of which are used in the transient analysis. If the average measured scram times are greater than the realistic scram times then the MCPR operating limits corresponding to the Maximum Allowable Average Scram Insertion Time must be implemented.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.2.2 (continued)

Determining MCPR operating limits based on interpolation between scram insertion times is not permitted. The average measured scram times and corresponding MCPR operating limit must be determined once within 72 hours after each set of scram time tests required by SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3 and SR 3.1.4.4 because the effective scram times may change during the cycle. The 72 hour Completion Time is acceptable due to the relatively minor changes in average measured scram times expected during the fuel cycle.

REFERENCES

1. NUREG-0562, June 1979.

2. 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.

3. XN-NF-80-19(P)(A) Volume 3, Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description," Exxon Nuclear Company, January 1987.

4. ANF-913(P)(A) Volume 1, Revision 1 and Volume 1 Supplements 2, 3, and 4, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," Advanced Nuclear Fuels Corporation, August 1990.

5. XN-NF-80-19 (P)(A), Volume 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, June 1986.

6. NE-092-001, Revision 1, "Susquehanna Steam Electric Station Units 1 & 2: Licensing Topical Report for Power Uprate with Increased Core Flow," December 1992, and NRC Approval Letter: Letter from T. E. Murley (NRC) to R. G. Byram (PP&L), "Licensing Topical Report for Power Uprate With Increased Core Flow, Revision 0, Susquehanna Steam Electric Station, Units 1 and 2 (PLA-3788) (TAC Nos. M83426 and M83427)," November 30, 1993.

7. EMF-1997, Revision 0 (October 1997) and Supplement 1, Revision 0 (January 1998), "ANFB-10 Critical Power Correlation," and associated NRC SER dated 7/17/98.

(continued)

BASES

Reference (continued)

8. XN-NF-79-71(P)(A) Revision 2, Supplements 1, 2, and 3, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," March 1986.

9. XN-NF-84-105(P)(A), Volume 1 and Volume 1 Supplements 1 and 2, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," February 1987.

10. ANF-1358(P)(A) Revision 1, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," Advanced Nuclear Fuels Corporation, September 1992.

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

B 3.10 SPECIAL OPERATIONS

B 3.10.7 Control Rod Testing-Operating

BASES

BACKGROUND

The purpose of this Special Operations LCO is to permit control rod testing, while in MODES 1 and 2, by imposing certain administrative controls. Control rod patterns during startup conditions are controlled by the operator and the rod worth minimizer (RWM) (LCO 3.3.2.1, "Control Rod Block Instrumentation"), such that only the specified control rod sequences and relative positions required by LCO 3.1.6, "Rod Pattern Control," are allowed over the operating range from all control rods inserted to the low power setpoint (LPSP) of the RWM. The sequences effectively limit the potential amount and rate of reactivity increase that could occur during a control rod drop accident (CRDA). During these conditions, control rod testing is sometimes required that may result in control rod patterns not in compliance with the prescribed sequences of LCO 3.1.6. These tests include SDM demonstrations, control rod scram time testing, control rod friction testing, and testing performed during the Startup Test Program (e.g. local criticality). This Special Operations LCO provides the necessary exemption to the requirements of LCO 3.1.6 and provides additional administrative controls to allow the deviations in such tests from the prescribed sequences in LCO 3.1.6.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the CRDA area summarized in References 1 and 2. CRDA analyses assume the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analyses. The RWM provides backup to operator control of the withdrawal sequences to ensure the initial conditions of the CRDA analyses are not violated. For special sequences developed for control rod testing, the initial control rod patterns assumed in the safety analysis of References 1 and 2 may not be preserved. Therefore special CRDA analyses are required to demonstrate that these special sequences will not result in unacceptable consequences, should a CRDA occur during the testing. These analyses, performed in accordance with an NRC approved methodology, are dependent on the specific test being performed.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	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. Control rod testing may be performed in compliance with the prescribed sequences of LCO 3.1.6, and during these tests, no exceptions to the requirements of LCO 3.1.6 are necessary. For testing performed with a sequence not in compliance with LCO 3.1.6, the requirements of LCO 3.1.6 may be suspended, provided additional administrative controls are placed on the test to ensure that the assumptions of the special safety analysis for the test sequence are satisfied. Assurances that the test sequence is followed can be provided by either programming the test sequence into the RWM, with conformance verified as specified in SR 3.3.2.1.8 and allowing the RWM to monitor control rod withdrawal and provide appropriate control rod blocks if necessary, or by verifying conformance to the approved test sequence by a second licensed operator or other qualified member of the technical staff. These controls are consistent with those normally applied to operation in the startup range as defined in the SRs and ACTIONS of LCO 3.3.2.1, "Control Rod Block Instrumentation."
APPLICABILITY	Control rod testing, while in MODES 1 and 2, with THERMAL POWER greater than the LPSP of the RWM, is adequately controlled by the existing LCOs on power distribution limits and control rod block instrumentation. Control rod movement during these conditions is not restricted to prescribed sequences and can be performed within the constraints of LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," LCO 3.2.2, 'MINIMUM CRITICAL POWER RATIO (MCPR)," LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," and LCO 3.3.2.1. With THERMAL POWER less than or equal to the LPSP of the RWM, the provisions of this Special Operations LCO are necessary to perform special tests that are not in conformance with the prescribed sequences of LCO 3.1.6.

(continued)

BASES

APPLICABILITY (continued)

While in MODES 3 and 4, control rod withdrawal is only allowed if performed in accordance with Special Operations LCO 3.10.3, "Single Control Rod Withdrawal-Hot Shutdown," of Special Operations LCO 3.10.4, "Single Control Rod Withdrawal-Cold Shutdown," which provide adequate controls to ensure that the assumptions of the safety analyses of Reference 1 and 2 are satisfied. During these Special Operations and while in MODE 5, the one-rod-out interlock (LCO 3.9.2, "Refuel Position One-Rod-Out Interlock,") and scram functions (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," and LCO 3.9.5, "Control Rod OPERABILITY-Refueling"), or the added administrative controls prescribed in the applicable Special Operations LCOs, provide mitigation of potential reactive excursions.

ACTIONS

A.1

With the requirements of the LCO not met (e.g., the control rod pattern is not in compliance with the special test sequence, the sequence is improperly loaded in the RWM) the testing is required to be immediately suspended. Upon suspension of the special test, the provisions of LCO 3.1.6 are no longer expected, and appropriate actions are to be taken to restore the control rod sequence to the prescribed sequence of LCO 3.1.6, or to shut down the reactor, if required by LCO 3.1.6.

SURVEILLANCE REQUIREMENTS

SR 3.10.7.1

With the special test sequence not programmed into the RWM, a second licensed operator or other qualified member of the technical staff is required to verify conformance with the approved sequence for the test. This verification must be performed during control rod movement to prevent deviations from the specified sequence. A Note is added to indicate that this Surveillance does not need to be performed if SR 3.10.7.2 is satisfied.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)	SR 3.10.7.2 When the RWM provides conformance to the special test sequence, the test sequence must be verified to be correctly loaded into the RWM prior to control rod movement. This Surveillance demonstrates compliance with SR 3.3.2.1.8, thereby demonstrating that the RWM is OPERABLE. A Note has been added to indicate that this Surveillance does not need to be performed if SR 3.10.7.1 is satisfied.
REFERENCE	1. FSAR 15.4.9 2. 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.

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 and 2.d) 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 and 2.d 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 and 2.d, 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 verfies 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.