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NG-00-0920

10 CFR 50.71(e)

Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
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Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License No: DPR-49
Transmittal of the Current DAEC Technical Specifications Bases and Revision 15 of the DAEC Updated Final Safety Analysis Report

File: A-116, A-365

This letter transmits the latest revision (No. 15) of the Updated Final Safety Analysis Report (UFSAR) for the Duane Arnold Energy Center (DAEC) as required by 10 CFR Section 50.71(e). Attachment 1 to this letter summarizes the changes included in this revision. Attachment 2 to this letter contains the latest revision of the DAEC Technical Requirements Manual (TRM). The TRM is incorporated by reference into UFSAR Chapter 16; the copy in Attachment 2 is provided for your information. Attachment 3 (10 copies enclosed) contains the UFSAR Revision 15 changed pages and accurately represents all physical changes made to the plant and a number of changes made to the UFSAR since the previous submittal in November, 1998.

The DAEC Technical Specification Bases Control Program requires that changes to the Bases implemented without prior NRC approval be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). Therefore, in accordance with the DAEC Technical Specifications 5.5.10.d, IES Utilities Inc. provides a copy of the current revision of the Technical Specifications Bases, dated May 12, 2000, as Attachment 4.

A053 1/11

NRR-091

This letter is true and accurate to the best of my knowledge and belief.

IES UTILITIES INC.

By Gary D. Van Middlesworth for DW
David Wilson
Vice President, Nuclear

State of Iowa
(County) of Linn

Signed and sworn to before me on this 1st day of June, 2000,
by Gary D. Van Middlesworth.

NOTARIAL SEAL IOWA NANCY S. FRANCK

Nancy S. Franck
Notary Public in and for the State of Iowa

9-28-01
Commission Expires

Attachments: 1) Summary of Changes
2) Revision 3 of DAEC TRM
3) Revision 15 of the DAEC UFSAR
4) DAEC Technical Specification Bases

cc: L. B. Swenzinski (w/o)
E. Protsch (w/o)
B. Mozafari (NRC-NRR)
J. Dyer (Region III)
NRC Resident Office
Docu

Summary of Changes

This revision (15) contains changes that resulted from physical changes to the plant since the UFSAR submittal on November 19, 1998 (NG-98-1918). These changes are described in our cyclic report pursuant to 10CFR50.59(b) which has been submitted by separate letter (NG-00-0229). Revision 15 also contains other changes that were identified via IES initiatives for continuing UFSAR improvement. These changes, when evaluated in accordance with 10 CFR 50.59(a), did not involve an unreviewed safety question. Various editorial changes were made.

Specific changes in the text are indicated by revision bars in the margin.

Revision 15, excluding editorial changes, is summarized below.

Change Memo Number	Description of Change
97-018	Revise HPCI Turbine Exhaust High Pressure Trip Point Per Maintenance Actions EMAs A40693 And A40694
97-063	Revise Well Water Description Per Modification ECP 1589
97-096	Add Design Drawing APED-B11-1<4> As UFSAR Figure 5.3-6
98-073	Reflect Revision 22 Of QAM In UFSAR Chapter 17.2
98-111	Revise Recirculation System Description To Reflect Previous Plant Modifications
98-115	Add LPCI Crosstie Description Per Modification ECP 1614
98-117	Clarify Description Of Secondary Containment Interlock Function Per Modification ECP 1592
98-119	Revise Fire Brigade Equipment And Training Commitments
98-122	Revise Reactor Water Cleanup Isolation Valves Stroke Times
98-123	Revise Diesel Generator Inspection And Testing Description
98-124	Incorporate Description Of Temporary Radwaste Equipment
98-125	Reflect Changes To The Chemistry Control Program For General Service Water And Circulating Water Systems
98-126	Revise UFSAR To Reflect Severe Accident Guidelines (SAGs)
98-127	Remove Reactor Protection System Trip Per Amendment 193
99-001	Remove River Water pH And Conductivity Monitoring Per Maintenance Action EMA A41222
99-002	Revise Description Of Main Steam Line Low Pressure Switches Per Maintenance Actions EMAs A38872 Through A38875
99-003	Revise Description Of Turbine Building Ventilation Per Modification PMP0082
99-004	Reference Safety Evaluation 98-107 In UFSAR Chapter 15
99-006	Add Description Of SOLOMON Stability Monitoring Software
99-007	Revise Description Of Reactor Pressure Narrow Range Recorder
99-008	Remove Specific Details Of Excess Flow Check Valve Test Procedure
99-009	Revise Reactor Water Cleanup Isolation Valve Stroke Time
99-010	Revise 480 Volt Motor Control Center Description
99-011	Editorial Corrections To Design Drawings In The UFSAR
99-013	Delete Figure 7.7-5 As Redundant
99-014	Update UFSAR Figure 1.2-2 In Accordance With Design Drawing BECH-C109 Revision 4
99-015	Replace Figure 12.3-5 With Corresponding Design Drawing BECH-A040

99-016	Make Various Editorial Changes And Corrections
99-017	Clarifications To Summary Of Process Pipe Lines And Valves For Primary Containment Table
99-018	Revise Safety Relief Valve Setpoint Tolerance Per Amendment 228
99-021	Add Ericsson Wireless Telephone Description Per Modification 1611
99-022	Revise Description To Reflect Installation Of Additional Evacuation And Fire Alarm Switches Per Maintenance Action EMA A36625
99-023	Reflect Storage Of GE-12 Fuel In The Spent Fuel Pool Per Amendment 226
99-024	Remove Offgas Automatic Isolation On Hi-Hi-Hi Post-Treatment Radiation Levels Per Maintenance Action EMA 44812
99-025	Update References To Current LOCA Analysis And Add Insights From LOCA Analysis Sensitivity Studies
99-026	Revise Reactor Pressure Vessel Transient Design
99-027	Remove Head Spray Piping Description Per Modification DCP 1551
99-028	Revise Description Of Drywell And Torus Sample Valves Power Supplies
99-030	Revise UFSAR To Reflect That Cooling Water Flow To The Residual Heat Removal (RHR) Pump Seal Water Coolers Is Not Required To Support Any RHR Operating Mode
99-031	Revise UFSAR To Reflect Use Of SQUG Methodology To Verify Seismic Adequacy Of Instrument AC Equipment
99-032	Revise Equipment Maintenance Descriptions
99-033	Revise UFSAR To Reflect ECCS Strainer Replacement
99-035	Revise Fire Plan And Fire Protection
99-036	Revise UFSAR To Reflect Second Noble Metal Chemical Addition
99-037	Remove Offgas Post Treatment Isolation Per Maintenance Action EMA A44812
99-039	Fully Incorporate UFSAR Change Request 97-077, Core Spray Injection Isolation Valves
99-040	Revise Fire Pump Limiting Condition For Operation (LCO) Time, Diesel Fire Pump Surveillance Interval And NRC Reporting Requirements
99-041	Add Description Of Weld Overlay On Recirculation Inlet Nozzles Per Modification ECP 1627
99-043	Revise UFSAR Per GE-12 Fuel Upgrade Project
99-045	Remove Contractor Assembly Trailer, OC Refuel Support Trailer, Mechanical Engineering Trailer And Engineering Support Trailer From UFSAR Site Plan
00-001	Correct Typographical Errors
00-002	Revise CARDOX Inoperability Fire Watch Requirements
00-003	Replace Figure 6.2-8 With Design Drawing BECH-M325 Revision 3
00-004	Chapter 13 Organizational Changes
00-006	Correct Chapter 15 Reference Annotation
00-010	Correct References To Safety/Relief Valve Recorders
00-013	Reflect Alliant Utilities' Affiliation With Mid-Continent Area Power Pool (MAPP) And Mid-America Interconnected Network (MAIN) Grid Reliability Regions
00-014	Revise River Water Temperature Monitoring Requirements Per Amendment 223
00-015	Various Pagination And Editorial Changes
00-016	Revise Description Of Preparation Period Before Cedar River Flood Peak.

Per the guidance of NEI 98-03, Revision 1, the following changes constitute UFSAR deletions. A description of those UFSAR deletions and the basis for change is provided below.

99-008 Description: Removed specific details of excess flow check valve test procedure.

Basis: This change was made to be consistent with Inservice Testing Relief Request VR-24 and supporting NRC Safety Evaluation dated March 28, 2000.

99-030 Description: As a result of recent analysis, the UFSAR was revised to reflect that cooling water flow to the Residual Heat Removal (RHR) Pump Seal Water Coolers is not required to support any RHR operating mode.

Basis: A 10 CFR 50.59 Safety Evaluation was performed and determined no unreviewed safety question existed. This change also resulted in a revision to Technical Specification Bases section 3.7.3.

99-039 Description: Removed closure time and line size information for the Core Spray Injection Isolation valves.

Basis: An UFSAR Change Request and 10 CFR 50.59 Safety Evaluation were performed in 1997 to remove this information from the UFSAR Revision 14. However, this information remained in UFSAR Revision 14 in error. This change removed the information from UFSAR Revision 15.

Additional UFSAR deletions resulted from plant or program modifications which were evaluated under 10CFR50.59, License Amendments, or constituted removal of redundant information which remains located elsewhere in the UFSAR.

Attachment 2

to NG-00-0920

Revision 3 of DAEC TRM

Attachment 3

to NG-00-0920

Revision 15 of DAEC UFSAR

Attachment 4

to NG-00-0920

DAEC Technical Specification Bases

T 1.1 Definitions (continued)

	overlapping, or total channel steps so that the entire channel is tested.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all required logic components (i.e., all required relays and contacts, trip units, solid state logic elements, etc.) of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential,overlapping, or total system steps so that the entire logic system is tested.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Technical Specifications Table 1.1-1 with fuel in the reactor vessel.
OPERABLE – OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 1658 MWt.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

T 1.0 USE AND APPLICATION

T 1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Requirements Manual to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLES

The following examples illustrate the use of logical connectors.

(continued)

T 1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE T1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. TLCO not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

(continued)

T 1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE T1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. TLCO not met.	A.1 Trip . . . OR A.2.1 Verify . . . AND A.2.2.1 Reduce . . . OR A.2.2.2 Perform . . . OR A.3 Align . . .	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

T 1.0 USE AND APPLICATION

T 1.3 Completion Times

PURPOSE

The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.

BACKGROUND

TLCOs specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with a TLCO state Conditions that typically describe the ways in which the requirements of the TLCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Times(s).

DESCRIPTION

The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the TLCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the TLCO Applicability.

If situations are discovered that require entry into more than one Condition at a time within a single TLCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition.

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 unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

(continued)

T 1.3 Completion Times

DESCRIPTION (continued)

However, when a subsequent division, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

a. Must exist concurrent with the first inoperability; and

b. Must remain inoperable or not within limits after the first inoperability is resolved.

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or

b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extensions do not apply to those TLCOs that have exceptions that allow completely separate re-entry into the Condition (for each division, subsystem, component or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual TLCOs.

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ." Example T1.3-3 illustrates one use of this type of Completion Time. The 10 day Completion Time specified for Condition A and B in Example T1.3-3 may not be extended.

(continued)

T 1.3 Completion Times (continued)

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE T1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to be in MODE 3 within 12 hours AND in MODE 4 within 36 hours. A total of 12 hours is allowed for reaching MODE 3 and a total of 36 hours (not 48 hours) is allowed for reaching MODE 4 from the time that Condition B was entered. If MODE 3 is reached within 6 hours, the time allowed for reaching MODE 4 is the next 30 hours because the total time allowed for reaching MODE 4 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 4 is the next 36 hours.

(continued)

T 1.3 Completion Times

EXAMPLES (continued)

EXAMPLE T1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Conditions A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. TLCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after TLCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in TLCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not expired, TLCO 3.0.3 may be exited and operation continued in accordance with Condition A.

(continued)

T 1.3 Completion Times

EXAMPLES

EXAMPLE T1.3-2 (continued)

While in TLCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, TLCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.

(continued)

T 1.3 Completion Times

EXAMPLES (continued)

EXAMPLE T1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X subsystem inoperable.	A.1 Restore Function X subsystem to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the TLCO
B. One Function Y subsystem inoperable.	B.1 Restore Function Y subsystem to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the TLCO
C. One Function X subsystem inoperable. AND One Function Y subsystem inoperable.	C.1 Restore Function X subsystem to OPERABLE status. OR C.2 Restore Function Y subsystem to OPERABLE status.	72 hours 72 hours

(continued)

T 1.3 Completion Times

EXAMPLES

EXAMPLE T1.3-3 (continued)

When one Function X subsystem and one Function Y subsystem are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each subsystem, starting from the time each subsystem was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second subsystem was declared inoperable (i.e., the time the situation described in Condition C was discovered).

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected subsystem was declared inoperable (i.e., initial entry into Condition A).

The Completion Times of Conditions A and B are modified by a logical connector, with a separate 10 day Completion Time measured from the time it was discovered the TLCO was not met. In this example, without the separate Completion Time, it would be possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the TLCO. The separate Completion Time modified by the phrase "from discovery of failure to meet the TLCO" is designed to prevent indefinite continued operation while not meeting the TLCO. This Completion Time allows for an exception to the normal "time zero" for beginning the Completion Time "clock". In this instance, the Completion Time "time zero" is specified as commencing at the time the TLCO was initially not met, instead of at the time the associated Condition was entered.

(continued)

T 1.3 Completion Times

EXAMPLES (continued)

EXAMPLE T1.3-4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.

Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.

If the Completion Time of 4 hours (plus the extension) expires while one or more valves are still inoperable, Condition B is entered.

(continued)

T 1.3 Completion Times

EXAMPLES (continued)

EXAMPLE T1.3-5

ACTIONS

---------------------------NOTE---------------------------
Separate Condition entry is allowed for each inoperable valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.

(continued)

T 1.3 Completion Times

EXAMPLES

EXAMPLE T1.3-5 (continued)

If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.

Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.

EXAMPLE T1.3-6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Perform TSR 3.x.x.x.	Once per 8 hours
	OR	
	A.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

(continued)

T 1.3 Completion Times

EXAMPLES

EXAMPLE T1.3-6 (continued)

Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "once per" Completion Time, which qualifies for the 25% extension, per TSR 3.0.2, to each performance after the initial performance. The initial 8 hour interval of Required Action A.1 begins when Condition A is entered and the initial performance of Required Action A.1 must be complete within the first 8 hour interval. If Required Action A.1 is followed and the Required Action is not met within the Completion Time (plus the extension allowed by TSR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.

If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.

(continued)

T 1.3 Completion Times

EXAMPLES (continued)

EXAMPLE T1.3-7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated.	1 hour AND Once per 8 hours thereafter
	AND	
	A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours
	AND	
	B.2 Be in MODE 4.	36 hours

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by TSR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1

(continued)

T 1.3 Completion Times

EXAMPLES

EXAMPLE T1.3-7 (continued)

is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.

IMMEDIATE COMPLETION TIME

When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

T 1.0 USE AND APPLICATION

T 1.4 Frequency

PURPOSE

The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION

Each TRM Surveillance Requirement (TSR) has a specified Frequency in which the Surveillance must be met in order to meet the associated TLCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the TSR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section T 3.0, TSR Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each TSR, as well as certain Notes in the Surveillance column that modify performance requirements.

Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR T3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillance, or both. Example T1.4-4 discusses these special situations.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated TLCO is within its Applicability, represent potential TSR 3.0.4 conflicts. To avoid these conflicts, the TSR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With a TSR satisfied, TSR 3.0.4 imposes no restriction.

The use of "met" or "performed" in these instances conveys specific meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not "met." "Performance" refers only to the requirement to specifically determine the ability to meet the acceptance

(continued)

T 1.4 Frequency

DESCRIPTION (continued)

criteria. TSR 3.0.4 restrictions would not apply if both the following conditions are satisfied:

a. The Surveillance is not required to be performed; and

b. The Surveillance is not required to be met or, even if required to be met, is not known to be failed.

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the TLCO (TLCO not shown) is MODES 1, 2, and 3.

EXAMPLE T1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example T1.4-1 contains the type of TSR most often encountered in the TRM. The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the interval specified in the Frequency is allowed by TSR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the TSR is not required to be met per TSR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the TLCO). If the interval specified by TSR 3.0.2 is exceeded while the unit is in a MODE or other specified condition in the Applicability of the TLCO, and the performance of the Surveillance is not

(continued)

T 1.4 Frequency

EXAMPLES

EXAMPLE T1.4-1 (continued)

otherwise modified (refer to Examples T1.4-3 and T1.4-4), then TSR 3.0.3 becomes applicable.

If the interval as specified by TSR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the TLCO for which performance of the TSR is required, the Surveillance must be performed within the Frequency requirements of TSR 3.0.2 prior to entry into the MODE or other specified condition. Failure to do so would result in a violation of TSR 3.0.4.

EXAMPLE T1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter

Example T1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example T1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the extension allowed by TSR 3.0.2.

(continued)

T 1.4 Frequency

EXAMPLES

EXAMPLE T1.4-2 (continued)

"Thereafter" indicates future performances must be established per TSR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

EXAMPLE T1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
------------------NOTE------------------ Not required to be performed until 12 hours after ≥ 25% RTP. -- Perform channel adjustment.	7 days

The interval continues whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day interval (plus the extension allowed by TSR 3.0.2), but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of TSR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power ≥ 25% RTP.

(continued)

T 1.4 Frequency

EXAMPLES

EXAMPLE T1.4-3 (continued)

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of TSR 3.0.3 would apply.

EXAMPLE T1.4-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
------------------NOTE------------------ Only required to be met in MODE 1. -- Verify leakage rates are within limits.	24 hours

Example T1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example T1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by TSR 3.0.2), but the unit was not in MODE 1, there would be no failure of the TSR nor failure to meet the TLCO. Therefore, no violation of TSR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency were not met), TSR 3.0.4 would require satisfying the TSR.

T 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

TLCO 3.0.1

TLCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in TLCO 3.0.2.

TLCO 3.0.2

Upon discovery of a failure to meet an TLCO, the Required Actions of the associated Conditions shall be met, except as provided in TLCO 3.0.5 and TLCO 3.0.6.

If the TLCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

TLCO 3.0.3

When an TLCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the TLCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 2 within 9 hours;

b. MODE 3 within 13 hours; and

c. MODE 4 within 37 hours.

Exceptions to this TLCO are stated in the individual TLCOs.

Where corrective measures are completed that permit operation in accordance with the TLCO or ACTIONS, completion of the actions required by TLCO 3.0.3 is not required.

TLCO 3.0.3 is only applicable in MODES 1, 2, and 3.

TLCO 3.0.4

When an TLCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This TLCO shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

(continued)

3.0 TLCO APPLICABILITY

TLCO 3.0.4 (continued)

Exceptions to this TLCO are stated in the individual TLCOs. These exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered allow unit operation in the MODE or other specified condition in the Applicability only for a limited period of time.

TLCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, and 3.

TLCO 3.0.5

Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to TLCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

TLCO 3.0.6

When a TRM supported system TLCO is not met solely due to a TS support system LCO not being met, the TRM Conditions and Required Actions associated with this supported system are not required to be entered. Only the TS support system LCO ACTIONS are required to be entered. This is an exception to TLCO 3.0.2 for the TRM supported system.

When a support system's Required Action (either TS or TRM)directs a supported system in the TRM to be declared inoperable or directs entry into Conditions and Required Actions for a TRM supported system, the applicable TRM Conditions and Required Actions shall be entered in accordance with TLCO 3.0.2.

When a TS supported system LCO is not met solely due to a TRM support system LCO not being met, the TS supported system Conditions and Required Actions are required to be entered Immediately, under the definition of OPERABILITY, as neither TS LCO 3.0.6 or TRM TLCO 3.0.6 apply.

When an inoperable Technical Specification support system, structure, or component (SSC) provides support to a TRM SSC, which, in turn, supports a supported SSC addressed in the Technical Specifications, Technical Specification LCO 3.0.6 remains applicable.

3.0 SURVEILLANCE REQUIREMENT (TSR) APPLICABILITY

TSR 3.0.1

TSRs shall be met during the MODES or other specified conditions in the Applicability for individual TLCOs, unless otherwise stated in the TSR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the TLCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the TLCO except as provided in TSR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

TSR 3.0.2

The specified Frequency for each TSR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this TSR are stated in the individual TLCOs.

TSR 3.0.3

If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the TLCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance.

If the Surveillance is not performed within the delay period, the TLCO must immediately be declared not met, and the applicable Condition(s) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the TLCO must immediately be declared not met, and the applicable Condition(s) must be entered.

(continued)

3.0 TSR APPLICABILITY (continued)

TSR 3.0.4	Entry into a MODE or other specified condition in the Applicability of an TLCO shall not be made unless the TLCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. TSR 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, and 3.

T 3.2 LINEAR HEAT GENERATION RATE (LHGR)

TLCO 3.2 All LHGRs shall be less than or equal to the limits specified in the Core Operating Limits Report (COLR).

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any LHGR not within limits.	A.1 Restore LHGR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.2.1	Verify all LHGRs are less than or equal to the limits specified in the COLR.	24 hours

T 3.3 INSTRUMENTATION

T 3.3.1 Alternate Rod Insertion (ARI) Instrumentation

TLCO 3.3.1 Two channels in a trip system for each ARI instrumentation Function listed below shall be OPERABLE:

a. Reactor Vessel Water Level – Low Low; and

b. Reactor Steam Dome Pressure – High.

APPLICABILITY: MODE 1.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function with one or more channels inoperable.	A.1 Restore ARI trip capability.	72 hours
B. Both Functions with ARI trip capability not maintained.	B.1 Restore ARI trip capability for one Function.	1 hour
C. Required Action and associated Completion Time of Conditions A or B not met.	C.1 Be in MODE 2.	8 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.3.1.1	Perform CHANNEL CHECK on the Reactor Vessel Water Level - Low Low Function.	12 hours
TSR 3.3.1.2	Perform CHANNEL FUNCTIONAL TEST.	12 months
TSR 3.3.1.3	Perform CHANNEL CALIBRATION. The Allowable Values shall be: a. Reactor Vessel Water Level – Low Low ≥ 112.65 inches; and b. Reactor Steam Dome Pressure – High: ≤ 1154.2 psig.	12 months
TSR 3.3.1.4	Perform LOGIC SYSTEM FUNCTIONAL TEST, including scram air header venting.	24 months

T 3.3 INSTRUMENTATION

T 3.3.2 Control Rod Block Instrumentation

TLCO 3.3.2 The control rod block instrumentation for each Function in Table T3.3.2-1 shall be OPERABLE.

APPLICABILITY: According to Table T3.3.2-1.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function 1, 2, or 3 channel inoperable.	A.1 Restore channel to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met. OR Two or more Function 1, 2, or 3 channels inoperable.	B.1 Place one channel in trip. OR B.2 Initiate Reactor Manual Control System rod withdrawal block.	1 hour 1 hour
C. One or more Function 4 or 5 channels inoperable.	C.1 Place channel in trip. OR C.2 Initiate Reactor Manual Control System rod withdrawal block.	1 hour 1 hour

SURVEILLANCE REQUIREMENTS

------------------------------------NOTES------------------------------------

1. Refer to Table T3.3.2-1 to determine which TSRs apply for each control rod block instrumentation Function.

2. 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 (or the redundant Function for Functions 4.a and 4.b) maintains control rod block capability.

	SURVEILLANCE	FREQUENCY
TSR 3.3.2.1	Perform CHANNEL CHECK.	24 hours
TSR 3.3.2.2	--------------------NOTE-------------------- For Functions 1.b, 1.c, 2, and 3, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -- Perform CHANNEL FUNCTIONAL TEST.	7 days
TSR 3.3.2.3	--------------------NOTES-------------------- 1. For Functions 1.a, 1.d, and 5, not required to be performed until 24 hours after entering MODE 1, but prior to exceeding 25% RTP. 2. For Functions 5.a. and 5.c, this TSR may be met by verifying receipt of the appropriate alarm. -- Perform CHANNEL FUNCTIONAL TEST.	92 days
TSR 3.3.2.4	Calibrate the trip units.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
TSR 3.3.2.5	-------------------NOTES--------------- 1. For Function 1.b, not required to be performed until 12 hours after entering MODE 2 from MODE 1. 2. Neutron detectors are excluded. --------------------------------------- Perform CHANNEL CALIBRATION.	92 days
TSR 3.3.2.6	Perform CHANNEL CALIBRATION.	184 days
TSR 3.3.2.7	-------------------NOTES--------------- 1. For Functions 2.a, 2.b and 2.d, not required to be performed until 12 hours after entering MODE 2 from MODE 1. 2. For Function 3.a, not required to be performed until 12 hours after detector count rate is < 100 cps or the IRM channels are on range 2 or below. 3. For Function 3.b, not required to be performed until 12 hours after the IRM channels are on range 7 or below. 4. For Function 3.d, not required to be performed until 12 hours after the IRM channels are on range 2 or below. 5. Neutron detectors are excluded. --------------------------------------- Perform CHANNEL CALIBRATION.	24 months
TSR 3.3.2.8	Verify that the SRM Detector Not Full In rod block is not bypassed when SRM count rate is < 73 cps.	24 months

Table T3.3.2-1 (page 1 of 2)
Control Rod Block Instrumentation

FUNCTION	Applicable MODES or other specified conditions	Minimum number of OPERABLE instrument channels	Surveillance Requirements	Allowable Values
1. APRM				
a. Flow-Biased Upscale	1	4	TSR 3.3.2.1 TSR 3.3.2.3 TSR 3.3.2.5	% Recirc. Flow / % RTP[a] 0 / ≤ 53.5 25 / ≤ 68.5 50 / ≤ 85.5 75 / ≤ 100.4 100 / ≤ 115.0
b. Upscale in Startup	2	4	TSR 3.3.2.1 TSR 3.3.2.2 TSR 3.3.2.5	≤ 13.4%
c. Inoperative	1	4	TSR 3.3.2.3	NA
	2	4	TSR 3.3.2.2	NA
d. Downscale	1	4	TSR 3.3.2.1 TSR 3.3.2.3 TSR 3.3.2.5	≥ 5%
2. IRM				
a. Detector Not Full In	2, 5	4	TSR 3.3.2.2 TSR 3.3.2.7	NA
b. Upscale	2, 5	4	TSR 3.3.2.1 TSR 3.3.2.2 TSR 3.3.2.7	≤ 116.7/125 of full scale
c. Inoperative	2, 5	4	TSR 3.3.2.2	NA
d. Downscale	2[b], 5	4	TSR 3.3.2.1 TSR 3.3.2.2 TSR 3.3.2.7	≥ 5/125 of full scale
3. SRM				
a. Detector Not Full In	2[c]	3	TSR 3.3.2.2 TSR 3.3.2.7 TSR 3.3.2.8	NA
	5	2	TSR 3.3.2.2 TSR 3.3.2.7 TSR 3.3.2.8	NA

(continued)

a) When reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating," the following Allowable Values apply:

% Recirc. Flow	% RTP
0	≤ 50.0
25	≤ 65.4
50	≤ 81.0
75	NA
100	NA

The trip setpoints may be reset by adjusting APRM gain or by recalibrating the APRMs.

b) Bypassed when the IRM channels are on range 1.

c) Bypassed if detector count rate is > 100 cps or the IRM channels are on range 3 or higher.

Table T3.3.2-1 (page 2 of 2)
Control Rod Block Instrumentation

FUNCTION	Applicable MODES or other specified conditions	Minimum number of OPERABLE instrument channels	Surveillance Requirements	Allowable Values
3. SRM (continued)				
b. Upscale	2(d)	3	TSR 3.3.2.1 TSR 3.3.2.2 TSR 3.3.2.7	≤ 10^5 cps
	5	2	TSR 3.3.2.1 TSR 3.3.2.2 TSR 3.3.2.7	≤ 10^5 cps
c. Inoperative	2(d)	3	TSR 3.3.2.2	NA
	5	2	TSR 3.3.2.2	NA
d. Downscale	2(e)	3	TSR 3.3.2.1 TSR 3.3.2.2 TSR 3.3.2.7	≥ 3 cps
	5	2	TSR 3.3.2.1 TSR 3.3.2.2 TSR 3.3.2.7	≥ 3 cps
4. Scram Discharge Volume Water Level - High				
a. Resistance Temperature Detector	1, 2, 5(f)	1	TSR 3.3.2.3 TSR 3.3.2.4 TSR 3.3.2.7	≤ 765 ft - 1.7 inches
b. Float Switch	1, 2, 5(f)	1	TSR 3.3.2.3 TSR 3.3.2.7	≤ 765 ft - 4.8 inches
5. Recirculation Flow				
a. Upscale	1	2	TSR 3.3.2.3 TSR 3.3.2.6	≤ 110%
b. Inoperative	1	2	TSR 3.3.2.3	NA
c. Comparator	1	2	TSR 3.3.2.3 TSR 3.3.2.6	≤ 10% flow deviation

d) Bypassed when the associated IRM channels are on range 8 or higher.
e) Bypassed when the IRM channels are on range 3 or higher.
f) With any control rod withdrawn from a core cell containing one or more fuel assemblies, except control rods withdrawn under the provisions of Technical Specification 3.10.5 or 3.10.6.

T 3.3 INSTRUMENTATION

T 3.3.3 Non-Type A, Non-Category 1 Post Accident Monitoring (PAM) Instrumentation

TLCO 3.3.3 The PAM Instrumentation for each Function in Table T3.3.3-1 shall be OPERABLE.

APPLICABILITY: According to Table T3.3.3-1.

ACTIONS

-----NOTES-----

1. TLCO 3.0.4 is not applicable.

2. Separate Condition entry is allowed for each Function. For Functions 1 and 2, separate Condition entry is allowed for each SV/SRV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Function 1 channels inoperable.	A.1 Verify Function 2 instruments for the affected SV/SRV are OPERABLE.	Immediately
	<u>AND</u>	
	A.2 Monitor suppression pool temperature.	Once per 12 hours
	<u>AND</u>	
	A.3 Restore channel to OPERABLE status.	30 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	12 hours
C. One or more Function 2 or 6 channels inoperable.	C.1	Initiate action to restore channels to OPERABLE status.	Immediately
D. One required Function 3, 4, or 5 channel inoperable.	D.1	Initiate the preplanned alternate monitoring method.	72 Hours
	<u>AND</u>		
	D.2	Restore channel to OPERABLE status.	7 days
E. Required Action and associated Completion Time of Condition D not met.	E.1	Submit a Special Report to the NRC outlining the preplanned alternate monitoring method, the cause of the inoperability, and plans and schedule for restoring the system to OPERABLE status.	14 days

SURVEILLANCE REQUIREMENTS

-----------------------------------NOTE-----------------------------------
Refer to Table T3.3.3-1 to determine which TSRs apply for each PAM instrumentation Function.

SURVEILLANCE		FREQUENCY
TSR 3.3.3.1	Perform CHANNEL CHECK.	7 days
TSR 3.3.3.2	Perform CHANNEL CHECK.	31 days
TSR 3.3.3.3	Perform CHANNEL FUNCTIONAL TEST for portion of the channel outside primary containment.	92 days
TSR 3.3.3.4	Perform CHANNEL CALIBRATION	12 months
TSR 3.3.3.5	Perform CHANNEL CALIBRATION	18 months
TSR 3.3.3.6	Perform CHANNEL CALIBRATION	24 months

Table T3.3.3-1 (page 1 of 1)
Non-Type A, Non-Category 1 PAM Instrumentation

FUNCTION	Applicable MODES or other specified conditions	Minimum number of OPERABLE instrument channels	Surveillance Requirements
1. Safety/Safety Relief Valve Position Indicator (Primary Detector)	1, 2	2/valve	TSR 3.3.3.2 TSR 3.3.3.3 TSR 3.3.3.6
2. Safety/Safety Relief Valve Position Indicator (Backup Thermocouple)	1, 2	1/valve	TSR 3.3.3.2 TSR 3.3.3.5
3. Reactor Building Exhaust Stack Extended Range Effluent Radiation Monitor	1, 2	1	TSR 3.3.3.1 TSR 3.3.3.4
4. Turbine Building Exhaust Stack Extended Range Effluent Radiation Monitor	1, 2	1	TSR 3.3.3.1 TSR 3.3.3.4
5. Offgas Stack Extended Range Effluent Radiation Monitor	1, 2	1	TSR 3.3.3.1 TSR 3.3.3.4
6. Containment Water Level Monitor	1, 2	2	TSR 3.3.3.2 TSR 3.3.3.5

T 3.3 INSTRUMENTATION

T 3.3.4 Reactor Coolant System (RCS) Conductivity Monitoring Instrumentation

TLCO 3.3.4 One RCS conductivity monitor shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS conductivity monitor inoperable in MODES 1, 2, and 3.	A.1 Install temporary in-line conductivity monitor. AND A.2 Obtain in-line conductivity measurement.	4 hours Once per 4 hours
B. Required Action and associated Completion Time for Condition A not met.	B.1 Obtain and analyze an RCS sample for conductivity.	Immediately AND Once per 4 hours thereafter
C. Required Action and associated Completion Time for Condition B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. RCS conductivity monitor inoperable in other than MODES 1, 2, and 3.	D.1 Install temporary in-line conductivity monitor. AND D.2 Obtain in-line conductivity measurement.	4 hours Once per 24 hours
E. Required Action and associated Completion Time for Condition D not met.	E.1 Obtain and analyze an RCS sample for conductivity.	Immediately AND Once per 24 hours thereafter
F. Required Action and associated Completion Time for Condition E not met.	F.1 Declare reactor coolant conductivity not within the limit of Table T3.4.1-1.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
TSR 3.3.4.1 Perform CHANNEL CHECK of the continuous conductivity monitor.	7 days

T 3.3 INSTRUMENTATION

T 3.3.6 Surveillance Instrumentation

TLCO 3.3.6 The Surveillance Instrumentation for each Function in Table T3.3.6-1 shall be OPERABLE.

APPLICABILITY: According to Table T3.3.6-1.

ACTIONS

-----------------------------------NOTES-----------------------------------

1. TLCO 3.0.4 is not applicable.

2. Separate Condition entry is allowed for each Function. For Function 7, Separate Condition entry is allowed for each IRM and APRM trip system.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1 Restore required channel(s) to OPERABLE status.	30 days
B. One or more Functions with two required channels inoperable.	B.1 Restore one required channel per Function to OPERABLE status.	7 days
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 4.	24 hours

SURVEILLANCE REQUIREMENTS

-----------------------------------NOTE-----------------------------------
Refer to Table T3.3.6-1 to determine which TSRs apply for each surveillance instrumentation Function.

SURVEILLANCE		FREQUENCY
TSR 3.3.6.1	Perform CHANNEL CHECK.	12 hours
TSR 3.3.6.2	-------------------NOTE------------------ Not required to be performed until 12 hours after THERMAL POWER $\geq$ 25% RTP. --------------------------------------- Verify the absolute difference between the Average Power Range Monitor (APRM) channels and the calculated power is $\leq$ 2% RTP plus any gain adjustment required by LCO 3.4.1, "Recirculation Loops Operating." while operating at $\geq$ 25% RTP.	24 hours
TSR 3.3.6.3	Perform CHANNEL CALIBRATION.	184 days

Table T3.3.6-1 (page 1 of 1)
Surveillance Instrumentation

FUNCTION	Type/Range	Applicable MODES or other specified conditions	Minimum number of OPERABLE instrument channels	Surveillance Requirements
1. Reactor Water Level Indication	Recorder, Indicator 158 - 218 inches	1, 2, 3	2	TSR 3.3.6.1 TSR 3.3.6.3
2. Reactor Pressure Indication	Recorder, Indicator 0 - 1200 psig	1, 2, 3	2	TSR 3.3.6.1 TSR 3.3.6.3
3. Drywell Pressure Indication	Recorder -10 to +90 psig	1, 2, 3	2	TSR 3.3.6.1 TSR 3.3.6.3
4. Drywell Temperature Indication	Recorder 0 - 350°F	1, 2, 3	2	TSR 3.3.6.1 TSR 3.3.6.3
5. Suppression Pool Water Temperature Indication	Recorder 20 - 220°F	1, 2, 3	2	TSR 3.3.6.1 TSR 3.3.6.3
6. Suppression Pool Water Level Indication	Recorder -10 to +10 inches water	1, 2, 3	2	TSR 3.3.6.1 TSR 3.3.6.3
7. IRM/APRM Indication	0 to 125 %	1(a), 2	2/trip system	TSR 3.3.6.1(b) TSR 3.3.6.2(b)

(a) IRMs not required to be OPERABLE in MODE 1.
(b) Applicable to APRMs only.

T 3.3 INSTRUMENTATION

T 3.3.7 Explosive Gas Monitoring Instrumentation

TLCO 3.3.7 Two Offgas Hydrogen Monitoring instrumentation channels shall be OPERABLE.

APPLICABILITY: During Offgas System operation.

ACTIONS

-----------------------------------NOTES-----------------------------------

1. TLCO 3.0.4 is not applicable.

2. Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Offgas Hydrogen Monitoring instrumentation channel inoperable.	A.1 Verify required Recombiner Temperature Sensors OPERABLE.	Immediately
	<u>AND</u>	
	A.2 Restore channel to OPERABLE status.	30 days

(continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Two Offgas Hydrogen Monitoring instrumentation channels inoperable. <u>OR</u> All Recombiner Temperature Sensors inoperable on the in-service recombiner(s).	B.1.1	Verify at least one Recombiner Temperature Sensor functional on the in-service recombiner(s).	Immediately
	<u>OR</u>		
	B.1.2	Verify at least one offgas hydrogen monitoring instrumenation channel OPERABLE.	Immediately
	<u>AND</u>		
	B.2.1	Collect offgas gas sample.	Once per 24 hours
	<u>AND</u>		
	B.2.2	Analyze gas sample for hydrogen.	4 hours from sample collection
C. Required Action and associated Completion Time for Conditions A, or B not met.	C.1	Submit a Special Report to the NRC outlining the cause of the inoperablility, reasons why the instrument(s) were not made OPERABLE in a timely manner.	30 days

SURVEILLANCE REQUIREMENTS

-----------------------------------NOTE-----------------------------------
When a channel is placed in an inoperable status solely for performance of required TRM Surveillance Requirements, entry into associated Conditions and Required Actions may be delayed for up to 6 hours.

SURVEILLANCE		FREQUENCY
TSR 3.3.7.1	Perform CHANNEL CHECK.	24 hours
TSR 3.3.7.2	Perform CHANNEL FUNCTIONAL TEST.	31 days
TSR 3.3.7.3	Perform CHANNEL CALIBRATION.	92 days

T 3.4 REACTOR COOLANT SYSTEM (RCS)

T 3.4.1 RCS Chemistry

TLCO 3.4.1 The chemistry of the RCS shall be maintained within the limits of Table T3.4.1-1.

APPLICABILITY: According to Table T3.4.1-1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Any chemistry parameter not within limits of Table T3.4.1-1 in MODE 1.	A.1	Initiate action to restore chemistry parameter(s) to within limits.	Immediately
	AND		
	A.2	--------NOTE--------- Only applicable when reactor coolant conductivity is greater than limit of Table T3.4.1-1. -------------------- Obtain and analyze an RCS sample for chloride concentration and pH.	Once per 8 hours
B. Any chemistry parameter not within limits of Table T3.4.1-1 in MODE 1 for 72 hours continuously or 720 hours cumulative in previous 365 days.	B.1	Be in MODE 2.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Any chemistry parameter not within limits of Table T3.4.1-1 in MODES 2 and 3.	C.1	--------NOTE--------- Only applicable when reactor coolant conductivity is greater than limit of Table T3.4.1-1. ---------------------	
		Obtain and analyze an RCS sample for chloride concentration and pH.	Once per 8 hours
	AND		
	C.2	Restore RCS chemistry to within limits.	48 hours
D. Required Action and associated Completion Time of Condition C not met. OR Conductivity > 10.0 µmho at 25°C in MODE 1. OR Chloride concentration > 500 ppb in MODE 1.	D.1	Be in MODE 3.	12 hours
	AND		
	D.2	Be in MODE 4.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Any chemistry parameter not within limits of Table T3.4.1-1 in MODES 4 and 5.	E.1	--------NOTE--------- Only applicable when reactor coolant conductivity is greater than limit of Table T3.4.1-1. ---------------------	
		Obtain and analyze an RCS sample for chloride concentration and pH.	Once per 8 hours
	<u>AND</u>		
	E.2	Restore chloride concentration to within limits.	24 hours
	<u>AND</u>		
	E.3	Restore conductivity and pH to within limits.	72 hours
F. Required Action and associated Completion Time of Condition E not met.	F.1	Determine RCS is acceptable for operation.	Prior to entering MODE 2 or MODE 3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.4.1.1	-----------------NOTE-------------------- Only required to be performed during Noble Metal Chemical Addition. -- Verify reactor coolant conductivity, pH and chloride concentration are within the limits of Table T3.4.1-1.	8 hours
TSR 3.4.1.2	Verify RCS chemistry limits for pH in Table T3.4.1-1 are met.	Once within 72 hours prior to entering MODES 2 or 3 from MODE 4
TSR 3.4.1.3	Verify reactor coolant conductivity and chloride concentration are within the limits of Table T3.4.1-1.	72 hours

Table T3.4.1-1 (page 1 of 1)
RCS Chemistry Limits

MODES	Chlorides	Conductivity µmhos/cm @ 25°C	pH
1	≤ 200 ppb	≤ 1.0	5.6 ≤ pH ≤ 8.6
2, 3	≤ 100 ppb	≤ 2.0[a]	5.6 ≤ pH ≤ 8.6[b]
4, 5	≤ 100 ppb	≤ 5.0	4.6 ≤ pH

(a) 10 µmhos/cm during Noble Metal Chemical Addition.

(b) 5.6 ≤ pH ≤ 9.6 during Noble Metal Chemical Addition.

T 3.5 DRYWELL SPRAY SYSTEM AND ES COMPARTMENT COOLING AND VENTILATION

T 3.5.1 Drywell Spray System

TLCO 3.5.1 Two Residual Heat Removal (RHR) drywell spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One RHR drywell spray subsystem inoperable.	A.1	Restore RHR drywell spray subsystem to OPERABLE status.	30 days
B. Both RHR drywell spray subsystems inoperable.	B.1	Restore at least one RHR drywell spray subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	C.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.5.1.1	Verify the drywell spray spargers are unobstructed by performing an air test on the drywell spray headers and nozzles.	60 months

T 3.5 DRYWELL SPRAY SYSTEM AND ES COMPARTMENT COOLING AND VENTILATION

T 3.5.2 Engineered Safeguards (ES) Compartment Cooling and Ventilation

TLCO 3.5.2 The following ECCS and RCIC unit coolers shall be OPERABLE:

a. One RCIC room unit cooler;

b. One HPCI room unit cooler; and

c. Two CS/RHR room unit coolers.

APPLICABILITY: When the associated pumps are required to be OPERABLE.

ACTIONS

------------------------------------NOTE------------------------------------

Separate condition entry is allowed for each unit cooler.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required unit room cooler inoperable.	A.1 Declare the associated pump(s) inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.5.2.1	-------------------NOTE------------------- Only required to be performed in conjunction with Surveillances of the associated pump. -- Verify forced air circulation by the required unit coolers.	92 days
TSR 3.5.2.2	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

T 3.7 PLANT SYSTEMS

T 3.7.1 River Level

TLCO 3.7.1 River level shall be < 757 feet.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. River level ≥ 757 feet.	A.1 Be in MODE 3. AND A.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.7.1.1	------------------NOTE------------------ Only required to be performed when river level is > 753 feet. --- Verify river level is within limits.	1 hour

T 3.7 PLANT SYSTEMS

T 3.7.2 Snubbers

TLCO 3.7.2 All safety-related snubbers shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.
MODES 4 and 5 when associated systems are required to be OPERABLE.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each snubber.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more snubbers unacceptable due to failure to meet TSR 3.7.2.2.	------------NOTE------------ The snubber(s) may be determined acceptable for establishing the next inspection interval by completion of Required Actions A.1.1 and A.1.2. ----------------------------	
	A.1.1 Clearly establish and remedy the cause of the rejection for that snubber and for other snubbers that may be generically susceptible.	72 hours
	<u>AND</u>	
	A.1.2 Perform TSR 3.7.2.3 in the as-found condition for that snubber.	72 hours
	<u>OR</u>	
	A.2.1 Justify continued operation with the unacceptable snubber(s).	72 hours
	<u>OR</u>	(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Declare unacceptable snubber(s) inoperable.	Immediately
B. ---------NOTE--------- Required Actions B.2, B.3, and B.4 shall be completed if this Condition is entered. --------------------- One or more snubbers inoperable.	B.1 Replace or restore snubber to OPERABLE status.	72 hours
	<u>AND</u>	
	B.2 Determine supported component is acceptable for continued operation.	72 hours
	<u>AND</u>	
	-----------NOTE--------- Only required to be performed for snubbers found inoperable during performance of TSR 3.7.2.3. ------------------------	
	B.3 Perform TSR 3.7.2.3. on an additional 5% sample of snubbers, based upon the type of snubber that failed.	Immediately
	<u>AND</u>	
		(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	-------------NOTE----------- Only required to be performed for snubbers that fail to lock-up or to move (frozen) during performance of TSR 3.7.2.3 ----------------------------	
	B.4.1 Determine cause of failure is not due to manufacturer or design deficency.	72 hours
	OR	
	B.4.2 Perform TSR 3.7.2.3 on all snubbers subject to the same defect.	Immediately
C. Required Action and associated Completion Time of Condiditon B not met.	C.1 Declare the supported system or subsystem inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.7.2.1	-------------------NOTE------------------- Only required to be performed if affected snubbers are attached to sections of systems that have experienced unexpected, potentially damaging transients, as determined from review of operational data or visual inspection of the systems. --	
	Perform a visual inspection of all snubbers attached to systems that have experienced unexpected, potentially damaging transients. In addition to the visual inspection acceptance criteria of TSR 3.7.2.2, freedom-of-motion of mechanical snubbers shall be verified by one of the following methods: a. Manually induced snubber movement; b. Evaluation of in-place snubber piston setting; or c. Stroking snubber through its full range of travel.	Once within 72 hours for accessible systems AND Once within 184 days for inaccessible systems

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
TSR 3.7.2.2	------------------NOTE------------------ The first inspection interval determined using Table T3.7.2-1 criteria shall be based on the previous inspection intervals established by the requirements in effect before Technical Specification Amendment 203 was issued. -- Perform visual inspection of required snubbers based on criteria for each category in Table T3.7.2-1 to verify: a. No indications of damage or impaired OPERABILITY; b. Attachments to foundations or supporting structures are secure; and c. Fasteners for the attachment of the snubber to component and snubber anchorage are secure.	In accordance with Table T3.7.2-1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
TSR 3.7.2.3	-------------------NOTE------------------ The representative sample selected for functional testing shall represent various configurations, operating environments, and range of sizes of snubbers. --- Perform in-place or bench functional test of a representative sample of 10% of each type (mechanical or hydraulic) of snubber in use. a. At least 25% of the snubbers in the representative sample shall include snubbers from the following three categories: 1. First snubber away from each reactor vessel nozzle; 2. Snubbers within 5 feet of heavy equipment (valve, pump, motor, turbine, etc.); and 3. Snubbers within 10 feet of the discharge from a safety relief valve. b. Hydraulic snubber functional test acceptance criteria shall be: 1. Activation (restraining action) is achieved within specified range of velocity or acceleration in both tension and compression; and 2. Snubber bleed or release rate is within the specified range in compression or tension. Snubbers specifically required not to displace under continuous load shall have this capability verified.	18 months (continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
TSR 3.7.2.3	(continued) c. Mechanical snubber functional test criteria shall be: 1. Drag force of any snubber in tension and compression is less than specified maximum drag force; and 2. Activation (restraining action) is achieved within specified range of velocity or acceleration in both tension and compression; and 3. Snubber release rate, where required, is within the specified range in compression or tension. Snubbers specifically required not to displace under continuous load shall have this capability verified.	
TSR 3.7.2.4	-------------------NOTE------------------- Test results may not be included for the resampling of TSR 3.7.2.3. -- Perform in-place or bench functional test of all snubbers which failed during the previous testing cycle. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. The functional test acceptance criteria are specified in TSR 3.7.2.3.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
TSR 3.7.2.5	---NOTE--- Not required to be performed if tested within the previous 24 months. --- Perform bench functional testing of repaired or replacement snubbers. The functional test acceptance cirteria are specified in TSR 3.7.2.3.	Prior to installation
TSR 3.7.2.6	Verify no snubber service life shall be exceeded in the next 24 month cycle by performance of a snubber service life record review.	18 months

Table T3.7.2-1 (Page 1 of 2)

Snubber Visual Inspection Intervals

NUMBER OF INOPERABLE SNUBBERS			
Population or Category (Notes 1 and 2)	Column A Extended Interval (Note 3)	Column B Repeat Interval (Note 4)	Column C Reduced Interval (Note 5)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

Note 1: The next visual inspection interval for a snubber population or category shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. All snubbers found connected to an inoperable common hydrolic fluid reservoir shall be counted as unacceptable for determining the next inspection interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use the next lower integer for the value of the limit for Columns A, B, or C if that integer includes a fractional value of unacceptable snubbers as determined by interpolation.

(continued)

Table T3.7.2-1 (Page 2 of 2)

Snubber Visual Inspection Intervals

Note 3: If the previous number of unacceptable snubbers is less than or equal to the number in Column A, the next inspection interval may be twice the previous interval, but not greater than 48 months.

Note 4: If the number of unacceptable snubbers is less than or equal to the number in Column B, but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of unacceptable snubbers is greater than or equal to the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C, but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.

T 3.7 PLANT SYSTEMS

T 3.7.3 Structural Integrity

TLCO 3.7.3 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Section XI of the ASME Boiler and Pressure Code and applicable Addenda as required by 10 CFR 50.55a.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
----------NOTE---------- Not applicable to component(s) that are isolated from service. ------------------------ A. Requirements of the LCO not met for Class 1 or Class 2 component in MODES 1, 2, and 3.	A.1 Be in MODE 3. AND A.2 Be in MODE 4	12 hours 36 hours
B. Requirements of the LCO for Class 1 or Class 2 component(s) not met in other than MODES 1, 2, and 3.	B.1 Restore structural integrity of component(s) to within limits. OR B.2 Isolate affected component(s).	Prior to entering MODE 2 or 3 from MODE 4 Prior to entering MODE 2 or 3 from MODE 4

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Requirements of the LCO not met for Class 3 component(s).	------------NOTE------------ TLCO 3.0.4 is not applicable ---------------------------- C.1 Initiate action to restore the structural integrity of component(s) to within limits.	Immediately
	<u>OR</u>	
	C.2 Initate action to isolate affected component(s).	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.7.3.1	Inservice inspection shall be performed in accordance with the requirements for ASME Class 1, 2, and 3 components contained in Section XI of the ASME Boiler and Pressure Code and applicable Addenda as required by 10 CFR 50.55a.	In accordance with the Inservice Inspection program
TSR 3.7.3.2	The inservice inspection program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods, personnel, and sample expansion included in this generic letter.	In accordance with NRC Generic Letter 88-01 as modified by NRC approved alternate measures

T 3.7 PLANT SYSTEMS

T 3.7.4 MECHANICAL VACUUM PUMP

TLCO 3.7.4 The mechanical vacuum pump shall be capable of being isolated and secured on a signal of high radioactivity in the steam lines.

APPLICABILITY: MODES 1, 2, and 3, with the mechanical vacuum pump in operation.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of TLCO not met.	A.1 Initiate action to isolate the mechanical vacuum pump.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
TSR 3.7.4.1 Verify the mechanical vacuum pump is secured and isolated on an actual or simulated automatic isolation signal.	24 months

T 3.7 PLANT SYSTEMS

T 3.7.5 POST ACCIDENT SAMPLING SYSTEM (PASS)

TLCO 3.7.5 One PASS flowpath and the corresponding capability to analyze samples from each of the following locations shall be OPERABLE:

a. Reactor Coolant;

b. Containment Atmosphere; and

c. Suppression Pool Water.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----NOTES-----

1. TLCO 3.0.4 is not applicable.

2. Separate Condition entry is allowed for each flowpath.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required flowpath inoperable.	A.1	Confirm a sample can be obtained within 24 hours of the time the decision is made to sample.	7 days
	AND		
	A.2	Restore sampling capability.	90 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Sample analysis capability inoperable.	B.1 Confirm that alternative sample analytical support services can be initiated within 24 hours of the time the decision is made to sample.	7 days
	<u>AND</u>	
	B.2 Restore sample analysis capability.	90 days
C. Required Action and associated Completion Time for Conditions A or B not met.	C.1 Be in MODE 3.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
TSR 3.7.5.1 Verify the ability to collect and analyze samples from each PASS flowpath.	12 months

T 3.8 ELECTRICAL POWER SYSTEMS

T 3.8.1 24 VDC Sources

TLCO 3.8.1 Both Division I and Division II 24 VDC electrical power subsystems shall be OPERABLE.

APPLICABILITY: When supported equipment is required to be OPERABLE.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more 24 VDC electrical power subsystems inoperable.	A.1	Declare associated supported equipment inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.8.1.1	Verify battery terminal voltage is ≥ 21.7 VDC on float charge.	7 days
TSR 3.8.1.2	Verify no visible corrosion at battery terminals and connectors. <u>OR</u> Verify battery connection resistance is ≤ 1.25E-4 ohms for the intercell connectors and ≤ 3E-4 ohms for the inter-tier, inter-rack, and terminal connections.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
TSR 3.8.1.3	Verify battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration that could degrade battery performance.	12 months
TSR 3.8.1.4	Remove visible corrosion and verify battery cell to cell and terminal connections are coated with anti-corrosion material.	12 months
TSR 3.8.1.5	Verify battery connection resistance is ≤ 1.25E-4 ohms for the intercell connectors and ≤ 3E-4 ohms for the inter-tier, inter-rack, and terminal connections.	12 months
TSR 3.8.1.6	Verify each required battery charger supplies ≥ 30.5 amps.	24 months
TSR 3.8.1.7	------------------NOTE-------------------- The performance discharge test in TSR 3.8.1.8 may be performed in lieu of the service test in TSR 3.8.1.7. -- Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
TSR 3.8.1.8	Verify battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test.	60 months <u>AND</u> 12 months when battery shows degradation or has reached 85% of expected life with capacity < 100% of manufacturer's rating <u>AND</u> 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating

T 3.8 ELECTRICAL POWER SYSTEMS

T 3.8.2 24 VDC Battery Parameters

TLCO 3.8.2 Battery cell parameters for the Division I and Division II 24 VDC batteries shall be within the limits of Table T3.8.2-1.

APPLICABILITY: When associated DC electrical power subsystems are required to be OPERABLE.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more batteries with one or more battery cell parameters not within Category A or B limits.	A.1	Verify pilot cell electrolyte level and float voltage meet Table T3.8.2-1 Category C limits.	1 hour
	<u>AND</u>		
	A.2	Verify parameters for required battery cells meet Table T3.8.2-1 Category C limits.	24 hours <u>AND</u> Once per 7 days thereafter
	<u>AND</u>		
	A.3	Restore parameters for required battery cells to Category A and B limits of Table T3.8.2-1.	31 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> One or more batteries with average electrolyte temperature of the representative cells not within limits. <u>OR</u> One or more batteries with one or more battery cell parameters for required battery cells not within Category C limits.	B.1 Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.8.2.1	Verify battery cell parameters meet Table T3.8.2-1 Category A limits.	7 days
TSR 3.8.2.2	Verify battery cell parameters meet Table T3.8.2-1 Category B limits.	92 days <u>AND</u> Once within 24 hours after battery discharge < 21 V <u>AND</u> Once within 24 hours after battery overcharge > 28 V
TSR 3.8.2.3	Verify average electrolyte temperature of representative cells is ≥ 65°F for each battery.	92 days

Table T3.8.2-1 (page 1 of 1)
Battery Cell Parameter Requirements

PARAMETER	CATEGORY A: LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: LIMITS FOR EACH CONNECTED CELL	CATEGORY C: LIMITS FOR EACH CONNECTED CELL
Electrolyte Level	> Minimum level indication mark, and ≤ ¼ inch above maximum level indication mark[(a)]	> Minimum level indication mark, and ≤ ¼ inch above maximum level indication mark[(a)]	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 V	≥ 2.13 V	> 2.07 V
Specific Gravity[(b)(c)(d)]	≥ 1.195	≥ 1.190 AND Average of all connected cells > 1.200	Not more than 0.020 below average of all connected cells AND Average of all connected cells ≥ 1.190

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum level during equalizing charges provided it is not overflowing.

(b) Corrected for electrolyte temperature and level. Level correction is not required, however, when on float charge and battery charging current is < 2 amps.

(c) A battery charging current of < 2 amps when on float charge is acceptable for meeting specific gravity limits following a battery recharge, for a maximum of 7 days. When charging current is used to satisfy specific gravity requirements, specific gravity of each connected cell shall be measured prior to expiration of the 7 day allowance.

(d) A battery charging current of < 2 amps when on float charge is acceptable for meeting specific gravity limits following water addition to the battery, for a maximum of 8 weeks. When charging current is used to satisfy specific gravity requirements, specific gravity of each connected cell shall be measured prior to expiration of the 8 week allowance.

T 3.8 ELECTRICAL POWER SYSTEMS

T 3.8.3 24 VDC Distribution System

TLCO 3.8.3 The Division I and Division II 24 VDC electrical distibution subsystems shall be OPERABLE.

APPLICABILITY: When supported equipment is required to be OPERABLE.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more 24 VDC electrical power distribution systems inoperable.	A.1 Declare required supported equipment inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
TSR 3.8.3.1 Verify correct breaker alignments and indicated power availability to required 24 VDC electrical power distribution systems.	7 days

T 3.8 ELECTRICAL POWER SYSTEMS

T 3.8.4 Battery Room Ventilation

TLCO 3.8.4 The Battery Room Ventilation System shall be OPERABLE and at least one exhaust fan shall be operating.

APPLICABILITY: When the associated 125 VDC, 24 VDC, or 250 VDC battery is required to be OPERABLE.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. No Battery Room Exhaust Fans operating.	A.1	Place one fan in operation.	1 hour
B. One required Battery Room Exhaust Fan inoperable.	B.1	Restore Battery Room Exhaust Fan to OPERABLE status.	7 days
C. Battery Room Ventilation System inoperable for reasons other than Condition B. OR Required Action and associated Completion Time for Conditions A or B not met.	C.1	Declare the affected battery(s) inoperable.	Immediately
	OR		
	C.2.1	Provide portable ventilation for the Battery Room.	4 hours
	AND		
	C.2.2	Verify Battery Room hydrogen concentration < 4% by volume.	Once per 24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.8.4.1	Verify at least one Battery Room Exhaust Fan is operating.	24 hours
TSR 3.8.4.2	Verify each required Battery Room Exhaust Fan operates.	31 days

T 3.9 RADIOACTIVE SOURCES

TLCO 3.9 Each sealed source containing radioactive material in excess of the quantities listed in 10 CFR 30.71, Schedule B, or containing > 0.1 μCi of radioactive material shall have < 0.005 μCi of removable contamination.

APPLICABILITY: At all times.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each source.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more sources with removable contamination ≥ 0.005 μCi.	A.1 Withdraw the source from use.	Immediately
	<u>AND</u>	
	A.2.1 Initiate action to decontaminate and repair the source.	Immediately
	<u>OR</u>	
	A.2.2 Initiate action to dispose of the source in accordance with NRC regulations.	Immediately

SURVEILLANCE REQUIREMENTS

---------------------------------------NOTES---------------------------------------

1. The test method shall be capable of detecting a minimum of 0.005 μCi of radioactive material on the source.

2. Tests for leakage and/or contamination shall be performed by the licensee or other persons specifically authorized by the NRC or an Agreement State.

	SURVEILLANCE	FREQUENCY
TSR 3.9.1	--------------------NOTES------------------ 1. Startup sources subject to core flux are excluded. 2. Sealed sources that are stored and not being used are excluded. --- Perform testing for leakage and/or contamination for each sealed source containing radioactive material in any form other than gas with a half life of greater than 30 days, excluding Hydrogen 3.	184 days
TSR 3.9.2	--------------------NOTE------------------- Not required to be performed if performed within the previous 184 days. --- Perform testing for leakage for each source that is stored and not in use.	Prior to use OR Prior to transfer to another licensee

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
TSR 3.9.3	Perform testing of sources transferred without a certificate indicating the test has been performed within 184 days prior to transfer.	Prior to use
TSR 3.9.4	Perform leak testing of startup sources.	Prior to being subjected to core flux AND Prior to and following any repair or maintenance

T 3.10 HYDROGEN CONCENTRATION

TLCO 3.10 The concentration of hydrogen in the Offgas System downstream of the recombiners shall be ≤ 4% by volume.

APPLICABILITY: MODES 1 and 2 during Offgas System operation.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Concentration of hydrogen downstream of recombiners not within limit.	A.1 Restore hydrogen concentration within limit.	48 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
TSR 3.10.1 Verify hydrogen concentration downstream of recombiners is within limit.	24 hours

TB 3.0 SURVEILLANCE REQUIREMENT (TSR) APPLICABILITY

BASES

TSRs

TSR 3.0.1 through TSR 3.0.4 establish the general requirements applicable to all TLCOs and apply at all times, unless otherwise stated.

TSR 3.0.1

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

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

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

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

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

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

(continued)

BASES

TSR 3.0.1 (continued)

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

TSR 3.0.2

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

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

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the TSRs. The exceptions to TSR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual TLCOs. The requirements of regulations take precedence over the TRM. Therefore, when a test interval is specified in the regulations, the test interval cannot be extended by the TRM, and the TSR includes a Note in the Frequency stating, "TSR 3.0.2 is not applicable."

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BASES

TSR 3.0.2 (continued)

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

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

TSR 3.0.3

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

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

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

(continued)

BASES

TSR 3.0.3 (continued)

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions or operational situations, is discovered not to have been performed when specified, TSR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

TSR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for TSRs is expected to be an infrequent occurrence. Use of the delay period established by TSR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

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

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

TSR 3.0.4

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

This TSR ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure proper operation of the unit.

The provisions of this TSR should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before

(continued)

BASES

TSR 3.0.4 (continued)

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

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

The provisions of TSR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of TSR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

The precise requirements for performance of TSRs are specified such that exceptions to TSR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the TSRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated TLCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the TLCO Applicability would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of TSRs annotation is found in Section T 1.4, Frequency.

(continued)

BASES

TSR 3.0.4 (continued)

TSR 3.0.4 is only applicable when entering MODE 3 from MODE 4, MODE 2 from MODE 3 or 4, or MODE 1 from MODE 2. Furthermore, TSR 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODE 1, 2, or 3. The requirements of TSR 3.0.4 do not apply in MODES 4 and 5, or in other specified conditions of the Applicability (unless in MODE 1, 2, or 3) because the ACTIONS of individual TLCOs sufficiently define the remedial measures to be taken.

TB 3.2 LINEAR HEAT GENERATION RATE (LHGR)

BASES

This Specification assures that the linear heat generation rate in any rod is less than the design linear heat generation rate and that the fuel cladding 1% plastic diametral strain linear heat generation rate is not exceeded during any abnormal operating transient if fuel pellet densification is postulated. The LHGR as a function of core height shall be checked daily during reactor operation at ≥ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the Maximum Total Peaking Factor (MTPF) would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern. Values for LHGR are contained in the Core Operating Limits Report (COLR).

TB 3.3 INSTRUMENTATION

TB 3.3.1 Alternate Rod Insertion (ARI) Instrumentation

BASES

On July 26, 1984, the NRC published their final rule on Anticipated Transients Without Scram (ATWS) (10 CFR 50.62). This rule requires all BWRs to make certain plant modifications to mitigate the consequences of the unlikely occurrence of a failure to scram during an anticipated operational transient. The bases for these modifications are described in Reference 2. The ATWS-ARI logic shares its instrumentation with the ATWS-RPT logic and is a two-out-of-two once logic. There are two redundant systems, only one of which is required to be OPERABLE. The ARI logic initiates the ARI System, which actuates solenoid valves that bleed the air off the scram header, causing the control rods to insert. The instrument setpoints are chosen such that the normal Reactor Protection System scram setpoints for reactor high pressure or low water level will be exceeded before the ARI setpoints are reached. Because ATWS is considered a very low probability event and is outside the normal design basis for the plant, the Surveillance Frequencies and TLCO requirements are less stringent than for safety-related instrumentation.

A Function is considered to maintaining ATWS-ARI trip capability when both channels in a trip system are OPERABLE or in trip. This requires two channels of the Function in the same trip system to each be OPERABLE or in trip.

<u>References for Bases Section TB 3.3.1</u>

1. 10 CFR 50.62.

2. NEDE-31096-P-A, "Anticipated Transients Without Scram; Response to NRC ATWS Rule, 10 CFR 50.62," December 1985.

TB 3.3 INSTRUMENTATION

TB 3.3.2 Control Rod Block Instrumentation

BASES

The control rod block Functions are provided to prevent excessive control rod withdrawal so that the MCPR does not decrease below the limit. The trip logic for this Function is 1 out of n; e.g., any trip on one of six APRMs, six IRMs, or four SRMs will result in a rod block. The minimum instrument channel requirements ensure sufficient instrumentation to ensure the single failure criterion is met.

The APRM rod block Function is flow-biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides a gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater that the limit.

The IRM rod block Function provides local as well as gross core protection. The scaling arrangement is such that the trip setting is less than a factor of 10 above the indicated level.

The SRM rod block Function provides SRM upscale, downscale, detector improper position, and inoperative signals to the reactor manual control system to block rod withdrawal under certain conditions. Any one SRM channel can initiate a rod block. Appropriate lights and annunciators are actuated to indicate the existence of these conditions.

A downscale indication on an APRM or IRM is an indication that the instrument has failed or the instrument is not sensitive enough. In either case, the instrument will not respond to changes in control rod motion and, thus, control rod motion is prevented. The downscale trips are set at 5% indicated onscale for APRMs and 5/125 full scale for IRMs.

Both of the scram discharge volume high level channels provide input to the "B" logic.

TB 3.3 INSTRUMENTATION

TB 3.3.3 Non-Type A. Non-Category 1 Post Accident Monitoring (PAM) Instrumentation

BASES

The primary purpose of the PAM instrumentation is to display plant variables that provide information required by the control room operators during accident situations 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. The PAM instrumentation supplements existing instrumentation that was designed to monitor primarily the normal operational ranges of these parameters.

Each of the Safety/Safety Relief Valve Position Indicator (primary detector) instrument channels is comprised of three instruments (pressure switches) which are arranged in a "two out of three" logic. When a channel of Safety/Safety Relief Valve Position Indication is inoperable. the suppression pool water temperature shall be monitored to observe any unexplained temperature increase which might be indicative of an open safety relief valve.

The Extended Range Effluent Radiation Monitor instrumentation channels consist of the indicator and the recorder. To be considered OPERABLE. the channel must be continuously recorded. These monitors shall be calibrated by means of a built-in check source or a known radioactive source. The requirements listed apply only to the high range monitors.

TB 3.3 INSTRUMENTATION

TB 3.3.4 Reactor Coolant System (RCS) Conductivity Monitoring Instrumentation

BASES

At DAEC, the conductivity of the reactor coolant is continuously monitored with the sample coming from the Reactor Water Cleanup System influent or the Reactor Recirculation System. Conductivity instrumentation will be checked every 7 days by instream measurements with an independent conductivity monitor to ensure accurate readings. If conductivity is within its normal range, chlorides and other impurities will also be in their normal ranges.

In the event that the reactor coolant conductivity cannot be continuously monitored, a temporary in-line conductivity monitor is to be installed in a location capable of monitoring the reactor coolant and the conductivity recorded periodically.

In the event that a temporary in-line conductivity monitor cannot be installed in a location capable of monitoring the reactor coolant, a reactor coolant grab sample is to be obtained and analyzed for conductivity periodically. Conductivity measurement of the reactor coolant grab sample will be conservative due to CO_2 absorption (i.e., RCS grab sample conductivity measurement ≥ RCS in-line conductivity measurement), and the limits specified in Table T3.4.1-1 still apply.

TB 3.3 INSTRUMENTATION

TB 3.3.6 Surveillance Instrumentation

BASES

The Surveillance Instrumentation provides information to the operator for use in monitoring conditions within the reactor vessel and primary containment. For each parameter monitored, as listed in Table T3.3.6-1, there are at least two channels of instrumentation. By comparing between the two channels, a near continuous monitoring of instrument performance is available. Any deviation in readings will initiate an early recalibration, thereby maintaining the quality of the instrument readings. In addition, the IRMs and APRMs are considered to be redundant to each other, as well as to the SRMs, due to the overlapping ranges.

TB 3.3 INSTRUMENTATION

TB 3.3.7 Explosive Gas Monitoring Instrumentation

BASES

Offgas Hydrogen Monitors are provided to ensure that the concentration of potentially explosive gas in the Offgas System downstream of the recombiners is maintained below the flammability limit of a hydrogen and oxygen mixture in the system. Keeping the mixture below its flammability limit provides assurance that the integrity and OPERABILITY of the Offgas System is maintained and that the radioactive material concentration in the offgas will be controlled in conformance with 10 CFR 50, Appendix A, Criterion 60. Calibration gas concentrations will be within the range of interest for hydrogen concentration and will not include 0% or 100% hydrogen concentration.

The functioning of the catalyst in the recombiner and the recombiner itself can be checked by a temperature recorder, which records the temperatures given by three thermocouple type temperature elements inserted in the top, middle, and bottom of each recombiner. At least one of these temperature elements should be OPERABLE to monitor proper functioning of each in-service recombiner.

Refer to Offsite Dose Assessment Manual (ODAM) Figure 3-1 for the location of effluent monitoring point R2.

The quarterly CHANNEL CALIBRATION of the Offgas Hydrogen Monitors shall include the use of at least two standard gas samples, each containing a known volume percent hydrogen in the range of the instrument, with the balance of nitrogen.

<u>References for Bases Section TB 3.3.7</u>

1. 10 CFR 50, Appendix A.

2. Offsite Dose Assessment Manual.

TB 3.4 REACTOR COOLANT SYSTEM (RCS)

TB 3.4.1 RCS Chemistry

BASES

Materials in the primary system are primarily stainless steel and the Zircaloy cladding. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on conductivity and chloride concentrations. Conductivity is limited because it is continuously measured and gives an indication of abnormal conditions and the presence of unusual materials in the coolant. Chloride limits are specified to prevent stress corrosion cracking of stainless steel. According to test data, allowable chloride concentrations could be set several orders of magnitude above the established limit at the oxygen concentration (200-300 ppb) experienced during power operation without causing significant failures. Zircaloy does not exhibit similar stress corrosion failures. However, there are some conditions under which the dissolved oxygen content of the reactor coolant water could be higher than 200-300 ppb, such as during MODES 2, 3, 4, and 5. During these periods, a limit of 100 ppb has been established to ensure that permissible chloride-oxygen combinations are not exceeded. Boiling occurs at higher steaming rates causing deaeration of the reactor water, thus maintaining oxygen concentrations at low levels and ensuring that the chloride-oxygen content is not such as would tend to induce stress corrosion cracking.

When conductivity is in its proper range, pH, chloride, and other impurities affecting conductivity must also be within their normal range. When conductivity becomes abnormal, then chloride measurements are made to determine whether or not they are also out of their normal operating values. This would not necessarily be the case. Conductivity could be higher due to the presence of a neutral salt which would not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are in fact high due to purposeful addition of additives. In the case of Boiling Water Reactors (BWR), however, where minimal additives are used and neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Methods available to the operator for correcting the off-normal condition include operation of the Reactor Water Cleanup (RWCU) System, reducing the input of impurities, and placing the reactor in MODE 4. The major benefit of MODE 4 is to reduce the temperature dependent corrosion rates and provide time for the RWCU System to re-establish the purity of the reactor coolant. During some periods of operation, conductivity or chloride concentration may exceed 5.0 μmho/cm or 200 ppb respectively because of the initial evolution of gases, the initial addition of dissolved metals, or the breaking out of chlorides entrapped in

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BASES (continued)

the system. The total time during which the conductivity or chloride concentration may exceed the specified limit must be limited to prevent stress corrosion cracking.

The conductivity of the reactor coolant is continuously monitored. Conductivity instrumentation will be checked every 7 days by instream measurements with an independent conductivity monitor to ensure accurate readings. It is not required to verify the pH limits of Table T3.4.1-1 are met when conductivity is < 1 μmho/cm. This is because the instrumentation utilized does not have the capability to measure pH when conductivity is < 1 μmho/cm. If conductivity is within its normal range, chlorides and other impurities will also be in their normal ranges. The reactor coolant samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Isotopic analyses to determine major contributors to activity can be performed by a gamma scan.

Noble Metal Chemical Addition (NMCA) has been developed by General Electric Nuclear Energy (GENE) as a method to enhance the effectiveness of Hydrogen Water Chemistry (HWC) in mitigating Intergranular Stress Corrosion Cracking (IGSCC) in BWR vessel internal components. In HWC, hydrogen gas is injected into the reactor coolant to lower oxygen concentration levels, which in turn lowers conductivity in the coolant to below the threshold for IGSCC. Additionally, use of NMCA will allow lower injection rates of HWC which in turn reduces plant radiation exposure over the life of the plant. The NMCA process will deposit a very thin, discontinuous layer of noble metals onto all wetted surfaces during the injection process. The treated surfaces will behave catalytically and promote oxidant-hydrogen recombination. This results in low corrosion potential of components at low hydrogen injection rates. Higher reactor water conductivity and pH are anticipated during the application due to the effect of non-corrosive noble metals on the measured conductivity.

Required Action F.1 specifies that the RCS shall be determined acceptable for operation should RCS chemistry limits be violated and unable to be restored within the Completion Time. This determination shall include the performance of an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the RCS.

TB 3.5 DRYWELL SPRAY SYSTEM AND ES COMPARTMENT COOLING AND VENTILATION

TB 3.5.1 Drywell Spray System

BASES

The Drywell Spray System for DAEC consists of two subsystems, each with two RHR pumps. The water may be routed to any combination of two spray headers in the drywell. The functional requirement of this system is predicated upon the use of one subsystem. An OPERABLE Drywell Spray subsystem exists when one RHR pump, and the associated piping (including spargers), valves, instrumentation, and controls are OPERABLE. Thus, there are ample spares for margin above the design conditions.

While no credit is assumed in the UFSAR accident analyses for use of Drywell Spray, its use can be beneficial in mitigating certain events. Consequently, loss of margin should be avoided and the equipment maintained in a state of OPERABILITY, thus a 30 day Completion Time is chosen for one loop of drywell spray being inoperable.

References for Bases Section TB 3.5.1

1. UFSAR Section 6.2.1.3.3.

2. Safety Evaluation 99-33.

TB 3.5 DRYWELL SPRAY SYSTEM AND ES COMPARTMENT COOLING AND VENTILATION

TB 3.5.2 Engineered Safeguards (ES) Compartment Cooling and Ventilation

BASES

One unit cooler in each pump compartment is capable of providing adequate ventilation flow and cooling. Engineering analyses indicate that the temperature rise in safeguards compartments without adequate ventilation flow or cooling is such that continued operation of the safeguards equipment or associated auxiliary equipment cannot be assured. Thus, the Applicability is whenever the associated pump(s) is required to be OPERABLE, which for the CS and RHR pumps also includes when required in MODES 4 and 5.

TB 3.7 PLANT SYSTEMS

TB 3.7.1 River Level

BASES

The site natural grade in the vicinity of the plant varies from about elevation 746 feet to elevation 750 feet. The maximum flood of record at the site occurred in 1961 and rose to elevation 746.5 feet. The Standard Project Flood as determined by the U.S. Army Corps of Engineers would flood the site to elevation 754.5 feet. Consequently, the plant site finished grade is at elevation 757.0 feet. The computed Maximum Probable Flood would have a discharge of 316,000 cfs and would reach an elevation of 764.1 feet at the site. In addition, a possible wave height of 2.8 feet, including runup, was computed as caused by a sustained wind of 45 mph acting over a maximum fetch of 1.5 miles. Thus, the maximum calculated water surface elevation is 766.9 MSL. Consideration was given to the possibility of ice jams creating a higher flood level, but an inspection of valley topography reveals that at no point could ice create a flood wave approaching that of the Maximum Probable Flood. As a result of the above indications, all essential structures have been designed for flood protection to elevation 767 feet. Monitoring the river level hourly whenever it exceeds elevation 753.0 feet ensures that the approach to 757.0 feet will be monitored.

<u>References for Bases Section TB 3.7.1</u>

1. UFSAR Section 2.4.3.1.

TB 3.7 PLANT SYSTEMS

TB 3.7.2 Snubbers

BASES

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or other severe transient, while accommodating normal; thermal motion during system startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of damage to piping as a result of a seismic or other event initiating dynamic loads or, in the case of a frozen snubber, exceeding allowable stress limits during system thermal transients. It is therefore required that all safety-related snubbers be OPERABLE during MODES 1, 2, and 3; and during MODES 4 and 5 when associated systems are required to be OPERABLE.

The TRM imposes SURVEILLANCE REQUIREMENTS for both visual inspections and functional testing of all safety-related snubbers. A visual inspection is the observation of the condition of installed snubbers to identify those that are damaged, degraded, or inoperable as caused by physical means, leakage, corrosion, or environmental exposure. The performance of visual examinations is a separate process that complements the functional testing program and provides additional confidence in snubber OPERABILITY.

Previously, a schedule was specified for snubber visual inspections that was based on the number of inoperable snubbers found during the previous visual inspection. Because the previous schedule for snubber visual inspections was based only the number of inoperable snubbers found during the previous inspection, a large number of inoperable snubbers found resulted in the visual inspection schedule being excessively restrictive. This not only resulted in spending a significant amount of resources, but also subjected plant personnel to unnecessary radiological exposure.

To alleviate this situation, the NRC developed an alternate schedule for visual inspections and issued it under Generic Letter 90-09, dated December 11, 1990. This alternate method maintains the same confidence level as the previous schedule and generally allows the performance of visual examinations and corrective action during plant outages.

The alternate inspection schedule is based on the number of unacceptable snubbers found during the previous inspection in proportion to the sizes of the various snubber populations or categories. A snubber is considered unacceptable if it fails the acceptance criteria of the visual inspection. This inspection interval is based on a fuel cycle and may be as long as two fuel cycles.

When the cause for rejection of a snubber during visual inspection is clearly established and remedied for that snubber, and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that

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BASES (continued)

snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to the cause of rejection of the snubber, or are similarly located or exposed to the same environmental conditions such as humidity, temperature, radiation, and vibration.

To verify that a snubber can operate within specific performance limits, a functional test is performed that typically involves removing the snubber and testing it on a specifically designed test stand. Functional testing provides a 95 percent confidence level that 90 to 100 percent of the snubbers operate within the specified acceptance limits.

To further increase the assurance of snubber reliability, functional tests will be performed once per operating cycle. These tests will include stroking of the snubbers to verify proper movement, restraining characteristics, and drag force (if applicable). Ten percent of the total of each type of snubber represents an adequate sample for such tests. Observed failures on these samples require testing of additional units.

The representative sample selected for functional testing shall represent the various configurations, operating environments, and range of sizes of snubbers. At least 25 percent of the snubbers in the representative sample shall include snubbers from the following three categories:

1. The first snubber away from each reactor vessel nozzle.
2. Snubbers within 5 feet of heavy equipment (valve, pump, turbine, etc.).
3. Snubbers within 10 feet of the discharge from a safety relief valve.

The 25 percent representative sample consists of those snubbers that meet the three categories above and have not been part of the last three representative samples.

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be a retested. Both the spare snubber and the repaired/reinstalled snubber shall not be included in the sample plan. Failure of this functional test shall determine whether or not the snubber mode of failure adversely affected the supported component or system.

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BASES (continued)

When a snubber is found inoperable, the subject snubbers are to be replaced or restored to OPERABLE status and an engineering evaluation performed. This evaluation is to determine the snubber mode of failure and identify any safety-related component or system that may have been adversely affected by the inoperability of the snubber in order to ensure that these components remain capable of meeting the designed service requirement. The engineering evaluation shall determine whether or not the snubber mode of failure adversely affected the supported component or system. In the event that the plant experiences a "potentially damaging transient," an inspection of the affected snubbers shall be performed. A "potentially damaging transient" is considered to be any event that causes physical damage to piping or component(s) that the snubber is supporting.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure snubbers periodically undergo a performance evaluation in view of age and operating conditions. The maximum expected service life for various seals, springs, and other critical parts shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. Due to implementation of the snubber service life monitoring program after several years of plant operation, the historical records to date may be incomplete. The records will be developed from engineering data available. If actual installation data is not available, the service life will be assumed to commence with the initial criticality of the plant. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

Removal of more than one snubber for a given system at a time is an acceptable practice. However, if all the snubbers are not restored within 72 hours, the system(s) supported by the snubbers must be declared inoperable.

For clarification, per Reference 1, an inoperable snubber is a properly fabricated, installed, and sized snubber which cannot pass its functional test. Simply removing a snubber for testing does not make the snubber inoperable. Therefore, engineering evaluations are not required simply for the purpose of removing a snubber for testing. However, the 72 hour Completion Time still applies since the snubber cannot perform its function. In summary, if a snubber is removed for testing, an engineering evaluation need not be performed. The snubber(s) must be returned to OPERABLE status within 72 hours, however, or the system(s) supported by the snubber(s) must be declared inoperable.

(continued)

BASES (continued)

References for Bases Section TB 3.7.2

1. NG-92-0969, Memo from R. Browning to Gary Middlesworth dated Feb. 25, 1992, "Technical Specification Clarification."

TB 3.7 PLANT SYSTEMS

TB 3.7.3 Structural Integrity

BASES

A pre-service inspection of Nuclear Class I components was conducted to ensure freedom from defects greater than code allowance; in addition, this served as a reference base for future inspections. Prior to operation, the Reactor Coolant System (RCS) as described in Article IS-120 of Section XI of the ASME Boiler and Pressure Vessel Code was inspected to provide assurance that the system was free of gross defects. In addition, the facility was designed such that gross defects should not occur throughout plant life. The pre-service inspection program was based on the 1970 Section XI of the ASME Code for in-service inspection. This inspection plan was designed to reveal problem areas (should they occur) before a leak in the coolant system could develop. The program was established to provide reasonable assurance that no Loss of Coolant Accident (LOCA) would occur at the DAEC as a result of leakage or breach of pressure-containing components and piping of the RCS, portions of the Emergency Core Cooling System (ECCS), and portions of the reactor coolant associated auxiliary systems.

A pre-service inspection was not performed on Nuclear Class II components since it was not required at that stage of DAEC construction when it would have been used. For these components, shop and in-plant examination records of components and welds will be used as a basis for comparison with in-service inspection data.

Visual examinations for leaks will be made periodically on ASME Section XI Class 1, 2, and 3 systems. The inspection program specified encompasses the major areas of the vessel and piping systems within the ASME Section XI boundaries.

The type of examination planned for each component depends on the location, accessibility, and type of potential defect. Direct visual examination is proposed wherever possible since it is fast and reliable. Surface examinations are planned where practical, and where added sensitivity is required. Ultrasonic examinations or radiography shall be used where defects can occur in concealed surfaces. Section 5.2.4 of the UFSAR provides details of the inservice inspection program.

Starting with the Cycle 9/10 Refueling Outage, an augmented inspection program was implemented to address concerns relating to Intergranular Stress Corrosion Cracking (IGSCC) in reactor coolant piping made of austenitic stainless steel. The augmented inspection program conforms to the NRC staff's positions set forth in Generic Letter 88-01 and NUREG 0313, Revision 2 for inspection schedule, inspection methods and personnel, and inspection sample expansion.

The first 10-year interval for inservice testing of pumps and valves in accordance with the ASME Code, Section XI commenced on February 1, 1975 and

(continued)

BASES

ended on January 31, 1985. The second 10-year inservice testing interval commenced on February 1, 1985 and ended on January 31, 1995. The third 10-year inservice testing interval commenced on February 1, 1995 and is scheduled to end on January 31, 2005. The second and third 10-year testing programs address the requirements of the ASME Code, Section XI, 1980 Edition with Addenda through Winter 1981, subject to the limitations and modifications of 10 CFR 50.55a. Section 3.9.6 of the UFSAR describes the inservice testing program.

Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

References for Bases Section TB 3.7.3

1. ASME Boiler and Pressure Vessel Code, Section XI, 1970 and 1980 Editions.

2. Generic Letter 88-01.

3. NUREG 0313, Revision 2.

4. UFSAR Section 3.1.2.2.6, Criterion 15, Rx Coolant System Design.

5. UFSAR Section 5.2, Integrity of RCPB.

6. UFSAR Section 3.8.4.5, Structural Acceptance Criteria.

TB 3.7 PLANT SYSTEMS

TB 3.7.4 Mechanical Vacuum Pump

BASES

The purpose of isolating the mechanical vacuum pump line is to limit the release of activity from the main condenser. During an accident, fission products could be transported from the reactor through the main steam lines to the condenser. The fission product radioactivity would be sensed by the main steam line radioactivity monitors which initiate isolation. High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure. In the event of gross fuel failure, the established setting for the main steam radiation monitors will trip the mechanical vacuum pump which in turn isolates the suction of the mechanical vacuum pump from the high and low pressure condensers. This prevents the release of untreated fission products to the environment via the mechanical vacuum pump.

TB 3.7 PLANT SYSTEMS

TB 3.7.5 Post Accident Sampling System (PASS)

BASES

The purpose of the PASS is to provide the means of obtaining representative samples from within primary containment for radiological and chemical analysis in association with a Loss of Coolant Accident. The PASS meets NUREG 0737, Section II.B.3 requirements regarding sampling capability. The PASS permits gathering of samples at any time during normal plant operations or during abnormal operations. The PASS allows personnel to obtain and analyze a sample without radiation exposures exceeding the criteria of 10 CFR 50, Appendix A, Criterion 19.

The Condition B requirement for alternative analytical support for containment hydrogen and oxygen may be satisfied by verifying that at least one of the H_2/O_2 in-line analyzers is on-line and capable of being lined up to each of the two sampling points (drywell and suppression chamber).

References for Bases Section TB 3.7.5

1. NUREG 0737. Section II.B.3.
2. 10 CFR 50, Appendix A.

TB 3.8 ELECTRICAL POWER SYSTEMS

TB 3.8.1 24 VDC Sources

BASES

The 24 VDC System provides power for reactor neutron monitoring and process radiation monitoring. The neutron monitoring function is fail-safe in that loss of 24 VDC power would cause the associated trip to occur. Two independent plus and minus 24 VDC sources are provided, each consists of a center tapped 48 V battery and two 24 V battery chargers that are fed from essential AC buses. Separation is provided for all equipment and feeders as in all safeguards systems.

Although the station batteries will deteriorate with time, utility experience indicates that there is almost no possibility of precipitous failure. The type of Surveillance Requirement described is one which has been demonstrated over the years to provide an indication of a cell becoming irregular or unserviceable long before it fails completely. The presence of physical damage or deterioration does not necessarily represent a failure of TSR 3.8.1.3, provided an evaluation determines that the physical damage or deterioration does not affect the OPERABILITY of the battery (its ability to perform its design function).

The service discharge test provides adequate indication of the batteries' ability to satisfy the design requirements (battery duty cycle) of the associated DC system. This test may be performed using simulated loads at the rates and for the durations specified in the design load profile, or, due to the constant nature of the normal system loads, using the connected load as the discharge source. Additionally, the specific gravity and voltage of each cell shall be determined after the discharge and recorded.

The performance discharge test provides adequate indication and assurance that the batteries have the specified amp hour capacity. The rate of discharge during this test shall be in accordance with the manufacturer's discharge characteristic curves. The results of these tests will be recorded and compared with the manufacturer's recommendations of acceptability.

References for Bases Section TB 3.8.1

1. UFSAR Section 8.3.2.

TB 3.8 ELECTRICAL POWER SYSTEMS

TB 3.8.2 24 VDC Battery Parameters

BASES

This LCO delineates the limits on electrolyte temperature, level, float voltage, and specific gravity for the 24 VDC electrical power subsystems batteries. Battery cell parameters must remain within acceptable limits to ensure availability of the required 24 VDC power. The battery cell parameters are required solely for the support of the associated 24 VDC electrical power subsystem. Therefore, parameters for required battery cells are only required to be within limits when the associated 24 VDC electrical power subsystem is required to be OPERABLE. The limit for average electrolyte temperature is ≥ 65°F for each battery.

Table T3.8.2-1

This Table delineates the limits on electrolyte level, float voltage, and specific gravity for three different categories. The meaning of each category is discussed below.

Category A defines the normal parameter limit for each designated pilot cell in each battery. The cells selected as pilot cells are those whose temperature, voltage, and electrolyte specific gravity approximate the state of charge of the entire battery.

The Category A limits specified for electrolyte level are based on manufacturer's recommendations and are consistent with the guidance in IEEE-450, with the extra ¼ inch allowance above the high water level indication for operating margin to account for temperature and charge effects. In addition to this allowance, footnote (a) to Table T3.8.2-1 permits the electrolyte level to be temporarily above the specified maximum level during an equalizing charge, provided it is not overflowing. It is acknowledged that, following completion of an equalizing charge, electrolyte level may temporarily be above the specified limit, but not overflowing, for a short period of time. The recovery time for electrolyte level after the equalizing charge is considered part of the equalizing process. These limits ensure that the plates suffer no physical damage, and that adequate electron transfer capability is maintained in the event of transient conditions. IEEE-450 recommends that electrolyte level readings should be made only after the battery has been at float charge for at least 72 hours.

The Category A limit specified for float voltage is ≥ 2.13 V per cell. This value is based on the recommendation of IEEE-450, which states that prolonged operation of cells below 2.13 V can reduce the life expectancy of cells. The Category A limit specified for specific gravity for each pilot cell is ≥ 1.195 (0.015 below the manufacturer's fully charged nominal specific gravity or a battery charging current that had stabilized at a low value). This value is characteristic of a charged cell with adequate capacity. According to

(continued)

BASES

IEEE-450, the specific gravity readings are based on a temperature of 77°F (25°C). The specific gravity readings are corrected for actual electrolyte temperature and level. For each 3°F (1.67°C) above 77°F (25°C), 1 point (0.001) is added to the reading; 1 point is subtracted for each 3°F below 77°F. The specific gravity of the electrolyte in a cell increases with a loss of water due to electrolysis or evaporation. Level correction will be in accordance with manufacturer's recommendations.

Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out.

The Category B limits specified for electrolyte level and float voltage are the same as those specified for Category A and have been discussed above. The Category B limit specified for specific gravity for each connected cell is ≥ 1.190 (0.020 below the manufacturer's fully charged, nominal specific gravity) with the average of all connected cells 1.200 (0.010 below the manufacturer's fully charged, nominal specific gravity). These values are based on manufacturer's recommendations. The minimum specific gravity value required for each cell ensures that the effects of a highly charged or newly installed cell do not mask overall degradation of the battery.

Category C defines the limits for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category C limits, the assurance of sufficient capacity described above no longer exists, and the battery must be declared inoperable.

The Category C limit specified for electrolyte level (above the top of the plates and not overflowing) ensures that the plates suffer no physical damage and maintain adequate electron transfer capability. The Category C Limit for voltage is based on IEEE-450, which states that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement.

The Category C limit on average specific gravity ≥ 1.190, is based on manufacturer's recommendations (0.020 below the manufacturer's recommended fully charged, nominal specific gravity). In addition to that limit, it is required that the specific gravity for each connected cell must be no less than 0.020 below the average of all connected cells. This limit ensures that the effect of a highly charged or new cell does not mask overall degradation of the battery.

The footnotes to Table T3.8.2-1 that apply to specific gravity are applicable to Category A, B, and C specific gravity.

Because of specific gravity gradients that are produced during the recharging

(continued)

BASES (continued)

process, delays of several days may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge of the designated pilot cell. This phenomenon is discussed in IEEE-450. Footnote (c) to Table T3.8.2-1 allows the float charge current to be used as an alternate to specific gravity for up to 7 days following a battery recharge. Within 7 days, each connected cell's specific gravity must be measured to confirm the state of charge. Following a minor battery recharge (such as equalizing charge that does not follow a deep discharge) specific gravity gradients are not significant, and confirming measurements may be made in less than 7 days.

Because of specific gravity gradients that are produced during the addition of water to battery cells, delays of several weeks may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of the batteries per IEEE Std 450-1980. Footnote (d) to Table T3.8.2-1 allows the float charge current to be used as an alternate to specific gravity for up to 8 weeks following a water addition. Within 8 weeks, each connected cell's specific gravity must be measured to confirm the state of charge.

TB 3.8 ELECTRICAL POWER SYSTEMS

TB 3.8.3 24 VDC Distribution System

BASES

The 24 VDC System provides power for reactor neutron monitoring and process radiation monitoring. The neutron monitoring function is fail-safe in that loss of 24 VDC power would cause the associated trip to occur. Two independent plus and minus 24 VDC distribution subsystems are provided, each supplied by a center tapped 48 V battery and two 24 V battery chargers that are fed from essential AC buses. The distribution subsystems are independent, with each having its own distribution panel. Separation is provided for all equipment and feeders as in all safeguards systems.

References for Bases Section TB 3.8.3

1. UFSAR Section 8.3.2.

TB 3.8 ELECTRICAL POWER SYSTEMS

TB 3.8.4 Battery Room Ventilation

BASES

The Battery Room Ventilation System consists of three exhaust fans and associated ductwork to each of the essential battery rooms. TLCO 3.8.4 is met whenever at least two of the three exhaust fans is OPERABLE, with at least one fan in operation.

The Battery Room is ventilated to prevent accumulation of hydrogen gas exceeding 4 percent concentration. On loss of Battery Room Ventilation, the use of portable ventilation equipment and sampling every 24 hours provides assurance that potentially hazardous quantities of hydrogen gas will not accumulate.

TSR 3.8.4.1 requires a daily check to ensure at least one exhaust fan is operating. The 24 hour frequency takes into consideration that two exhaust fans are normally inservice and the availability of alarms in the control room.

Since normal battery room ventilation system operation uses two fans, the TSR 3.8.4.2 two-fan OPERABLE requirement is normally demonstrated without requiring any shifting of fans. The 31 day Frequency is consistent with the known reliability of the fan motors and controllers, and the redundancy available in the system.

TB 3.9 RADIOACTIVE SOURCES

BASES

Ingestion or inhalation of source material may give rise to total body or organ irradiation. TRM T 3.9 ensures that leakage from radioactive material sources does not exceed allowable limits. In the unlikely event that those quantities of radioactive by-product materials of interest to this TRM Section which are exempt from leakage testing are ingested or inhaled, they represent less than one maximum permissible body burden for total body irradiation. The limits for all other sources (including alpha emitters) are based upon 10 CFR 70.39(c) limits for plutonium.

References for Bases Section TB 3.9

1. 10 CFR 70.39(c).

2. 10 CFR 30.71.

TB 3.10 HYDROGEN CONCENTRATION

BASES

TRM T 3.10 is provided to ensure that the concentration of potentially explosive gas in the Offgas System downstream of the recombiners is maintained below the flammability limit of a hydrogen and oxygen mixture in the system. Keeping the mixture below its flammability limit provides assurance that the integrity and OPERABILITY of the Offgas System is maintained and that the radioactive material concentration in the offgas will be controlled in conformance with 10 CFR 50, Appendix A, Criterion 60.

<u>References for Bases Section TB 3.10</u>

1. 10 CFR 50, Appendix A.

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B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

UFSAR 3.1.2.2.1 (Ref. 1) and SLs ensure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients and Abnormal Operational Transients.

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish an SL, such that the MCPR is not less than the limit specified in Specification 2.1.1.2. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical barriers that separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses, which occur from reactor operation significantly above design conditions.

While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross, rather than incremental, cladding deterioration. Therefore, the fuel cladding SL is defined with a margin to the conditions that would produce onset of transition boiling (i.e., MCPR = 1.00). These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity SL ensures that during normal operation and during Abnormal Operational Transients, at least 99.9% of the fuel rods in the core do not experience transition boiling.

(continued)

BASES

BACKGROUND (continued)

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The reactor vessel water level SL ensures that adequate core cooling capability is maintained during all MODES of reactor operation. Establishment of ECCS initiation setpoints higher than the SL provides margin to the SL but is independent of the SL.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and Abnormal Operational Transients. The reactor core SLs are established to preclude violation of the fuel design criterion that a MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with the other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR limit.

2.1.1.1 Fuel Cladding Integrity

GE critical power correlations are applicable for all critical power calculations at pressures ≥ 785 psig and core flows ≥ 10% of rated flow. For operation at low pressures or low flows, another basis is used, as follows:

> Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.5 psi. Analyses (Ref. 2) show that with a bundle flow of 28×10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be > 28 x 10^3 lb/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 50% RTP. Thus, a THERMAL POWER limit of 25% RTP for reactor pressure < 785 psig is conservative.

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which transition boiling is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid transition boiling, considering the power distribution within the core and all uncertainties.

The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations. Details of the fuel cladding integrity SL calculation are given in Reference 2. Reference 2 also includes a tabulation of the uncertainties used in the determination of the MCPR SL and of the nominal values of the parameters used in the MCPR SL statistical analysis.

For SLO, the SLMCPR is greater to account for the increased uncertainties.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL is used to initiate the actions of 10 CFR 50.36(c)(1). This limit is not used as a reference point for any RPS or ECCS automatic actuation setpoints. These setpoints ensure automatic actuation of safety functions that maintain fuel cladding integrity even if the reactor vessel water level SL is violated. The reactor vessel water level SL has been established at 15 inches above the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate SLs for effective action.

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

(continued)

BASES (continued)

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 3). Therefore, it is required to fully insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal. When operating in MODES 1 or 2, it is expected that a manual scram would be required to insert all insertable control rods within the two hour time limit. When operating in MODES 3, 4, or 5, all insertable control rods could possibly be inserted manually within the two hour time limit.

REFERENCES

1. UFSAR 3.1.2.2.1.

2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved revision).

3. 10 CFR 100.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on reactor steam dome pressure protects the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity. According to UFSAR Sections 3.1.2.2.5, "Criterion 14 - Reactor Coolant Pressure Boundary," and 3.1.2.2.6, "Criterion 15 - Reactor Coolant System Design" (Ref. 1), the Reactor Coolant Pressure Boundary (RCPB) shall be designed with sufficient margin to ensure that the design conditions are not exceeded during normal operation and Abnormal Operational Transients.

During normal operation and Abnormal Operational Transients, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, in accordance with ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core may be done under LCO 3.10.1, "System Leakage and Hydrostatic Testing Operation." Following inception of unit operation, RCS components shall be pressure tested in accordance with the requirements of the Inservice Inspection Program (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB, reducing the number of protective barriers designed to prevent radioactive releases from exceeding the limits specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4). If this occurred in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The RCS safety/relief valves and the Reactor Protection System Reactor Vessel Steam Dome Pressure-High Function have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The SL is equal to the most limiting transient pressure allowed by the applicable codes for the reactor pressure vessel and Reactor Coolant System (RCS) piping. Since the SL is measured in the reactor steam dome, the SL is adjusted to ensure the most limiting transient pressure allowed by the applicable code is not exceeded at the lowest elevation of the RCS or the RPV.

The reactor pressure vessel (design pressure of 1250 psig at 575°F) is designed to Section III of the ASME, Boiler and Pressure Vessel Code, 1965 Edition, including Addenda through the summer of 1967 and various code cases (Ref. 5), which permits a maximum pressure transient of 110% of design pressure (110% x 1250 psig = 1375 psig). The RCS piping (suction: 1150 psig at 562°F; discharge 1325 psig at 562°F) is designed to the USAS, Nuclear Power Piping Code, Section B31.1, 1967 Edition (Ref. 6) which permits a maximum pressure transient of 120% of design pressure (suction: 120% x 1150 psig = 1380 psig; discharge: 120% x 1325 psig = 1590 psig) during 1% of the operating period.

To determine the SL, the maximum pressure transient limits allowed by the applicable codes were adjusted to account for the added pressure due to elevation sensed at the lowest points in the reactor pressure vessel and reactor coolant system piping. To achieve the most conservative pressure adjustment due to elevation, it was assumed the reactor pressure vessel was completely flooded with cold water. The highest elevation (head spray line flange) of the reactor pressure vessel is 841.2 feet. The lowest elevation (CRD housing flange) of the reactor pressure vessel is 760.8 feet. The lowest elevation (drain line off the bottom of the 'A' recirculation loop) of the RCS piping is 743.4 feet.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The maximum steam dome pressure allowed by the applicable code for the reactor pressure vessel is 1340 psig (1375 psig - [(841.2 ft - 760.8 ft) x 0.433 psi/ft] = 1340 psig). The maximum steam dome pressure allowed by the applicable code for the RCS piping is 1337.7 psig (1380 psig - [(841.2 ft - 743.4 ft) x 0.433 psi/ft] = 1337.7 psig). Therefore, the most limiting steam dome pressure allowed is 1337.7 psig. The SL is conservatively rounded to 1335 psig, which corresponds to the maximum steam dome pressure allowed by the applicable code for the RCS.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings is 120% of design pressures of 1150 psig for suction piping and 1325 psig for discharge piping. The most limiting of these allowances is the 120% of the suction piping design pressures; therefore, the SL on maximum allowable RCS pressure is established at 1335 psig as measured at the reactor steam dome.

APPLICABILITY

SL 2.1.2 applies in all MODES.

SAFETY LIMIT VIOLATIONS

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). Therefore, it is required to fully insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also assures that the probability of an accident occurring during this period is minimal.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Sections 3.1.2.2.5 and 3.1.2.2.6.
2. ASME, Boiler and Pressure Vessel Code, Section III, 1965 Edition, Addenda Summer 1967, Article 9.
3. Inservice Inspection Program, Latest Approved Edition.
4. 10 CFR 100.
5. ASME, Boiler and Pressure Vessel Code, Section III, 1965 Edition, Addenda Summer 1967 and Code Cases 1332-4, 1335-2 and 1420.
6. ASME, USAS, Nuclear Power Piping Code, Section B31.1, 1967 Edition.

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs

LCO 3.0.1 through LCO 3.0.7 establish the general requirements applicable to all Specifications in Sections 3.1 through 3.10 and apply at all times, unless otherwise stated.

LCO 3.0.1

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

LCO 3.0.2

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

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

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

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering

(continued)

BASES

LCO 3.0.2 (continued)

ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

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

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Condition no longer exists. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.9, "RCS Pressure and Temperature (P/T) Limits."

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

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

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BASES (continued)

LCO 3.0.3

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

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

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

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

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

(continued)

BASES

LCO 3.0.3 (continued)

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

a. The LCO is now met.

b. A Condition exists for which the Required Actions have now been performed.

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

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

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

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.8, "Spent Fuel Storage Pool Water Level." LCO 3.7.8 has an Applicability of "During movement of irradiated fuel

(continued)

BASES

LCO 3.0.3 (continued)

assemblies in the spent fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.8 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.8 of "Suspend movement of irradiated fuel assemblies in the spent fuel storage pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It precludes placing the unit in a MODE or other specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. Unit conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the unit being required to exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability

(continued)

BASES

LCO 3.0.4 (continued)

that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing MODES or other specified conditions while in an ACTIONS Condition, either in compliance with LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.4 is only applicable when entering MODE 3 from MODE 4, MODE 2 from MODE 3 or 4, or MODE 1 from MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODE 1, 2, or 3. The requirements of LCO 3.0.4 do not apply in MODES 4 and 5, or in other specified conditions of the Applicability (unless in MODE 1, 2, or 3) because the ACTIONS of individual specifications sufficiently define the remedial measures to be taken.

LCO 3.0.5

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

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

b. The OPERABILITY of other equipment.

(continued)

BASES

LCO 3.0.5 (continued)

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed SRs. This Specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the SRs.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of an SR on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of an SR on another channel in the same trip system.

LCO 3.0.6

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

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability.

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BASES

LCO 3.0.6 (continued)

However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the plant is maintained in a safe condition in the support system's Required Actions.

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

Specification 5.5.11, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

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

(continued)

BASES

LCO 3.0.6 (continued)

the LCO in which the loss of safety function exists are required to be entered.

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Special Operations LCOs in Section 3.10 allow specified TS requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Special Operations LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Special Operations LCOs is optional. A special operation may be performed either under the provisions of the appropriate Special Operations LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Special Operations LCO, the requirements of the Special Operations LCO shall be followed. When a Special Operations LCO requires another LCO to be met, only the requirements of the LCO statement are required to be met regardless of that LCO's Applicability (i.e., should the requirements of this other LCO not be met, the ACTIONS of the Special Operations LCO apply, not the ACTIONS of the other LCO). However, there are instances where the Special Operations LCO's ACTIONS may direct the other LCO's ACTIONS be met. The Surveillances of the other LCO are not required to be met, unless specified in the Special Operations LCO. If conditions exist such that the Applicability of any other LCO is met, all the other LCO's requirements (ACTIONS and SRs) are required to be met concurrent with the requirements of the Special Operations LCO.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications in Sections 3.1 through 3.10 and apply at all times, unless otherwise stated.

SR 3.0.1

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

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

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

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

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

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

(continued)

BASES

SR 3.0.1 (continued)

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

Some examples of this process are:

a. Control Rod Drive maintenance during refueling that requires scram testing at > 800 psi. However, if other appropriate testing is satisfactorily completed and the scram time testing of SR 3.1.4.3 is satisfied, the control rod can be considered OPERABLE. This allows startup to proceed to reach 800 psi to perform other necessary testing.

b. High pressure coolant injection (HPCI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPCI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.

SR 3.0.2

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

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

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BASES

SR 3.0.2 (continued)

transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. Therefore, when a test interval is specified in the regulations, the test interval cannot be extended by TS, and the SR includes a Note in the Frequency stating, "SR 3.0.2 is not applicable."

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

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

SR 3.0.3

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

(continued)

BASES

SR 3.0.3 (continued)

performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

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

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

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

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

(continued)

BASES

SR 3.0.3 (continued)

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

SR 3.0.4

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

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit.

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

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

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

(continued)

BASES

SR 3.0.4 (continued)

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs annotation is found in Section 1.4, Frequency.

SR 3.0.4 is only applicable when entering MODE 3 from MODE 4, MODE 2 from MODE 3 or 4, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODE 1, 2, or 3. The requirements of SR 3.0.4 do not apply in MODES 4 and 5, or in other specified conditions of the Applicability (unless in MODE 1, 2, or 3) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

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 Accidents;

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 the UFSAR (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 (Ref. 4) and fuel assembly insertion error during refueling (Ref. 5) accidents. 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 shut down

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 10 CFR 50.36(c)(2)(ii).

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. 5).

(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 is applicable for situations such as when a Required Action results in the reactor mode switch being placed in the shutdown position and not all insertable control rods have properly inserted, with SDM not met. This Action is also specifically applicable to the one rod withdrawn condition. Although the single control rod withdrawn condition is covered by Section 3.10 (Special Operations), the SDM limits of LCO 3.1.1 are still applicable. Therefore, the specified Required Action for SDM not within limits to insert all insertable control rods is appropriate and 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

(continued)

ACTIONS

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

control rods. Action must continue until all insertable control rods are fully inserted. This Action is applicable for situations such as when a Required Action results in the reactor mode switch being placed in the shutdown position and not all insertable control rods have properly inserted, with SDM not met. This Action is also specifically applicable to the one rod withdrawn condition. Although the single control rod withdrawn condition is covered by Section 3.10 (Special Operations), the SDM limits of LCO 3.1.1 are still applicable. Therefore, the specified Required Action for SDM not within limits to insert all insertable control rods is appropriate and 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 (SBGT) subsystem is OPERABLE; and secondary containment isolation capability for each associated secondary containment penetration flow path not isolated that is assumed to be isolated to mitigate radioactivity releases is available (i.e., at least one secondary containment isolation valve or damper and associated instrumentation are OPERABLE, or other acceptable administrative controls to assure isolation capability. These administrative controls consist of stationing a dedicated operator who is in continuous communications with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated.) OPERABILITY may be verified by 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.

(continued)

BASES (continued)

ACTIONS (continued)

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 completion of movement of a component to a safe condition. Inserting control rods or removing fuel from the core will reduce the total reactivity and are therefore excluded from the suspended actions.

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 SBGT subsystem is OPERABLE; and secondary containment isolation capability for each associated secondary containment penetration flow path not isolated that is assumed to be isolated to mitigate radioactivity releases (i.e., at least one secondary containment isolation valve or damper and associated instrumentation are OPERABLE, or other acceptable administrative controls to assure isolation capability. These administrative controls consist of stationing a dedicated operator who is in continuous communications with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated.). If CORE ALTERATIONS were in progress, secondary containment, one SBGT subsystem and secondary containment isolation capability would already be OPERABLE since OPERABILITY of these systems is a prerequisite for performing CORE ALTERATIONS. It is the intent of Required Actions E.3, E.4 and E.5 to maintain these systems OPERABLE until SDM can be restored. Therefore, a verification of the OPERABILITY of secondary containment, one SBGT subsystem and secondary containment isolation capability is required within one hour. OPERABILITY may be verified by an administrative

BASES (continued)

ACTIONS

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

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

Adequate SDM must be verified to ensure that the reactor can be made subcritical from any initial operating condition. This can be accomplished by a test, an evaluation, or a combination of the two. Adequate SDM is demonstrated before or during the first startup after fuel movement, shuffling within the reactor pressure vessel, or control rod replacement. 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 decreased by "R", which is the difference between the calculated value of BOC core reactivity during the operating cycle and the calculated maximum 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. 7). For the SDM verifications 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 verified 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 testing. Local critical tests require the withdrawal of out of

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

sequence control rods. This testing would therefore require bypassing of the rod worth minimizer to allow the out of sequence withdrawal, and therefore additional requirements must be met (see "Special Operations" 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 Modes 3 and 4, analytical calculation of SDM may be used to assure the requirements of SR 3.1.1.1 are met. During MODE 5, adequate SDM is required to ensure that the reactor does not reach criticality during control rod withdrawals. An evaluation of each in-vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. For example, bounding evaluations that verify adequate SDM for the most reactive configurations during the refueling may be performed to verify acceptability of the entire fuel movement sequence. These bounding evaluations 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. UFSAR, Section 3.1.2.3.7.
2. UFSAR, Section 15.4.7.
3. NEDE-24011-P-A-9-US, "General Electric Standard Application for Reactor Fuel," Supplement for United States, Revision as specified in the COLR.
4. UFSAR, Section 15.4.3.
5. UFSAR, Section 15.4.4.

(continued)

BASES (continued)

REFERENCES (continued)

6. UFSAR, Section 4.3.2.4.1.

7. NEDE-24011-P-A-9, "General Electric Standard Application for Reactor Fuel," Section 3.2.4.1. Revision as specified in the COLR.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Anomalies

BASES

BACKGROUND

In accordance with the UFSAR (Ref. 1), reactivity shall be controllable such that subcriticality is maintained under cold conditions and acceptable fuel design limits are not exceeded during normal operation and abnormal operational transients. Therefore, reactivity anomaly is used as a measure of the predicted versus measured core reactivity during power operation. The continual confirmation of core reactivity is necessary to ensure that the Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity anomaly could be the result of unanticipated changes in fuel reactivity or control rod worth or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in assuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers, producing zero net reactivity.

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading and the fuel loaded in the previous cycles provide excess positive reactivity beyond that required to sustain steady state operation at the beginning of cycle (BOC). When the reactor is critical at RTP and operating moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, and whatever neutron

(continued)

BASES

BACKGROUND (continued)	poisons (mainly xenon and samarium) that are present in the fuel. The predicted core reactivity, as represented by control rod density, is calculated by a 3D core simulator code as a function of cycle exposure. This calculation is performed for projected operating states and conditions throughout the cycle. The core reactivity is determined from control rod densities for actual plant conditions and is then compared to the predicted value for the cycle exposure.
APPLICABLE SAFETY ANALYSES	Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations (Ref. 2). In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod drop accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity anomaly provides additional assurance that the nuclear methods provide an accurate representation of the core reactivity. The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted rod density for identical core conditions at BOC do not reasonably agree, then the assumptions used in the reload cycle design analysis or the calculation models used to predict rod density may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured value. Thereafter, any significant deviations in the measured rod density from the predicted rod density that develop during fuel depletion may be an indication that the assumptions of the DBA and transient analyses are no longer valid, or that an unexpected change in core conditions has occurred. Reactivity anomalies satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO

The reactivity anomaly limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between monitored and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the "Nuclear Design Methodology" are larger than expected. A limit on the difference between the monitored and the predicted rod density of ± 1% Δk/k has been established based on engineering judgment. A > 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

APPLICABILITY

In MODE 1, most of the control rods are withdrawn and steady state operation is typically achieved. Under these conditions, the comparison between predicted and monitored core reactivity provides an effective measure of the reactivity anomaly. In MODE 2, control rods are typically being withdrawn during a startup. In MODES 3 and 4, all control rods are fully inserted and therefore the reactor is in the least reactive state, where monitoring core reactivity is not necessary. In MODE 5, fuel loading results in a continually changing core reactivity. SDM requirements (LCO 3.1.1) ensure that fuel movements are performed within the bounds of the safety analysis, and an SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, shuffling). The SDM test, required by LCO 3.1.1, provides a direct comparison of the predicted and monitored core reactivity at cold conditions; therefore, reactivity anomaly is not required during these conditions.

ACTIONS

A.1

Should an anomaly develop between measured and predicted core reactivity, the core reactivity difference must be restored to within the limit to ensure continued operation is within the core design assumptions. Restoration to within the limit could be performed by an evaluation of the

(continued)

BASES

ACTIONS

A.1 (continued)

core design and safety analysis to determine the reason for the anomaly. This evaluation normally reviews the core conditions to determine their consistency with input to design calculations. Measured core and process parameters are also normally evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models may be reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 72 hours is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

B.1

If the core reactivity cannot be restored to within the 1% Δk/k limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 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.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Verifying the reactivity difference between the monitored and predicted rod density is within the limits of the LCO provides added assurance that plant operation is maintained within the assumptions of the DBA and transient analyses. The Plant Process Computer calculates the rod density for the reactor conditions obtained from plant instrumentation. A comparison of the monitored rod density to the predicted rod density at the same cycle exposure is used to calculate the reactivity difference. The comparison is required when the core reactivity has potentially changed by a significant amount. This may occur following a refueling in which new fuel assemblies are loaded, fuel assemblies are shuffled within the core, or control rods are replaced or shuffled. Control rod replacement refers to the decoupling

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1 (continued)

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. Also, core reactivity changes during the cycle. The 24 hour interval after reaching equilibrium conditions following a startup is based on the need for equilibrium xenon concentrations in the core, such that an accurate comparison between the monitored and predicted rod density can be made. For the purposes of this SR, the reactor is assumed to be at equilibrium conditions when steady state operations (no control rod movement or core flow changes) at ≥ 75% RTP have been obtained. Additionally, the Reactor Engineer or individual fulfilling this role will normally be involved with determining equilibrium xenon conditions. The 1000 MWD/T Frequency was developed, considering the relatively slow change in core reactivity with exposure and operating experience related to variations in core reactivity. This comparison requires the core to be operating at power levels which minimize the uncertainties and measurement errors, in order to obtain meaningful results. Therefore, the comparison is only done when in MODE 1.

REFERENCES

1. UFSAR, Sections 3.1.2.3.7, 3.1.2.3.9, and 3.1.2.3.10.
2. UFSAR, Chapter 15.4.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Control Rod OPERABILITY

BASES

BACKGROUND

Control rods are components of the Control Rod Drive (CRD) System, which is the primary reactivity control system for the reactor. In conjunction with the Reactor Protection System, the CRD System provides the means for the reliable control of reactivity changes to ensure under conditions of normal operation, including anticipated operational transients, that specified acceptable fuel design limits are not exceeded. In addition, the control rods provide the capability to hold the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the CRD System. The CRD System is designed to satisfy the requirements specified in the UFSAR (Ref. 1).

The CRD System consists of 89 locking piston control rod drive mechanisms (CRDMs) and a hydraulic control unit for each drive mechanism. The locking piston type CRDM is a double-acting hydraulic piston, which uses condensate water as the operating fluid. Accumulators provide additional energy for scram. An index tube and piston, coupled to the control rod, are locked at fixed increments by a collet mechanism. The collet fingers engage notches in the index tube to prevent unintentional withdrawal of the control rod, but without restricting insertion.

This Specification, along with LCO 3.1.4, "Control Rod Scram Times," and LCO 3.1.5, "Control Rod Scram Accumulators," ensure that the performance of the control rods in the event of a Design Basis Accident (DBA) or transient meets the assumptions used in the safety analyses of References 2, 3, and 4.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in the evaluations involving control rods are presented in References 2, 3, and 4. The control rods provide the primary means for rapid reactivity control (reactor scram), for maintaining the reactor subcritical and for limiting the potential effects of reactivity insertion events caused by

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

malfunctions in the CRD System. (e.g., the control rod velocity limiter). The capability to insert the control rods provides assurance that the assumptions for scram reactivity in the DBA and transient analyses are not violated. Since the SDM ensures the reactor will be subcritical with the highest worth control rod withdrawn (assumed single failure), the additional failure of a second control rod to insert, if required, could invalidate the demonstrated SDM and potentially limit the ability of the CRD System to hold the reactor subcritical. If the control rod is stuck at an inserted position and becomes decoupled from the CRD, a control rod drop accident (CRDA) can possibly occur. Therefore, the requirement that all control rods be OPERABLE ensures the CRD System can perform its intended function.

The control rods also protect the fuel from damage which could result in release of radioactivity. The limits protected are 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)"), and the fuel damage limit (see Bases for LCO 3.1.6, "Rod Pattern Control") during reactivity insertion events.

The negative reactivity insertion (scram) provided by the CRD System provides the analytical basis for determination of plant thermal limits and provides protection against fuel damage limits during a CRDA. The Bases for LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6 discuss in more detail how the SLs are protected by the CRD System.

Control rod OPERABILITY satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The OPERABILITY of an individual control rod is based on a combination of factors, primarily, the scram insertion times, the control rod coupling integrity, and the ability to determine the control rod position. Accumulator OPERABILITY is addressed by LCO 3.1.5. The associated scram accumulator status for a control rod only affects the scram insertion times; therefore, an inoperable accumulator does

(continued)

BASES

LCO (continued)

not immediately require declaring a control rod inoperable. Although not all control rods are required to be OPERABLE to satisfy the intended reactivity control requirements, strict control over the number and distribution of inoperable control rods is required to satisfy the assumptions of the DBA and transient analyses.

APPLICABILITY

In MODES 1 and 2, the control rods are assumed to function during a DBA or transient and are therefore required to be OPERABLE in these MODES. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod OPERABILITY during these conditions. Control rod requirements in MODE 5 are located in LCO 3.9.5, "Control Rod OPERABILITY – Refueling."

ACTIONS

The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each control rod. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable control rod. Complying with the Required Actions may allow for continued operation, and subsequent inoperable control rods are governed by subsequent Condition entry and application of associated Required Actions.

A.1, A.2, A.3, and A.4

A control rod is considered stuck if it will not insert by either CRD drive water or scram pressure. With a fully inserted control rod stuck, no actions are required as long as the control rod remains fully inserted. The Required Actions are modified by a Note, which allows the Rod Worth Minimizer (RWM) to be bypassed if required to allow continued operation. LCO 3.3.2.1, "Control Rod Block Instrumentation," provides additional requirements when the RWM is bypassed to ensure compliance with the CRDA analysis. With one withdrawn control rod stuck, the local scram reactivity rate assumptions may not be met if the stuck control rod separation criteria are not met. Therefore, a verification that the separation criteria are met must be performed immediately. The separation criteria are not met

(continued)

BASES

ACTIONS

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

if a) the stuck control rod occupies a location adjacent to two "slow" control rods, b) the stuck control rod occupies a location adjacent to one "slow" control rod, and the one "slow" control rod is also adjacent to another "slow" control rod, or c) the stuck control rod occupies a location adjacent to one "slow" control rod when there is another pair of "slow" control rods adjacent to one another. The description of "slow" control rod is provided in LCO 3.1.4, "Control Rod Scram Times." In addition, the associated control rod drive must be disarmed in 2 hours. The allowed Completion Time of 2 hours is acceptable, considering the reactor can still be shut down, assuming no additional control rods fail to insert, and provides a reasonable time to perform the Required Action in an orderly manner. The control rod must be isolated from both scram and normal insert and withdraw pressure. Isolating the control rod from scram and normal insert and withdraw pressure prevents damage to the CRDM. The control rod must be isolated by isolating the hydraulic control unit from the scram and normal insert and withdraw pressure, while still maintaining cooling water to the CRD.

Monitoring of the insertion capability of each withdrawn control rod must also be performed within 24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP)of the RWM. SR 3.1.3.2 and SR 3.1.3.3 perform periodic tests of the control rod insertion capability of withdrawn control rods. Testing each withdrawn control rod ensures that a generic problem does not exist. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." The Required Action A.3 Completion Time only begins upon discovery of Condition A concurrent with THERMAL POWER greater than the actual LPSP of the RWM, since the notch insertions may not be compatible with the requirements of rod pattern control (LCO 3.1.6) and the RWM (LCO 3.3.2.1). The allowed Completion Time of 24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM provides a reasonable time to test the control rods, considering the potential for a need to reduce power to perform the tests.

(continued)

BASES

ACTIONS

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

To allow continued operation with a withdrawn control rod stuck, an evaluation of adequate SDM is also required within 72 hours. Should a DBA or transient require a shutdown, to preserve the single failure criterion, an additional control rod would have to be assumed to fail to insert when required. Therefore, the original SDM demonstration may not be valid. The SDM must therefore be evaluated (by measurement or analysis) with the stuck control rod at its stuck position and the highest worth OPERABLE control rod assumed to be fully withdrawn.

The allowed Completion Time of 72 hours to verify SDM is adequate, considering that with a single control rod stuck in a withdrawn position, the remaining OPERABLE control rods are capable of providing the required scram and shutdown reactivity. Failure to reach MODE 4 is only likely if an additional control rod adjacent to the stuck control rod also fails to insert during a required scram. Even with the postulated additional single failure of an adjacent control rod to insert, sufficient reactivity control remains to reach and maintain MODE 3 conditions (Ref. 5).

B.1

With two or more withdrawn control rods stuck, the plant must be brought to MODE 3 within 12 hours. The occurrence of more than one control rod stuck at a withdrawn position increases the probability that the reactor cannot be shut down if required. Insertion of all insertable control rods eliminates the possibility of an additional failure of a control rod to insert. 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 and C.2

With one or more control rods inoperable for reasons other than being stuck in the withdrawn position, operation may continue, provided the control rods are fully inserted within 3 hours and disarmed (electrically or hydraulically) within 4 hours. During the 3 hour time limit, attempts to

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

re-couple the rod, by inserting the rod and withdrawing it to the "full-out" position to verify that the rod does not go to the "overtravel" position, may be made. However, numerous attempts to re-couple the rod should be avoided to prevent further damage to the rod or drive. If the rod can not be re-coupled, inserting the 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. The control rods can be electrically disarmed by disconnecting power from all four directional control valve solenoids. Required Action C.1 is modified by a Note, which allows the RWM to be bypassed if required to allow insertion of the inoperable control rods and continued operation. LCO 3.3.2.1 provides 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.

D.1 and D.2

Out of sequence control rods may increase the potential reactivity worth of a dropped control rod during a CRDA. At ≤ 10% RTP, the generic Banked Position Withdrawal Sequence (BPWS) analysis (Ref. 5) requires inserted control rods not in compliance with BPWS to be separated by at least two OPERABLE control rods in all directions, including the diagonal. Therefore, if two or more inoperable control rods are not in compliance with BPWS and not separated by at least two OPERABLE control rods, action must be taken to restore compliance with BPWS or restore the control rods to OPERABLE status. Condition D is modified by a Note indicating that the Condition is not applicable when > 10% RTP, since the BPWS is not required to be followed under these conditions, as described in the Bases for LCO 3.1.6. The allowed Completion Time of 4 hours is acceptable, considering the low probability of a CRDA occurring.

(continued)

BASES

ACTIONS (continued)

E.1

If any Required Action and associated Completion Time of Condition A, C, or D are not met, or there are nine or more inoperable control rods, 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. This ensures all insertable control rods are inserted and places the reactor in a condition that does not require the active function (i.e., scram) of the control rods. The number of control rods permitted to be inoperable when operating above 10% RTP (e.g., no CRDA considerations) could be more than the value specified, but the occurrence of a large number of inoperable control rods could be indicative of a generic problem, and investigation and resolution of the potential problem should be undertaken. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

The position of each control rod must be determined to ensure adequate information on control rod position is available to the operator for determining control rod OPERABILITY and controlling rod patterns. Control rod position may be determined by the use of OPERABLE position indicators, by moving control rods to a position with an OPERABLE indicator, by use of TIP traces, by alternate rod position determination methods, or by the use of other appropriate methods. The 24 hour Frequency of this SR is based on operating experience related to expected changes in control rod position and the availability of control rod position indications in the control room.

SR 3.1.3.2 and SR 3.1.3.3

Control rod insertion capability is demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. The control rod may then be returned to its original position. This ensures the control rod is not stuck and is free to insert on a scram signal. These Surveillances are

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.2 and SR 3.1.3.3 (continued)

not required when THERMAL POWER is less than or equal to 20% RTP since the notch insertions may not be compatible with the requirements of the Banked Position Withdrawal Sequence (BPWS) (LCO 3.1.6) and the RWM (LCO 3.3.2.1). The 7 day Frequency of SR 3.1.3.2 is based on operating experience related to the changes in CRD performance and the ease of performing notch testing for fully withdrawn control rods. Partially withdrawn control rods are tested at a 31 day Frequency, based on the potential power reduction required to allow the control rod movement and considering the large testing sample of SR 3.1.3.2. Furthermore, the 31 day Frequency takes into account operating experience related to changes in CRD performance. At any time, if a control rod is immovable (e.g., due to an inoperable insert or withdrawn solenoid valve), a determination of that control rod's ability to be moved with scram pressure OPERABILITY must be made and appropriate action taken.

These SRs are modified by Notes that allow 7 days and 31 days respectively, after withdrawal of the control rod and increasing power to above 20% RTP, to perform the Surveillance. This acknowledges that the control rod must be first withdrawn and THERMAL POWER must be increased to above 20% RTP before performance of the Surveillance, and therefore the Notes avoid potential conflicts with SR 3.0.3 and SR 3.0.4.

SR 3.1.3.4

Verifying that the scram time for each control rod to notch position 04 is ≤ 7 seconds provides reasonable assurance that the control rod will insert when required during a DBA or transient, thereby completing its shutdown function. This SR is performed in conjunction with the control rod scram time testing of SR 3.1.4.1 and SR 3.1.4.2. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," and the functional testing of SDV vent and drain valves in LCO 3.1.8, "Scram Discharge Volume (SDV) Vent and Drain Valves," overlap this Surveillance to provide complete testing of the assumed safety function. The associated Frequencies are acceptable, considering the more frequent testing performed to

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.4 (continued)

demonstrate other aspects of control rod OPERABILITY and operating experience, which shows scram times do not significantly change over an operating cycle.

SR 3.1.3.5

Coupling verification is performed to ensure the control rod is connected to the CRDM and will perform its intended function when necessary. The Surveillance requires verifying a withdrawn control rod does not go to the withdrawn overtravel position. The overtravel position feature provides a positive check on the coupling integrity since only an uncoupled CRD can reach the overtravel position. The verification is required to be performed any time a control rod is withdrawn to the "full out" position (notch position 48) or prior to declaring the control rod OPERABLE after work on the control rod or CRD System that could affect coupling. This includes control rods inserted one notch and then returned to the "full out" position during the performance of SR 3.1.3.2. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved and operating experience related to uncoupling events.

REFERENCES

1. UFSAR, Sections 3.1.2.3.7, 3.1.2.3.8, 3.1.2.3.9, and 3.1.2.3.10
2. UFSAR, Section 4.3.2.4 and 4.3.2.5.
3. UFSAR, Section 5.2.2.
4. UFSAR, Section 15.1.
5. NEDO-21231, "Banked Position Withdrawal Sequence," Section 7.2, January 1977.

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 and Design Basis Accidents (DBAs) 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 pressure, or reactor pressure is applied 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 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 (Ref. 5) and, therefore, also provides protection against violating fuel damage limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control"). For the reactor vessel overpressure protection analysis (Ref. 3), 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 10 CFR 50.36(c)(2)(ii).

LCO

The scram times specified in Table 3.1.4-1 are required to ensure that the scram reactivity assumed in the DBA and transient analysis is met (Ref. 6). 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., 89 x 7% ≈ 6) to have scram times exceeding the specified limits (i.e., "slow" control rods) assuming 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 two of the allowed "slow" control rods may occupy adjacent locations.

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 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 two 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 a refueling or after a shutdown ≥ 120 days, 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, only those CRDs associated with the core cells affected by the fuel movements are required to be scram time tested in accordance with SR 3.1.4.2. 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

When work that could affect the scram insertion time is performed on a control rod or CRD System, or when fuel movement within the reactor pressure vessel occurs, 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. Alternatively, a control rod scram test during hydrostatic pressure testing could also satisfy both criteria. When only fuel movement occurs, then only those control rods associated with the core cells affected by the fuel movement are required to be scram time tested.

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. UFSAR, Section 3.1.2.2.1
2. UFSAR, Section 4.3.2.4 and 4.3.2.5.
3. UFSAR, Section 5.2.2.
4. UFSAR, Section 15.1.
5. NEDE-24011-P-A-9, "General Electric Standard Application for Reactor Fuel," Section 3.2.4.1 Revision as specified in the COLR.
6. 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 3.1.5 Control Rod Scram Accumulators

BASES

BACKGROUND	The control rod scram accumulators are part of the Control Rod Drive (CRD) System and are provided to ensure that the control rods scram under varying reactor conditions. The control rod scram accumulators store sufficient energy to fully insert a control rod at any reactor vessel pressure. The accumulator is a hydraulic cylinder with a free floating piston. The piston separates the water used to scram the control rods from the nitrogen, which provides the required energy. The scram accumulators are necessary to scram the control rods within the required insertion times of LCO 3.1.4, "Control Rod Scram Times."
APPLICABLE SAFETY ANALYSES	The analytical methods and assumptions used in evaluating the control rod scram function are presented in References 1, 2, and 3. The Design Basis Accident (DBA) and transient analyses assume that all of the control rods scram at a specified insertion rate. OPERABILITY of each individual control rod scram accumulator, along with LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.4, ensures that the scram reactivity assumed in the DBA and transient analyses can be met. The existence of an inoperable accumulator may invalidate prior scram time measurements for the associated control rod. The scram function of the CRD System, and therefore the OPERABILITY of the accumulators, protects the MCPR Safety Limit (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and 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 (see Bases for LCO 3.1.4). In addition, the scram function at low reactor vessel pressure (i.e., startup conditions) provides protection against violating fuel design limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control").

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	Control rod scram accumulators satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The OPERABILITY of the control rod scram accumulators is required to ensure that adequate scram insertion capability exists when needed over the entire range of reactor pressures. The OPERABILITY of the scram accumulators is based on maintaining adequate accumulator pressure.
APPLICABILITY	In MODES 1 and 2, the scram function is required for mitigation of DBAs and transients, and therefore the scram accumulators must be OPERABLE to support the scram function. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod scram accumulator OPERABILITY during these conditions. Requirements for scram accumulators in MODE 5 are contained in LCO 3.9.5, "Control Rod OPERABILITY - Refueling."
ACTIONS	The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each control rod scram accumulator. This is acceptable since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable accumulator. Complying with the Required Actions may allow for continued operation and subsequent inoperable accumulators governed by subsequent Condition entry and application of associated Required Actions. <u>A.1 and A.2</u> With one control rod scram accumulator inoperable and the reactor steam dome pressure ≥ 900 psig, the control rod may be declared "slow," since the control rod will still scram at the reactor operating pressure but may not satisfy the required scram times in Table 3.1.4-1.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Required Action A.1 is modified by a Note indicating that declaring the control rod "slow" only applies if the associated control scram time was within the limits of Table 3.1.4-1 during the last scram time surveillance. Otherwise, the control rod would already be considered "slow" and the further degradation of scram performance with an inoperable accumulator could result in excessive scram times. In this event, the associated control rod is declared inoperable (Required Action A.2) and LCO 3.1.3 is entered. This would result in requiring the affected control rod to be fully inserted and disarmed, thereby satisfying its intended function in accordance with ACTIONS of LCO 3.1.3.

The allowed Completion Time of 8 hours is reasonable, based on the large number of control rods available to provide the scram function and the ability of the affected control rod to scram only with reactor pressure at high reactor pressures.

B.1, B.2.1, and B.2.2

With two or more control rod scram accumulators inoperable and reactor steam dome pressure ≥ 900 psig, adequate pressure must be supplied to the charging water header. With inadequate charging water pressure, all of the accumulators could become inoperable, resulting in a potentially severe degradation of the scram performance. Therefore, within 1 hour from discovery of charging water header pressure < 940 psig concurrent with Condition B, adequate charging water header pressure must be restored. The allowed Completion Time of 1 hour is reasonable, to place a CRD pump into service to restore the charging water header pressure, if required. This Completion Time is based on the ability of the reactor pressure alone to fully insert all control rods and the need to place the CRD pump into service in a controlled manner. The CRD system provides water to the Reactor Level Instruments Reference Legs Backfill System. Unless properly isolated from the CRD system prior to a CRD pump start, there is the possibility of introducing air into the Reactor Level Instruments Reference Legs Backfill System. Air in the Reactor Level Instruments Reference Legs could cause inadvertent ECCS

(continued)

BASES

ACTIONS

B.1, B.2.1 and B.2.2 (continued)

initiations and PCIS isolations.

The control rod may be declared "slow," since the control rod will still scram using only reactor pressure, but may not satisfy the times in Table 3.1.4-1. Required Action B.2.1 is modified by a Note indicating that declaring the control rod "slow" only applies if the associated control rod scram time is within the limits of Table 3.1.4-1 during the last scram time surveillance. Otherwise, the control rod would already be considered "slow" and the further degradation of scram performance with an inoperable accumulator could result in excessive scram times. In this event, the associated control rod is declared inoperable (Required Action B.2.2) and LCO 3.1.3 entered. This would result in requiring the affected control rod to be fully inserted and disarmed, thereby satisfying its intended function in accordance with ACTIONS of LCO 3.1.3.

The allowed Completion Time of 1 hour is reasonable, based on the ability of only the reactor pressure to scram the control rods and the low probability of a DBA or transient occurring while the affected accumulators are inoperable.

C.1 and C.2

With one or more control rod scram accumulators inoperable and the reactor steam dome pressure < 900 psig, the pressure supplied to the charging water header must be adequate to ensure that accumulators remain charged. With the reactor steam dome pressure < 900 psig, the function of the accumulators in providing the scram force becomes much more important since the scram function could become severely degraded during a depressurization event or at low reactor pressures. Therefore, immediately upon discovery of charging water header pressure < 940 psig, concurrent with Condition C, all control rods associated with inoperable accumulators must be verified to be fully inserted. Withdrawn control rods with inoperable accumulators may fail to scram under these low pressure conditions. The associated control rods must also be declared inoperable within 1 hour. The allowed Completion Time of 1 hour is reasonable for Required Action C.2, considering the low

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BASES (continued)

ACTIONS

C.1 and C.2 (continued)

probability of a DBA or transient occurring during the time that the accumulator is inoperable.

D.1

The reactor mode switch must be immediately placed in the shutdown position if either Required Action and associated Completion Time associated with loss of the CRD charging pump (Required Actions B.1 and C.1) cannot be met. This ensures that all insertable control rods are inserted and that the reactor is in a condition that does not require the active function (i.e., scram) of the control rods. The insertion of a manual scram prior to placing the reactor mode switch in the Shutdown position is permitted by the definition of an Immediate Completion Time. This Required Action is modified by a Note stating that the action is not applicable if all control rods associated with the inoperable scram accumulators are fully inserted, since the function of the control rods has been performed.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

SR 3.1.5.1 requires that the accumulator pressure be checked every 7 days to ensure adequate accumulator pressure exists to provide sufficient scram force. The primary indicator of accumulator OPERABILITY is the accumulator pressure. 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 (Ref. 1). Declaring the accumulator inoperable when the minimum pressure is not maintained ensures that significant degradation in scram times does not occur. The 7 day Frequency has been shown to be acceptable through operating experience and takes into account indications available in the control room.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 4.3.2.4 and 4.3.2.5.
2. UFSAR, Section 5.2.2.
3. UFSAR, Section 15.1.

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 (Refs. 1 and 6) of a design basis CRDA (i.e., a CRDA resulting in a peak fuel energy deposition of 280 cal/gm) have shown that if the peak fuel enthalpy remains below 280 cal/gm, then the maximum reactor pressure will be less than the required ASME Code limits (Ref. 7) and the calculated offsite doses will be well within the required limits (Ref. 5).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 (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. Generic analysis of the BPWS (Ref. 1) has demonstrated that the 280 cal/gm fuel damage limit will not be violated during a CRDA while following the BPWS Mode of operation. The generic BPWS analysis (Ref. 8) also evaluates the effect of fully inserted, inoperable control rods not in compliance with the sequence, to allow a limited number (i.e., eight) and distribution of fully inserted, inoperable control rods. The Reduced Notch Worth Procedure (RNWP) (Ref. 9) is an extension of the BPWS and may be used at DAEC during control rod withdrawal to avoid high notch worth scrams.

Rod pattern control satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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

(continued)

BASES

APPLICABILITY (continued)

that the consequences of a CRDA are acceptable, since the reactor will remain subcritical with a single control rod withdrawn.

ACTIONS

A.1 and A.2

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.

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 second licensed operator or a qualified member of the technical staff (i.e., personnel trained in accordance with an approved training program). 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. 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.

(continued)

BASES (continued)

ACTIONS (continued)

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 must 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 second licensed operator or 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 Required Action to place the reactor mode switch in the Shutdown position within 1 hour does not preclude a manual scram from being inserted beforehand. 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.

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.

(continued)

BASES (continued)

REFERENCES

1. NEDE-24011-P-A-9-US, "General Electric Standard Application for Reactor Fuel, Supplement for United States," Section 2.2.3.1, Revision as specified in the COLR.

2. Letter from T. A. Pickens (BWROG) to G.E. Laines (NRC), "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A", BWROG-8644, August 15, 1988.

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

9. GE SIL No. 316, "Reduced Notch Worth Procedures," November 1979.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND

The SLC System is designed to provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory (which is at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive, xenon free state without taking credit for control rod movement. In addition, the SLC System is relied upon to satisfy the requirements of 10 CFR 50.62 (Ref. 1) on Anticipated Transient Without Scram (ATWS).

The SLC System consists of a boron solution storage tank, two positive displacement pumps, two explosive valves that are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged near the bottom of the core shroud, where it then mixes with the cooling water rising through the core. A smaller tank containing demineralized water is provided for testing purposes.

APPLICABLE SAFETY ANALYSES

The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that enough control rods cannot be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to add negative reactivity to compensate for all of the various reactivity effects that could occur during plant operations. To meet this objective, it is necessary to inject a quantity of boron, using both SLC pumps, which produces a concentration of 600 ppm of natural boron, in the reactor coolant at 70°F. To allow for potential leakage and imperfect mixing in the reactor system, an amount of boron equal to 25% of the amount cited above is added (Ref. 2). The volume versus concentration limits in Figure 3.1.7-1 and the temperature versus concentration limits in Figure 3.1.7-2 are calculated

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

such that the required concentration is achieved accounting for dilution in the RPV with normal water level and including the water volume in the residual heat removal shutdown cooling piping and in the recirculation loop piping. This quantity of borated solution is the amount that is above the pump suction shutoff level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected. The NRC's final rule on Anticipated Transients Without Scram (ATWS), 10 CFR 50.62, requires that the SLC System be modified to provide an equivalent shutdown capability of 86 gpm at 13.4 weight percent natural boron for a 251 inch I.D. vessel. For DAEC, ATWS equivalence is achieved by running both SLC system pumps simultaneously at a minimum combined flow of 45 gpm at a nominal boron concentration of 13% weight percent natural boron, (Ref. 3). The equivalence is also met if both pumps supply their minimum Technical Specification flowrate of 26.2 gpm each with a solution concentration of at least 11.2 weight percent natural boron. Although the SLC pump circuitry has been modified to run both pumps simultaneously in order to comply with the ATWS rule requirements, only one of the two SLC pumps is needed for meeting the original design basis.

The SLC System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii) because operating experience and probabilistic risk assessments have shown the SLC System to be important to public health and safety. Thus, it is retained in the Technical Specifications.

LCO

The OPERABILITY of the SLC System provides backup capability for reactivity control independent of normal reactivity control provisions provided by the control rods. The OPERABILITY of the SLC System is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the RPV, including the OPERABILITY of the pumps and valves. Two SLC subsystems are required to be OPERABLE; each contains an OPERABLE pump, an explosive valve, and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1 and 2, shutdown capability is required. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate controls to ensure that the reactor remains subcritical. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") ensures that the reactor will not become critical. Therefore, the SLC System is not required to be OPERABLE when only a single control rod can be withdrawn.

ACTIONS

A.1

If one SLC subsystem is inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystem is adequate to perform the original licensing basis shutdown function. However, the overall capability is reduced since the remaining OPERABLE subsystem cannot meet the requirements of Reference 1. The 7 day Completion Time is based on the availability of an OPERABLE subsystem capable of performing the original licensing basis SLC System function and the low probability of an event occurring concurrent with the failure of the Control Rod Drive (CRD) System to shut down the plant.

B.1

If both SLC subsystems are inoperable at least one subsystem must be restored to OPERABLE status within 8 hours. The allowed Completion Time of 8 hours is considered acceptable given the low probability of the failure of the control rods to shut down the reactor.

(continued)

BASES (continued)

ACTIONS (continued)

C.1

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to 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.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1, SR 3.1.7.2, and SR 3.1.7.3

SR 3.1.7.1 through SR 3.1.7.3 are 24 hour Surveillances verifying certain characteristics of the SLC System (e.g., the volume and temperature of the borated solution in the storage tank), thereby ensuring SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure that the proper borated solution volume and temperature, including the temperature of the pump suction piping, are maintained. Maintaining a minimum specified borated solution temperature is important in ensuring that the boron remains in solution and does not precipitate out in the storage tank or in the pump suction piping. The temperature versus concentration curve of Figure 3.1.7-2 ensures that a 5°F margin will be maintained above the saturation temperature. The 24 hour Frequency is based on operating experience and has shown there are relatively slow variations in the measured parameters of volume and temperature.

SR 3.1.7.4

SR 3.1.7.4 verifies the continuity of the explosive charges in the injection valves to ensure that proper operation will occur if required. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on operating experience and has demonstrated the reliability of the explosive charge continuity.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.5

This Surveillance requires an examination of the sodium pentaborate solution by using chemical analysis to ensure that the proper concentration of boron exists in the storage tank. SR 3.1.7.5 must be performed anytime boron or water is added to the storage tank solution to determine that the boron solution concentration is within the specified limits. SR 3.1.7.5 must also be performed anytime the temperature is restored to within the limits of Figure 3.1.7-2, to ensure that no significant boron precipitation occurred. The 31 day Frequency of this Surveillance is appropriate because of the relatively slow variation of boron concentration between surveillances.

SR 3.1.7.6

Demonstrating that each SLC System pump develops a flow rate ≥ 26.2 gpm at a discharge pressure ≥ 1150 psig when pumping demineralized water to the test tank ensures that pump performance has not degraded below design values during the fuel cycle. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this Surveillance is in accordance with the Inservice Testing Program.

SR 3.1.7.7 and SR 3.1.7.8

These Surveillances ensure that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. The pump and explosive valve tested should be alternated such that both complete flow paths are

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.7 and SR 3.1.7.8 (continued)

tested every 48 months at alternating 24 month intervals. The Surveillance may be performed in separate steps to prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. The 24 month Frequency 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 these components usually pass the Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Demonstrating that all heat traced piping between the boron solution storage tank and the suction inlet to the injection pumps is unblocked ensures that there is a functioning flow path for injecting the sodium pentaborate solution. Acceptable methods for verifying that the suction piping is unblocked include pumping from the storage tank to the test tank or establishing flow from the pump suction drains.

The 24 month Frequency is acceptable since there is a low probability that the subject piping will be blocked due to precipitation of the boron from solution in the heat traced piping. This is especially true in light of the temperature verification of this piping required by SR 3.1.7.3. However, if, in performing SR 3.1.7.3, it is determined that the temperature of this piping has fallen below the specified minimum, SR 3.1.7.8 must be performed once within 24 hours after the piping temperature is restored to within the limits of Figure 3.1.7-2.

REFERENCES

1. 10 CFR 50.62.

2. UFSAR, Section 9.3.4.3.

3. NEDC-30859, "Duane Arnold ATWS Assessments" December 1984.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

BASES

BACKGROUND

The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. As discussed in Reference 3, the SDV vent and drain valves need not be considered Primary Containment Isolation Valves (PCIVs) for the Scram Discharge System. The SDV is a volume of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two SDVs (headers) and two instrument volumes, each receiving approximately one half of the control rod drive (CRD) discharges. The two instrument volumes are connected to a common drain line with two valves in series; and each header is connected to a common vent line with two valves in series, for a total of four vent and drain valves. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram. The design and functions of the SDV are described in Reference 1.

APPLICABLE SAFETY ANALYSES

The Design Basis Accident and transient analyses assume all of the control rods are capable of scramming. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to:

a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR 100 (Ref. 2); and

b. Open on scram reset to maintain the SDV vent and drain path open so that there is sufficient volume to accept the reactor coolant discharged during a scram.

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. For a bounding leakage case, the offsite doses are well within the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	limits of 10 CFR 100 (Ref. 2), and adequate core cooling is maintained (Ref. 3). The SDV vent and drain valves allow continuous drainage of the SDV during normal plant operation to ensure that the SDV has sufficient capacity to contain the reactor coolant discharge during a full core scram. To automatically ensure this capacity, a reactor scram (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation") is initiated if the SDV water level in the instrument volume exceeds a specified setpoint. The setpoint is chosen so that all control rods are inserted before the SDV has insufficient volume to accept a full scram. SDV vent and drain valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The OPERABILITY of all SDV vent and drain valves ensures that the SDV vent and drain valves will close during a scram to contain reactor water discharged to the SDV piping. Since the vent and drain lines are provided with two valves in series, the single failure of one valve in the open position will not impair the isolation function of the system. Additionally, the valves are required to open on scram reset to ensure that a path is available for the SDV piping to drain freely at other times.
APPLICABILITY	In MODES 1 and 2, scram may be required; therefore, the SDV vent and drain valves must be OPERABLE. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate controls to ensure that the reactor remains subcritical. Also, during MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Therefore, the SDV vent and drain valves are not required to be OPERABLE in these MODES since the reactor is subcritical and only one rod may be withdrawn and subject to scram.
ACTIONS	The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each SDV vent and

(continued)

BASES

ACTIONS (continued)

drain line. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable SDV line. Complying with the Required Actions may allow for continued operation, and subsequent inoperable SDV lines are governed by subsequent Condition entry and application of associated Required Actions.

A.1

When one SDV vent or drain valve is inoperable in one or more lines, the valves must be restored to OPERABLE status within 7 days. The Completion Time is reasonable, given the level of redundancy in the lines and the low probability of a scram occurring while the valve(s) are inoperable. The SDV is still isolable since the redundant valve in the affected line is OPERABLE. During these periods, the single failure criterion may not be preserved, and a higher risk exists to allow reactor water out of the primary system during a scram.

B.1

If both valves in a line are inoperable, the line must be isolated to contain the reactor coolant during a scram. When a line is isolated, the potential for an inadvertent scram due to high SDV level is increased. Required Action B.1 is modified by a Note that allows periodic draining and venting of the SDV when a line is isolated. During these periods, the line may be unisolated under administrative control. This allows any accumulated water in the line to be drained, to preclude a reactor scram on SDV high level. This is acceptable since the administrative controls ensure the valve can be closed quickly, by a dedicated operator, if a scram occurs with the valve open.

The 8 hour Completion Time to isolate the line is based on the low probability of a scram occurring while the line is not isolated and unlikelihood of significant CRD seal leakage.

(continued)

BASES

ACTIONS (continued)

C.1

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 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.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

During normal operation, the SDV vent and drain valves should be in the open position (except when performing SR 3.1.8.2) to allow for drainage of the SDV piping. Verifying that each valve is in the open position ensures that the SDV vent and drain valves will perform their intended functions during normal operation. This SR does not require any testing or valve manipulation; rather, it involves verification that the valves are in the correct position. This SR is modified by a Note that allows the SR to be met for OPERABLE valves that are temporarily closed while performing the testing required by SR 3.1.8.2. The Note is necessary to avoid potential conflicts between the two SRs created by SR 3.0.1.

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

SR 3.1.8.2

During a scram, the SDV vent and drain valves should close to contain the reactor water discharged to the SDV piping. Cycling each valve through its complete range of motion (closed and open) ensures that the valve will function properly during a scram. The Frequency is based on

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.2 (continued)

operating experience and takes into account the level of redundancy in the system design as well as being in accordance with the Inservice Testing Program.

SR 3.1.8.3

SR 3.1.8.3 is an integrated test of the SDV vent and drain valves to verify total system performance. After receipt of a simulated or actual scram signal, the closure of the SDV vent and drain valves is verified. The closure time of 30 seconds after receipt of a scram signal is based on the bounding leakage case evaluated in the accident analysis (Ref. 2). Similarly, after receipt of a simulated or actual scram reset signal, the opening of the SDV vent and drain valves is verified. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1 and the scram time testing of control rods in LCO 3.1.3 overlap this Surveillance to provide complete testing of the assumed safety function. The 24 month Frequency 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 these components usually pass the Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 4.6.2.3.
2. 10 CFR 100.
3. NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," August 1981.

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 the fuel design limits identified in Reference 1 are not exceeded during Abnormal Operational Transients and that the Peak Cladding Temperature (PCT) during the postulated design basis Loss Of Coolant Accident (LOCA) does not exceed the limits specified in 10 CFR 50.46.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 1 and 2. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), anticipated operational transients, and normal operation that determine the APLHGR limits are presented in References 1, 2, 3, 4, 5, and 6.

Fuel design evaluations are performed to demonstrate that the 1% limit on the fuel cladding plastic strain and other fuel design limits described in Reference 1 are not exceeded during Abnormal Operational Transients for operation with LHGRs up to the operating limit LHGR. APLHGR limits are equivalent to the LHGR limit for each fuel rod divided by the local peaking factor of the fuel assembly. APLHGR limits are developed as a function of exposure and the various operating core flow and power states to ensure adherence to fuel design limits during the limiting Abnormal Operational Transients (Refs. 4, 5, and 6). The actual APLHGR values for the GE fuel design are lattice-type dependent and are explicitly modeled by the plant process computer. The lattice-type dependent values can be found in Reference 10. The Core Operating Limits Report (COLR) APLGHR limit is a nominal representation of the lattice-dependent values, (i.e., the most limiting lattice-type, other than the natural uranium bundle ends), which can be used to conservatively model the APLGHR limit if the plant process computer becomes unavailable. Flow dependent APLHGR limits are determined using the three dimensional BWR simulator

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

code (Ref. 7) to analyze slow flow runout transients. The flow dependent multiplier, $MAPFAC_f$, is dependent on the maximum core flow runout capability. The maximum runout flow is dependent on the existing setting of the core flow limiter in the Recirculation System.

Based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions, power dependent multipliers, $MAPFAC_p$, are also generated. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which turbine stop valve closure and turbine control valve fast closure scram trips are bypassed (approximately 30% RTP), both high and low core flow $MAPFAC_p$ limits are provided for operation at power levels between 25% RTP and the previously mentioned bypass power level (30% RTP). The exposure dependent APLHGR limits are reduced by $MAPFAC_p$ and $MAPFAC_f$ at various operating conditions to ensure that all fuel design criteria are met for normal operation and Abnormal Operational Transients. A complete discussion of the analysis code is provided in Reference 8.

LOCA analyses are then performed to ensure that the above determined APLHGR limits are adequate to meet the PCT and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. A complete discussion of the analysis is provided in Reference 9. 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 an assembly. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by its local peaking factor. A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.

For single recirculation loop operation, the MAPFAC multiplier is limited due to the conservative analysis assumption of an earlier departure from nucleate boiling with one recirculation loop available, resulting in a more severe cladding heatup during a LOCA (Ref. 4).

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

(continued)

BASES (continued)

LCO

The APLHGR limits specified in the COLR are the result of the fuel design, DBA, and transient analyses. For two recirculation loops operating, the limit is determined by multiplying the smaller of the $MAPFAC_p$ and $MAPFAC_f$ factors times the exposure dependent APLHGR limits. With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating", the limit is determined by multiplying the exposure dependent APLHGR limit by the smaller of either $MAPFAC_p$, $MAPFAC_f$, and a specific single recirculation loop multiplier (Ref. 4).

APPLICABILITY

The APLHGR limits are primarily derived from fuel design evaluations and LOCA and transient analyses that are assumed to occur at high power levels. Design calculations (Ref. 6) and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor scram function, in addition to the average power range monitor scram in Startup, provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, 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 and transient analyses 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 and within design limits of the fuel rods. 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 transient or 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 because in general, a power reduction from full power would normally have already been initiated as part of Required Action A.1.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. They 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 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel" (latest approved version).
2. UFSAR, Chapter 4.3.1.
3. UFSAR, Chapter 15.6.6.
4. NEDO-24272, Duane Arnold Energy Center Single-Loop Operation, Supplemented by DAEC Supplement to NEDC-32915P, "Duane Arnold Energy Center GE12 Fuel Upgrade Project", APED L12-003, March 2000.

(continued)

BASES

REFERENCES (continued)

5. NEDC-30626. General Electric Boiling Water Reactor Extended Load Line Limit Analysis for Duane Arnold Energy Center.

6. NEDC-30813-P, Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program for the Duane Arnold Energy Center.

7. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.

8. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.

9. NEDC-31310P, Duane Arnold Energy Center SAFER/GESTR-LOCA Loss of Coolant Accident Analysis, Supplemented by NEDC-32915P, Duane Arnold Energy Center GE12 Fuel Upgrade Project, November 1999.

10. NEDE-31152P, “GE Fuel Bundle Designs”, (latest version referenced in COLR).

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 transition boiling to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods avoid transition boiling 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 Abnormal Operational Transients. Although fuel damage does not necessarily occur if a fuel rod actually experienced transition boiling (Ref. 1), the critical power at which transition boiling 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 Abnormal Operational Transients to establish the operating limit MCPR are presented in References 2, 3, 4, 5, 6, and 7. 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 either reactor pressure or power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR).

The minimum Operating Limit MCPR bounds the sum of the Safety Limit MCPR and the largest ΔCPR. Values for the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Operating Limit MCPR are contained in the COLR.

The required MCPRs at rated power are determined using the GEMINI transient analysis methods described in Reference 2. These limits (ODYN Option A) were derived by using the GE 67B scram times, given in ITS 3.1, which are based upon extensive operating plant data, as well as GE test data. The ODYN Option B scram insertion times were statistically derived from the 67B data to ensure that the resulting Operating Limit from the transient analysis would, with 95% probability at the 95% confidence level, result in the Safety Limit MCPR not being exceeded. The scram time parameter τ is a measure of the conformance of the actual plant control rod drive performance to that used in the ODYN Option B licensing basis.

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state ($MCPR_f$ and $MCPR_p$, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Refs. 5, 6, and 7). The $MCPR_f$s were calculated such that, for the maximum core flow rate and core thermal power along a conservative load line, the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit MCPR. Using this relative bundle power, the MCPRs were calculated at different points along this conservative load line corresponding to different core flows. Flow dependent MCPR limits are determined by steady state thermal hydraulic methods with key physics response inputs benchmarked using the three dimensional BWR simulator code (Ref. 8) to analyze slow flow runout transients. At less than 100% of rated power and/or flow, the required Operating Limit MCPR is the larger value of the flow-dependent MCPR ($MCPR_f$) or the power-dependent multiplier (K_p) times the rated power MCPR at the existing core power/flow state. The required Operating Limit MCPR is a function of flow in order to protect the fuel from inadvertent core flow increases such that the Safety Limit MCPR requirement can be assured. The operating limit is dependent on the maximum core flow limiter setting in the Recirculation System. The resulting $MCPR_f$s are given in the COLR.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Power dependent MCPR limits ($MCPR_p$) are determined mainly by the one dimensional transient code (Ref. 9). This limit protects the fuel from Abnormal Operational Transients, including localized events, such as a rod withdrawal error, other than those resulting from inadvertent core flow increases, which are covered by the flow-dependent MCPR limits. This power-dependent MCPR limit was developed based upon bounding analyses for the most limiting transient at the given core power level. Further information on the MCPR operating limits for off-rated conditions is presented in Reference 7 and the COLR. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scrams are bypassed (approximately 30% RTP), high and low flow $MCPR_p$ operating limits are provided for operating between 25% RTP and the previously mentioned bypass power level (30% RTP).

During SLO, the Operating Limit MCPR must be increased to account for the increased uncertainties used in the statistical analyses to derive the Safety Limit MCPR (see Reference 5 and the COLR).

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

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 $MCPR_f$ and $MCPR_p$ 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. Statistical analyses indicate that the nominal value of the initial MCPR expected at 25% RTP is > 3.5. Studies of the variation of limiting transient

(continued)

BASES

APPLICABILITY (continued)

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 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 because in general, a power reduction from full power would normally have already been initiated as part of Required Action A.1.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. It 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 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

SR 3.2.2.2

Because the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. Therefore, in order to perform SR 3.2.2.2, the value of T, which is a measure of the actual scram speed distribution compared with the assumed distribution must first be determined. The MCPR operating limit is then determined based on an interpolation between the applicable limits for Option A (scram times of LCO 3.1.4,"Control Rod Scram Times") and Option B (realistic scram times) analyses. The parameter T must be determined once within 72 hours after each set of scram time tests required by SR 3.1.4.1 and SR 3.1.4.2 because the effective scram speed distribution may change after maintenance that could affect scram times. The 72 hour Completion Time is acceptable due to the large inherent margin to operating limits at low power.

REFERENCES

1. NUREG-0562, June 1979.
2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version).
3. UFSAR, Chapters 4.2.3, 4.4.2, and 4.4.4.
4. UFSAR, Chapter 15.4.

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BASES

REFERENCES (continued)

5. NEDO-24272, Duane Arnold Energy Center Single-Loop Operation, Supplemented by DAEC Supplement to NEDC-32915P, "Duane Arnold Energy Center GE12 Fuel Upgrade Project", APED L12-003, March 2000.

6. NEDC-30626, General Electric Boiling Water Reactor Extended Load Line Limit Analysis for Duane Arnold Energy Center.

7. NEDC-30813-P Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program for the Duane Arnold Energy Center.

8. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.

9. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.

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 Pressure Boundary (RCPB), minimize the energy that must be absorbed following a Loss of Coolant Accident (LOCA), and prevent inadvertent criticality. 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) (Ref. 11).

The RPS, as shown in the UFSAR, Figure 7.2-2 (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, Electro Hydraulic Control (EHC) 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 and the Manual Scram signal, which have only two input signals). Most channels include on-off sensors, bi-stable trip circuits, or trip units that compare measured input signals with pre-established setpoints. When the setpoint is exceeded, the sensor/bi-stable/trip unit

(continued)

BASES

BACKGROUND (continued)

acutates the channel output relay, which then outputs an RPS trip signal to the trip logic.

The RPS is comprised of two independent trip systems (A and B) with three logic channels in each trip system (automatic logic channels A1 and A2, B1 and B2, and manual logic channels A3 and B3) 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; however, this time delay is not a safety-related function.

Two scram pilot valves are located in the hydraulic control unit for each Control Rod Drive (CRD). Each scram pilot valve is solenoid 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 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.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The actions of the RPS are assumed in the safety analyses of References 1, 2, and 3. The RPS initiates a reactor scram prior to when monitored parameter values exceed 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 RCPB, and the containment by minimizing the energy that must be absorbed following a LOCA.

RPS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). 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, where specified.

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 exceeds the setpoint, the associated device (e.g., on-off sensor, bi-stable trip circuit, or trip unit) changes state. Analytical Limits, where established, are the limiting values of the process parameters used in safety analysis to define the margin to unacceptable consequences. Margin is provided between the Allowable Value and the Analytical Limits to allow for

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

process, calibration (i.e., M&TE) and some instrument uncertainties. Additional margin is provided between the Allowable Value and the trip setpoint to allow for the remaining instrument uncertainties (e.g., drift). The Trip Setpoints derived 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 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 or other specified conditions 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 MODES 1 and 2 and 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 specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

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 damage resulting from Abnormal Operational 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. 2). While no direct credit is taken in the accident analysis for this function, the IRM does provide 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 groups of IRM channels, with three IRM channels inputting to each trip system. The analysis of Reference 3 assumes that one channel in each trip system is bypassed. Therefore, four channels with two 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 125 divisions of a 125 division scale.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

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

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 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. The IRM trips are automatically bypassed when the reactor mode switch is placed in RUN.

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.

Four channels of Intermediate Range Monitor - Inop with two 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

2.a. Average Power Range Monitor Neutron Flux - Upscale, Startup

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 125% of RTP. For operation at low power (i.e., MODE 2), the Average Power Range Monitor Neutron Flux - Upscale, Startup Function is capable of generating a trip signal that prevents fuel damage resulting from Abnormal Operational Transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux - Upscale, Startup Function will, in reality, provide a secondary scram to the Intermediate Range Monitor Neutron Flux - High Function because of their relative setpoints. However, with the IRMs on Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux - Upscale, Startup Function will provide the primary trip signal for a corewide increase in power.

No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux - Upscale, Startup Function (the Control Rod Drop Accident (CRDA) analysis conservatively assumes the Average Power Range Monitor High Valve clamp provides this trip to mitigate the CRDA). However, this 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.

The APRM System is divided into two groups of channels with three APRM channel inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Neutron Flux - Upscale, Startup with two channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal. In addition, to

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Average Power Range Monitor Neutron Flux-Upscale, Startup (continued)

provide adequate coverage of the entire core, at least 9 LPRM inputs for APRM channels A, B, C, and D, and 13 LPRM inputs for APRM channels E and F are required. In addition, at least two LPRM inputs from each of the four axial levels at which the LPRMs are located are required for each APRM channel.

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

The Average Power Range Monitor Neutron Flux-Upscale, Startup Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists.

Other than when required by the Shutdown Margin Test - Refueling LCO (LCO 3.10.8), the Average Power Range Monitor Neutron Flux - Upscale, Startup Function need not be OPERABLE in MODE 5, as adequate protection against neutron flux excursions is maintained by the Refueling Interlocks and IRMs.

In MODE 1, the Average Power Range Monitor High Value Clamp Function provides protection against reactivity transients and the RWM and rod block monitor protect against control rod withdrawal error events.

2.b. Average Power Range Monitor Flow Biased-High

The Average Power Range Monitor Flow Biased-High Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. 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). While the DAEC design basis accident analysis of Abnormal Operational Transients takes no credit for the Average Power Range Monitor Flow Biased-High Function to mitigate those events, a special event does take credit for this RPS Function. The Average Power Range Monitor Flow Biased-High Function does provide protection against transients at lower power and core flow levels where the possibility of thermal-hydraulic

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor Flow Biased-High (continued)

instabilities exist (e.g., dual recirculation pump trip). If a thermal-hydraulic instability occurs, the Average Power Range Monitor Flow Biased-High Function will initiate a scram before the Average Power Range Monitor-High Value Clamp, thus protecting the fuel cladding integrity by ensuring that the MCPR Safety Limit is not exceeded (Ref. 12).

The APRM System is divided into two groups of channels with three APRM inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Flow Biased-High 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. In addition, to provide adequate coverage of the entire core, at least 9 LPRM inputs for APRM channels A, B, C, and D, and 13 LPRM inputs for APRM channels E and F are required. In addition, at least two LPRM inputs from each of the four axial levels at which the LPRMs are located are required for each APRM channel. Each APRM channel receives two total drive flow signals representative of total core flow. The total drive flow signals are generated by four flow units, two of which supply signals to the trip system A APRMs, while the other two supply signals to the trip system B APRMs. Each flow unit signal is provided by summing up the flow signals from the two recirculation loops. To obtain the most conservative reference signals, the total flow signals from the two flow units (associated with a trip system as described above) are routed to a low auction circuit associated with each APRM. Each APRM's auction circuit selects the lower of the two flow unit signals for use as the scram trip reference for that particular APRM. Each required Average Power Range Monitor Flow Biased-High channel only requires an input from one OPERABLE flow unit, since the individual APRM channel will perform the intended function with only one OPERABLE flow unit input. However, in order to maintain single failure criteria for the Function, at least one required Average Power Range Monitor Flow Biased-High channel in each trip system must be

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor Flow Biased-High (continued)

capable of maintaining an OPERABLE flow unit signal in the event of a failure of an auction circuit, or a flow unit, in the associated trip system. The flow units are assigned to bypass switches in the same manner as they are distributed in the APRM System. The bypass switch arrangement prevents the bypassing of more than one flow unit supplying signals to the Rod Block circuitry. If a flow unit is inoperable, it may be bypassed; however, the bypass function does not remove the flow unit's output to the APRM circuit. When a flow unit is placed in a position other than "operate" or "standby", its output to the low auction circuit will be maximum, making the low auction circuit select the input from the operating flow unit. If both flow units associated with an APRM channel are in positions other than "operate" or "standby", that APRM channel shall be considered inoperable.

The Allowable Values are calculated from the nominal trip setpoint straight line equations (e.g., 0.58W + 62% RTP for two-loop operation and 0.58W + 58.5% RTP for single-loop operation) at discrete flow points. Because the instrument uncertainties are non-linear, the Allowable Values do not lie in a straight line and do not lend themselves to being characterized by a simple equation. The Allowable Values are in accordance with the DAEC Setpoint Methodology. These Allowable Values are derived from analyses that take credit for the Average Power Range Monitor Flow Biased-High Function for the mitigation of possible thermal-hydraulic instabilities.

Resetting of APRM Flow Biased Scram Setpoint High for single recirculation loop operation (SLO) may be accomplished by either adjusting the APRM gain such that indicated power is 3.5% higher than actual or by recalibrating the APRMs to lower the setpoint by 3.5%. Adjusting the gain is a simple procedure where the APRM output is adjusted to indicate core power plus 3.5%. Full recalibration to change the actual trip setpoint is more complicated and time consuming. Adjusting the gain can be accomplished in a matter of minutes but requires readjusting the gain whenever core power is changed. The full recalibration method also requires a second recalibration to restore the original setpoint when the plant re-enters two recirculation loop

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor Flow Biased-High (continued)

operation. For these reasons, when the plant enters SLO, the APRM gain is initially adjusted to lower the setpoints as required by LCO 3.4.1, "Recirculation Loops Operating." Continued adjustment of APRM gains is acceptable if it is expected that the plant will operate in SLO for a limited period of time. If the plant is expected to be in SLO for an extended period of time, the APRM setpoints should be fully recalibrated to eliminate the need to adjust APRM gain whenever power is changed.

The Average Power Range Monitor Flow Biased–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 High Value Clamp

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases. The Average Power Range Monitor High Value Clamp Function is capable of generating a trip signal to prevent fuel damage or excessive RCS pressure and is credited in the DAEC accident and transient analyses for mitigating these events. In particular, the overpressurization protection analysis of Reference 4, the Average Power Range Monitor High Value Clamp Function is assumed to terminate the Main Steam Isolation Valve (MSIV) closure event and, along with the Safety Relief Valves (SRVs), limits 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 High Value Clamp Function to terminate the CRDA.

The APRM System is divided into two groups of channels with three APRM channels inputting to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor High Value Clamp with two

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.c. Average Power Range Monitor High Value Clamp (continued)

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. In addition, to provide adequate coverage of the entire core, at least 9 LPRM inputs for APRM channels A, B, C, and D, and 13 LPRM inputs for APRM channels E and F are required. In addition, at least two LPRM inputs from each of the four axial levels at which the LPRMs are located are required for each APRM channel.

The Allowable Value is based on the Analytical Limit assumed in the analysis of Abnormal Operational Transients.

The Average Power Range Monitor High Value Clamp 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 High Value Clamp Function is assumed in the CRDA analysis, which is applicable in MODE 2, the Average Power Range Monitor Neutron Flux–Upscale, Startup Function conservatively bounds the assumed trip and, together with the assumed IRM trips, provides adequate protection. Therefore, the Average Power Range Monitor High Value Clamp Function is not required in MODE 2.

2.d. Average Power Range Monitor–Inop

This signal provides assurance that a minimum number of APRMs are OPERABLE. Anytime an APRM mode switch is moved to any position other than "Operate," an APRM module is unplugged, the electronic operating voltage is low, or the APRM has too few LPRM inputs (< 9 for APRM Channels A, B, C, and D, and < 13 for APRM channels E and F), an inoperative trip signal will be received by the RPS, unless the APRM is bypassed. Since only one APRM in each trip system may be bypassed, only one APRM in each 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

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

Four channels of Average Power Range Monitor - Inop with two channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal.

There is no Allowable Value for this Function.

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

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. No specific safety analysis takes direct credit for this Function. However, the Reactor Vessel Steam Dome Pressure - High Function initiates a scram for transients that results 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 scram on the Average Power Range Monitor High Value Clamp signal, not 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 switches 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

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

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

scram from this Function on a valid signal. The Function is 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

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 the single low reactor vessel water level to substantially reduce the heat generated in the fuel from fission. The Reactor Vessel Water Level - Low Function is assumed in the analysis of the Design Basis Accident (DBA-LOCA) (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 signals are initiated from four level switches 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 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 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 will not be required.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPICABILITY

4. Reactor Vessel Water Level—Low (continued)

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 and Low Low Low 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 High Value Clamp Function, along with the SRVs, 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 7 (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 one position switch that provides an input to both the RPS A and B trip systems. There are a total of eight MSIV closure channels, with four channels in each trip system. Each channel receives input from an inboard MSIV and an outboard MSIV. The logic for the Main Steam Isolation Valve—Closure Function is arranged such that either the inboard or outboard valve on two main steam lines closing may produce a half scram, while the closing of either the inboard or outboard valves on three or more of the main steam lines will always produce a full scram.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5. Main Steam Isolation Valve-Closure (continued)

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.

Eight channels of the Main Steam Isolation Valve-Closure Function, with four 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. Therefore, this 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 a secondary scram signal to Reactor Vessel Water Level-Low for LOCA events inside the drywell, and credit is taken for a scram initiated from this Function in the analysis of the containment response to a LOCA. This Function is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

High drywell pressure signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to ensure that the primary containment design pressure is not exceeded upon the occurrence of a DBA LOCA inside primary containment, while the nominal trip setpoint is selected to avoid spurious trips due to non-LOCA events.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6. Drywell Pressure - High (continued)

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.

7a, 7b. 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 UFSAR. However, they are retained to ensure the RPS 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 thermal probes 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 thermal probe 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

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7a, 7b. Scram Discharge Volume Water Level - High (continued)

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 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 7. This Function also actuates the End of Cycle Recirculation Pump Trip (EOC-RPT) System. 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 (or appropriate MCPR penalty with the EOC-RPT System inoperable), 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 two TSVs closing may produce a half scram, while the closing of three or more TSVs will always produce a full scram. This Function must be enabled at THERMAL POWER ≥ 30% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure. Because an increase in the main turbine bypass flow can affect this Function nonconservatively (THERMAL POWER is derived from turbine first stage pressure), the main turbine bypass valves must not cause the trip Functions to be bypassed in order to maintain this Function OPERABLE when THERMAL POWER is ≥ 30%.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

8. Turbine Stop Valve-Closure (continued)

In addition, other steam loads, such as second stage reheaters in operation below 30% RTP, must be accounted for in establishing this setpoint. Otherwise, turbine first stage pressure would be non-conservative with respect to the 30% RTP RPS bypass. 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 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 High Value Clamp Functions are adequate to maintain the necessary safety margins.

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 7. This Function also actuates the EOC-RPT System. The generator load rejection event analysis actually uses a response time for the TCV fast closure (i.e., from event initiation to the start of the TCV closure ≤ 30 msec). The setting for the pressure switch corresponds to the response time. 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 (or appropriate MCPR penalty with the EOC-RPT System inoperable) ensures that the MCPR SL is not exceeded.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low (continued)

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 transmitter 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 normally accomplished automatically by pressure switches sensing turbine first stage pressure. Because an increase in the main turbine bypass flow can affect this Function nonconservatively (THERMAL POWER is derived from turbine first stage pressure), the main turbine bypass valves must not cause the trip Functions to be bypassed in order to maintain this Function OPERABLE when THERMAL POWER is ≥ 30% RTP. In addition, other steam loads, such as second stage reheaters in operation below 30% RTP, must be accounted for in establishing this setpoint. Otherwise, turbine first stage pressure would be non-conservative with respect to the 30% RTP RPS bypass.

The Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Allowable Value is selected high enough to detect imminent TCV fast closure and corresponds to the response time (≤ 30 msec) used in the transient analysis.

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 High Value Clamp Functions are adequate to maintain the necessary safety margins.

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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 the scram valve pilot solenoids. The manual scram logic channels (A3 and B3) are separate from 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 two channels, each of which provides input into one of the RPS manual scram logic channels.

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

Two channels of Reactor Mode Switch - Shutdown Position Function, with one channel 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 the scram valve pilot solenoids. The manual scram logic channels (A3 and B3) are separate from 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 manual scram logic channels. In order to cause a scram it is necessary that the 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.

Two channels of Manual Scram with one channel in each trip system 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 (Ref. 9) 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 condition per Required Actions A.1 and A.2. Placing the inoperable channel in trip (or the associated trip system in

(continued)

ACTIONS

A.1 and A.2 (continued)

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.

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

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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, isolation, or RPT), Condition D must be entered and its Required Action taken.

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 and the IRM and APRM Functions, 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 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 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

(continued)

BASES

ACTIONS (continued)

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, and G.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 Action E.1 is 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 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.

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BASES (continued)

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 (Ref. 9) 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.

SR 3.3.1.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between 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 are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.1 (continued)

channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.1.2

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 3.4.1, "Recirculation Loops Operating," allows the APRMs to be reading greater than actual THERMAL POWER to effectively lower the APRM Flow Biased High setpoints by 3.5% for single recirculation loop operation. 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 power plus 3.5%. The Frequency of once per 24 hours 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 and APLHGR). At ≥ 25% RTP, the Surveillance is required to have been satisfactorily performed within the previous 24 hours, in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above 25% if the 24 hour 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 consideration of providing a reasonable time in which to complete the SR.

SR 3.3.1.1.3

The surveillance frequency extensions for various RPS functions are permitted by Reference 9, provided the automatic scram contactors are functionally tested weekly. There are four pairs of RPS automatic scram contactors (i.e., K14 relay contacts) with each pair associated with an automatic scram logic (A1, A2, B1, and B2). The automatic scram contactors can be functionally tested without the

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.3 (continued)

necessity of using an automatic scram function trip. This functional test can be accomplished by placing the associated RPS Test Switch in the trip position, which will deenergize a pair of the automatic scram contactors and in turn, trip the associated RPS logic. The RPS Test Switches were not specifically credited in the accident analysis and thus, do not have any OPERABILITY requirements of their own. However, because the Manual Scram pushbuttons at the DAEC are not configured the same as the generic model used in Reference 9, (i.e., they are in a separate RPS logic - A3 and B3), the RPS Test Switches have been found to be functionally equivalent to the Manual Scram pushbuttons in the generic model for performing the weekly functional test of the automatic scram contactors required by Reference 9. If an RPS Test Switch(es) is (are) not available for performing this test, it is permissible to take credit for a CHANNEL FUNCTIONAL TEST of an automatic RPS trip function (i.e., SR 3.3.1.1.9), if performed within the required Frequency for this Surveillance, as it will also test the K14 relay contacts.

The Frequency of 7 days is based upon the reliability analysis in Reference 9.

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. The CHANNEL FUNCTIONAL TEST verifies acceptable response by verifying the change of state of at least one contact on the relay which inputs into the trip logic. The required contacts not tested during the CHANNEL FUNCTIONAL TEST are tested under the LOGIC SYSTEM FUNCTIONAL TEST. This is acceptable because operating experience shows that the contacts not tested during the CHANNEL FUNCTIONAL TEST normally pass the LOGIC SYSTEM FUNCTIONAL TEST.

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 and APRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.4 (continued)

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 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. The CHANNEL FUNCTIONAL TEST verifies acceptable response by verifying the change of state of at least one contact on the relay which inputs into the trip logic. The required contacts not tested during the CHANNEL FUNCTIONAL TEST are tested under the LOGIC SYSTEM FUNCTIONAL TEST. This is acceptable because operating experience shows that the contacts not tested during the CHANNEL FUNCTIONAL TEST normally pass the LOGIC SYSTEM FUNCTIONAL TEST. A Frequency of 7 days provides an acceptable level of system average availability over the Frequency and is based on the reliability analysis using the concepts developed in Reference 10.

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. This is required prior to withdrawing SRMs from the fully inserted position since indication is being transitioned from the SRMs to the IRMs.

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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.6 and SR 3.3.1.1.7 (continued)

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 (i.e., approximately one-half decade of range). Overlap between SRMs and IRMs similarly exists when, prior to withdrawing the SRMs from the fully inserted position, IRMs are indicating at least 5/40 on range 1 before SRMs have reached 10^6 counts per second.

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 using analytical methods with input from the axial flux profiles measured by the Traversing Incore Probe (TIP) System. This establishes the relative local flux profile for appropriate representative input to the APRM System. The 1000 MWD/T Frequency is based on operating experience with LPRM sensitivity changes.

SR 3.3.1.1.9 and SR 3.3.1.1.13

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. The CHANNEL FUNCTIONAL TEST verifies acceptable response by verifying the change of state of at least one contact on the relay which inputs into the trip logic.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.9 and SR 3.3.1.1.13 (continued)

The required contacts not tested during the CHANNEL FUNCTIONAL TEST are tested under the LOGIC SYSTEM FUNCTIONAL TEST. This is acceptable because operating experience shows that the contacts not tested during the CHANNEL FUNCTIONAL TEST normally pass the LOGIC SYSTEM FUNCTIONAL TEST. The 92 day Frequency of SR 3.3.1.1.9 is based on the reliability analysis of Reference 9.

The 24 month Frequency 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

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.1.1-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than accounted for in the appropriate setpoint methodology. The Frequency of 92 days is based on the reliability analysis of Reference 9.

SR 3.3.1.1.11, SR 3.3.1.1.12 and SR 3.3.1.1.14

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test 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. The CHANNEL CALIBRATION for Functions 5 and 8 shall consist of the physical inspection and actuation of these position switches.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.11, SR 3.3.1.1.12 and SR 3.3.1.1.14 (continued)

Note 1 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 calorimetric calibration (SR 3.3.1.1.2) every 24 hours and the 1000 MWD/T LPRM calibration against the TIPs (SR 3.3.1.1.8). A second Note is provided that requires the APRM and 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.

The Frequency of SR 3.3.1.1.11 is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.12 is based upon the assumption of a 184 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.14 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 3.3.1.1.15

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

The 24 month Frequency 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint. Because main turbine bypass flow, as well as other turbine steam loads, can affect this setpoint nonconservatively (THERMAL POWER is derived from turbine first stage pressure), the main turbine bypass valves must remain closed (except during required testing or upon actual demand) at THERMAL POWER ≥ 30% RTP to ensure that the calibration remains valid. If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ 30% RTP, either due to open main turbine bypass valve(s) or other reasons, such as changes in turbine steamload to the Main Steam Reheaters), 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

The Average Power Range Monitor Flow Biased-High Function uses the recirculation loop drive flows to vary the trip setpoint. This SR ensures that the total loop drive flow signals from the flow units used to vary the setpoint is appropriately compared to a calibrated flow signal and, therefore, the APRM Function accurately reflects the required setpoint as a function of flow. Each flow signal from the respective flow unit must be ≤ 110% of the calibrated flow signal. If the flow unit signal is not within the limit, that flow unit may be bypassed, and its output to the low auction circuit will be maximum, making the low auction circuit select the input from the operating flow unit.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.17 (continued)

The Frequency of 24 months is based on engineering judgment, operating experience, the reliability of this instrumentation, the other surveillances performed on the components of the flow biasing network, and the fact that a half scram will be present for an extended period of time during the performance of this surveillance.

SR 3.3.1.1.18

This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. The RPS Response Time test only applies to the Functions of Reactor Vessel Water Level - Low and Reactor Vessel Steam Dome Pressure - High. These RPS Functions are the only ones that were identified, in a program conducted prior to the first refueling outage, that require sensor response time testing. This test may be performed in one measurement or in overlapping segments, with verification that all components are tested. The RPS RESPONSE TIME acceptance criteria are included in Reference 13.

RPS RESPONSE TIME tests are conducted on a 24 month STAGGERED TEST BASIS. 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 3.3.1.1.19

This SR ensures that the RPS logic system response times are less than or equal to the maximum value assumed in the accident analysis. The RPS logic system response time test is measured from the opening of the sensor contact up to and including the opening of the trip actuator contacts. As such, this test does not include the sensor response time. All RPS Functions except the RPS Manual Scram and Reactor Mode Switch - Shutdown Position are included in this test.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.19 (continued)

These two RPS Functions are excluded since they directly trip their scram solenoid relays without any intervening devices, thus there is nothing to response time test. This test may be performed in one measurement or in overlapping segments, with verification that all components are tested. The RPS logic system response time acceptance criteria are included in Reference 13. RPS logic system response time tests are conducted on a 24 month STAGGERED TEST BASIS. 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.

REFERENCES

1. UFSAR, Figure 7.2-2.

2. UFSAR, Section 15.4.2.

3. NEDO-23842, "Continuous Control Rod Withdrawal in the Startup Range," April 18, 1978.

4. UFSAR, Section 5.2.2.

5. UFSAR, Section 15.4.7.

6. UFSAR, Section 6.3.3.

7. UFSAR, Chapter 15.

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. Reliability of Engineered Safety Features as a Function of Testing Frequency, Volume 9, No. 4, July-August 1968.

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BASES

REFERENCES (continued)

11. DBD-A61-005, Setpoints and Margins Topical Design Bases Document

12. NEDC-32915P, "Duane Arnold Energy Center GE12 Fuel Upgrade Project", November 1999.

13. UFSAR 7.2.1.3

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 very 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 maintained fully inserted until the count rate is greater than a minimum allowed count rate (a control rod block is set at this condition). After SRM to Intermediate Range Monitor (IRM) overlap is demonstrated (as required by SR 3.3.1.1.6), 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 (i.e., "dunking chambers") 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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

System (RPS) Instrumentation"; IRM Neutron Flux-High and Average Power Range Monitor (APRM) Neutron Flux- Upscale, Startup 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 UFSAR 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 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 IRM 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 is not required to be OPERABLE, since it is not capable of monitoring neutron flux in the fueled region of the core. 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

(continued)

BASES

LCO (continued)

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 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 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 MODE 2 (prior to the IRMs being on scale on Range 3), and MODES 3, 4, and 5 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.

(continued)

BASES

ACTIONS

A.1 and B.1 (continued)

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 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 flux monitoring capability by the IRMs. 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 below IRM Range 3, if the required number of SRMs are 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.

(continued)

BASES

ACTIONS (continued)

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 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 SRMs occurring during this interval.

E.1 and E.2

With one or more required SRMs 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 fully 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

As noted at the beginning of the SRs, the SRs for each SRM Applicable MODE or other specified conditions are found in the SRs column of Table 3.3.1.2-1.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

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

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

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, but more frequent, 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 when the fueled region encompasses more than one SRM, 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. 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2.2 (continued)

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 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 with the detector fully inserted into the core. The requirement of at least 3 cps assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of RTP which is used in the analysis of transients in cold conditions. With few fuel assemblies loaded, the SRMs may 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 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

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. The CHANNEL FUNCTIONAL TEST verifies acceptable response by verifying the change of state of at least one contact on the relay which inputs into the trip logic. The required contacts not tested during the CHANNEL FUNCTIONAL TEST are tested under the LOGIC SYSTEM FUNCTIONAL TEST. This is acceptable because operating experience shows that the contacts not tested during the CHANNEL FUNCTIONAL TEST normally pass the LOGIC SYSTEM FUNCTIONAL TEST. SR 3.3.1.2.5 is 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 in MODES 3 and 4 and core reactivity changes are due solely to control rod movement in MODE 2, 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.

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.

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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. Note 1 to the Surveillance allows the neutron detectors to be 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 None.

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. It is assumed to function to block further control rod withdrawal to preclude a MCPR Safety Limit (SL) violation. 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. A signal from one Average Power Range Monitor (APRM) channel assigned to each Reactor Protection System (RPS) trip system supplies a reference signal for the RBM channel in the same trip system. This reference signal is used to determine which RBM range setpoint (low, intermediate, or high) is enabled. 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. 1).

(continued)

BASES

BACKGROUND (continued)	The purpose of the RWM is to control rod patterns during startup and shutdown, 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 based on position indication for each control rod. The RWM also uses feedwater flow and steam flow signals to determine when the reactor power is above the preset power level at which the RWM is automatically bypassed (Ref. 2). However, the RWM has been physically modified at the DAEC so that it controls rod patterns over the entire operating range, up to 100% RTP (i.e., it is not automatically bypassed above a preset power level). The RWM is a single channel system that provides input into both RMCS rod block circuits. 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	<u>1. Rod Block Monitor</u> The RBM is designed to prevent violation of the MCPR SL and the cladding 1% plastic strain fuel design limit that may result from a single control Rod Withdrawal Error (RWE) event. The analytical methods and assumptions used in evaluating the RWE event are summarized in Reference 3. A statistical analysis of RWE events was performed to determine the RBM response for both channels for each event. From these responses, the fuel thermal performance as a function of RBM Allowable Value was determined. The Allowable Values are chosen as a function of power level. Based on the specified Allowable Values, operating limits are established.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor (continued)

The RBM Function satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

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 from this Function. The actual setpoints are calibrated consistent with the DAEC Instrument Setpoint Methodology. The RBM Downscale Function is not required to have a specified Allowable Value setpoint, since this Function does not affect the Rod Withdrawal Error analysis (Ref. 3).

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., pressure switch) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. Analytical Limits, where established, are the limiting values of the process parameters used in safety analysis to define the margin to unacceptable consequences. Margin is provided between the Allowable Value and the Analytical Limits to allow for process, calibration (i.e., M&TE) and some instrument uncertainties. Additional margin is provided between the Allowable Value and the trip setpoint to allow for the remaining instrument uncertainties (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 is assumed to mitigate the consequences of a RWE event when operating ≥ 30% RTP. Below this power level, the consequences of an RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor (continued)

(Ref. 3). When operating < 90% RTP, analyses (Ref. 3) have shown that with an initial MCPR ≥ 1.70, no RWE event will result in exceeding the MCPR SL. Also, the analyses demonstrate that when operating at ≥ 90% RTP with MCPR ≥ 1.40, no RWE event will result in exceeding the MCPR SL (Ref. 3). Therefore, under these conditions, the RBM is also not required to be OPERABLE.

2. Rod Worth Minimizer

The RWM enforces a rod pattern which is consistent with the Banked Position Withdrawal Sequence (BPWS) to ensure that the initial conditions of the CRDA analysis are not violated. The analytical methods and assumptions used in evaluating the CRDA are summarized in References 4, 5, 6, and 7. 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."

The RWM Function satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

Since the RWM is a hardwired system 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. 7). 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, even though the RWM will enforce compliance with the BPWS when the reactor is at any power level. When THERMAL POWER is > 10% RTP, there is no possible control rod configuration that results in a control rod worth that could

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2. Rod Worth Minimizer (continued)

exceed the 280 cal/gm fuel damage limit during a CRDA (Refs. 5 and 7). In MODES 3 and 4, all control rods are required to be inserted into the core; 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.

3. Reactor Mode Switch - Shutdown Position

During MODES 3 and 4, and during MODE 5 when the reactor mode switch is 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 10 CFR 50.36 (c)(2)(ii).

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, "Refuel Position One-Rod-Out Interlock"), 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

(continued)

BASES

ACTIONS

A.1 (continued)

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 24 hours is based on the low probability of an event occurring coincident with a failure in the remaining OPERABLE channel.

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 1 hour. 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 1 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 if a reactor startup with an inoperable RWM during withdrawal of one or more of the 12 rods was not performed in the last calendar year. These requirements minimize the number of reactor startups initiated with RWM inoperable. 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. 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.

(continued)

BASES

ACTIONS

C.1, C.2.1.1, C.2.1.2, and C.2.2 (continued)

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 trained in accordance with an approved training program.

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. Under these circumstances, movement of control rods is in compliance with the BPWS and Required Action C.2.2 is met, provided the number, type (e.g., stuck, slow, etc.) and separation criteria of Reference 6 are met.

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 trained in accordance with an approved training program. 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

(continued)

BASES

ACTIONS

E.1 and E.2 (continued)

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 second 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 (Ref. 9) 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. The CHANNEL FUNCTIONAL TEST verifies acceptable response by verifying the change of state of at least one contact on the relay which inputs into the trip logic. The required contacts not tested during the CHANNEL FUNCTIONAL TEST are tested under the LOGIC SYSTEM FUNCTIONAL TEST. This is acceptable because operating experience shows that the contacts not tested during the CHANNEL FUNCTIONAL TEST normally pass the LOGIC SYSTEM FUNCTIONAL TEST. It includes the Reactor Manual Control Multiplexing System input.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1.1 (continued)

Testing of the Reactor Manual Control Multiplexing System input shall include inputs of "no rod selected," "peripheral rod selected," and "center rod selected with two, three, or four LPRM strings around it" (Ref. 10).

The Frequency of 92 days is based on reliability analyses (Ref. 8).

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 verifies acceptable response by verifying the change of state of at least one contact on the relay which inputs into the trip logic. The required contacts not tested during the CHANNEL FUNCTIONAL TEST are tested under the LOGIC SYSTEM FUNCTIONAL TEST. This is acceptable because operating experience shows that the contacts not tested during the CHANNEL FUNCTIONAL TEST normally pass the LOGIC SYSTEM FUNCTIONAL TEST. 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 for SR 3.3.2.1.2, and by attempting to select a control rod, in each fully inserted group, not in compliance with the prescribed sequence and verifying a selection error occurs. 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 at ≤ 10% RTP in MODE 2, and 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).

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1.4

The RBM setpoints are automatically varied as a function of power. Three Allowable Values are specified in Table 3.3.2.1-1, each within a specific power range. The power at which the control rod block Allowable Values, which are verified during the CHANNEL CALIBRATION, automatically change are based on the APRM signal's input to each RBM channel. Below the minimum power setpoint, the RBM is automatically bypassed. These power Allowable Values must be verified periodically to be within the specified ranges to ensure that the Analytical Limits for the ranges specified in Table 3.3.2.1-1 are met. If any power range setpoint is nonconservative, then the affected RBM channel is considered inoperable. Alternatively, the power range channel can be placed in the conservative condition (i.e., enabling the proper RBM setpoint). 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 3.3.1.1.7. The 184 day Frequency is based on the actual trip setpoint methodology utilized for these channels.

SR 3.3.2.1.5

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies 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 DAEC Instrument 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 3.3.1.1.7.

The Frequency is based upon the assumption of a 184 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

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 verifies acceptable response by verifying the change of state of at least one contact on the relay which inputs into the trip logic. The required contacts not tested during the CHANNEL FUNCTIONAL TEST are tested under the LOGIC SYSTEM FUNCTIONAL TEST. This is acceptable because operating experience shows that the contacts not tested during the CHANNEL FUNCTIONAL TEST normally pass the LOGIC SYSTEM FUNCTIONAL TEST. 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 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 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

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

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 7.6.1.8.

2. UFSAR, Section 7.7.4.7.

3. NEDC-30813-P, "Average Power Range Monitor, Rod Block Monitor, and Technical Specification Improvement (ARTS) Program for the Duane Arnold Energy Center," December 1984.

4. 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.

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

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

7. 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.

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

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

10. Licensee Event Report #97-07, "Inadequate Testing of the Reactor Mode Switch to Shutdown Position Rod Block Function and Rod Block Monitor."

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 determine that the actions automatically initiated by the Engineered Safety Features (ESF) equipment have successfully accomplished their safety functions for Design Basis Events. The instruments that monitor these variables are designated as non-Type A, Category I, in accordance with Regulatory Guide 1.97 (Ref. 1). Note that since the ESF at the DAEC accomplish their safety functions by automatic control, the DAEC does not have any Type A, Category I variables (Ref. 2).

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 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;
- 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The plant specific Regulatory Guide 1.97 Safety Evaluation Report (Ref. 2) found that DAEC is in conformance with, or is justified in deviating from the guidance of Regulatory Guide 1.97 for each Post Accident Monitoring variable. 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 instruments are important for reducing public risk and satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

LCO 3.3.3.1 requires at least two OPERABLE channels for all applicable Functions (except some Primary Containment Isolation Valve (PCIV) Position Functions, as allowed by Footnotes (a) and (b) to Table 3.3.3.1-1) 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 an accident. Provision for at least two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information.

Exceptions to the two channel requirement are allowed for some PCIV position indications. In these cases, the important information is the status of the primary containment penetrations. The LCO requires two position indicators for each primary containment penetration flow path. In most cases, this requires one position indicator for each PCIV in the penetration, which 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 status. Exceptions are provided by footnotes to Table 3.3.3.1-1, for situations such as when normally active PCIVs are known to be closed and deactivated. For such situations, position indication is not needed to determine status. Therefore, the position indication for valves in these states are not required to be OPERABLE. An exception is also allowed for penetration flow paths with only one installed control room indication channel. For such situations, only one position indication channel is required to be OPERABLE.

(continued)

BASES

LCO (continued)

The following list is a discussion of the specified instrument Functions listed in Table 3.3.3.1-1 in the accompanying LCO.

1. Reactor Steam Dome Pressure

Reactor steam dome pressure is a Category I 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 transmitters with a range of 0 psig to 1500 psig (i.e., wide range) monitor pressure. The pressure transmitters provide input both to pressure indicators and to pen recorders. Wide range indicators and recorders are the primary indication used by the operator during an accident. A channel is considered OPERABLE if either the indicator or recorder is OPERABLE, or if the HPCI or RCIC steam line pressure indicator is OPERABLE, provided at least one channel is continuously recorded. If no channel is recorded, both channels must be declared inoperable and the appropriate Condition entered. Therefore, the PAM Specification deals primarily with this portion of the instrument channel. The HPCI and RCIC steam line pressure indicators (both with a range of 0 psig to 1500 psig) also provide backup means of monitoring reactor steam dome pressure and meet the required qualification requirements for Category I instrumentation.

2. Reactor Vessel Water Level

Reactor vessel water level is a Category I variable provided to support monitoring of core cooling and to verify operation of the ECCS. The wide range water level channels, in combination with the fuel zone level channels, provide the PAM Reactor Vessel Water Level Function. The wide range water level channels measure from 218 inches above the Top of Active Fuel (TAF) down to a point eight inches above the TAF. The fuel zone level channels measure from 218 inches above the TAF to a level below the bottom of the core support plate, or 153 inches below the TAF. Wide range water level is measured by two independent level transmitters. The output from these channels is displayed

(continued)

LCO

2. Reactor Vessel Water Level (continued)

by two independent level indicators and is one of the primary indications used by the operator during an accident. The other primary indication used by the operator during an accident is the fuel zone water level, which is measured by four level transmitters (two of which provide input to one channel with the other two level transmitters providing input to the other channel). The output from each channel is displayed on an independent level indicator and recorded on an independent pen recorder. A channel is considered OPERABLE if either the level indicator or pen recorder is OPERABLE, provided at least one channel is continuously recorded. If no channel is recorded, both channels must be declared inoperable and the appropriate condition entered. Therefore, the PAM Specification deals specifically with these portions of the instrument channels.

The wide range water level instruments are compensated for variation in reactor water density and are calibrated to be most accurate at operational pressure and temperature, while the fuel zone level instruments are uncompensated for recirculation pump flow, and are most accurate under no flow conditions.

3. Suppression Pool Water Level

Suppression pool water level is a Category I 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. The wide range suppression pool water level measurement provides the operator with sufficient information to assess the status of both the RCPB and the water supply to the ECCS. The wide range water level indicators monitor the suppression pool water level from 1.5 feet above the bottom of the pool to 16 feet above the bottom of the pool. The primary means of monitoring this variable is from two wide range suppression pool water level signals that are transmitted from individual level transmitters. Each channel consists of an independent level indicator, and an independent pen recorder. A channel is considered OPERABLE if either the level indicator or the pen recorder is OPERABLE, provided at least one channel is continuously recorded. If no channel

(continued)

BASES

LCO

3. Suppression Pool Water Level (continued)

is recorded, both channels must be declared inoperable and the appropriate condition entered. Therefore, the PAM Specification deals specifically with these portions of the instrument channel.

4. Drywell Pressure

Drywell pressure is a Category I variable provided to detect a breach of the RCPB and to verify ECCS functions that operate to maintain RCS integrity. Two narrow range drywell pressure signals are transmitted from separate pressure transmitters. Each of these pressure transmitters feeds a signal to both a pressure indicator and a pen recorder, which indicate from -5 psig to +5 psig. Two wide range drywell pressure signals are also transmitted from separate pressure transmitters and are continuously recorded and displayed on two control room recorders. This portion of the channel (i.e., wide range) covers the pressure range for pressures greater than 5 psig (the pen recorder range is 0 psig to 250 psig). (Note: while the wide range transmitters also feed individual indicators, the indicated range is only 0 to 100 psig, which does not meet the Regulatory Guide 1.97 required range for this variable of 4 times the design pressure of 56 psig, and thus is not used to define channel OPERABILITY.) Both a narrow range instrument (i.e., either an indicator or recorder) and a wide range instrument (i.e., a pen recorder) are required to be OPERABLE in order for a channel of the Drywell Pressure Function to be considered OPERABLE. However, the full range (-5 psig to +250 psig) of drywell pressure must be continuously recorded, using any combination of transmitters and recorders. If the full range is not being continuously recorded, all channels must be declared inoperable and the appropriate Condition entered. These instruments (i.e., indicators and recorders) are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

5. Primary Containment Area Radiation

Primary containment area radiation is a Category I variable provided to monitor the potential of significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency

(continued)

BASES

LCO

5. Primary Containment Area Radiation (continued)

plans. For the DAEC, primary containment area radiation PAM instrumentation consists of the following: 1) Two independent channels of drywell area radiation monitoring; each channel consists of a radiation element, which transmits a signal to a radiation indicating monitor and to a pen on a two pen recorder; and, 2) Two independent channels of suppression chamber area radiation monitoring; each channel consists of a radiation element, which transmits a signal to a radiation indicating monitor and to a pen on a two pen recorder that receives a companion input from the drywell area channel, noted above. For both the drywell area and suppression chamber area radiation monitoring channels, a channel is considered OPERABLE if either the radiation indicating monitor or its pen on the recorder is OPERABLE, provided at least one channel is continuously recorded. If no channel is recorded, both channels must be declared inoperable and the appropriate Condition entered.

6. Primary Containment Isolation Valve (PCIV) Position

PCIV position is a Category I variable 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 one channel of valve position indication in the control room to be OPERABLE for each 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.

For the DAEC, the PCIV position PAM instrumentation consists of the individual open (i.e., red) and closed (i.e., green)

(continued)

BASES

LCO

6. Primary Containment Isolation Valve (PCIV) Position (continued)

position indicating lights for each active PCIV. These indicating lights are located on various control panels in the control room.

7. Drywell and Suppression Chamber Hydrogen and Oxygen Analyzers

Drywell and suppression chamber hydrogen and oxygen analyzers are Category I instruments provided to detect high hydrogen or oxygen concentration conditions that represent a potential for containment breach. This variable is also important in verifying the adequacy of mitigating actions. For the DAEC, the drywell and suppression chamber hydrogen and oxygen analyzers PAM instrumentation consists of the following: 1) Two independent channels of hydrogen monitoring; (to be considered OPERABLE, each channel must be capable of obtaining and analyzing a sample from both the drywell atmosphere and the suppression chamber atmosphere); and, 2) Two independent channels of oxygen monitoring; (to be considered OPERABLE, each channel must be capable of obtaining and analyzing a sample from both the drywell atmosphere and the suppression chamber atmosphere). Each of the four channels are displayed on an indicator and via a pen on a two pen recorder (i.e., there are two recorders, each with an oxygen indicating pen and a hydrogen indicating pen). A channel is considered OPERABLE if either the indicator or the pen recorder is OPERABLE, provided at least one channel is continuously recorded. If no channel is recorded, both channels must be declared inoperable and the appropriate condition entered. The normal condition for the drywell and suppression chamber hydrogen and oxygen analyzers is the Standby Mode.

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1 and 2. These variables are related to diagnosing the success of the automatic 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

Note 1 has been added to the ACTIONS to exclude the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the passive function of the instruments, the operator's ability to diagnose an accident using alternative instruments and methods, and the low probability of an event requiring these instruments.

Note 2 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 or for separate penetration flow paths for the PCIV Position Function. As such, a Note has been provided that allows separate Condition entry for each inoperable PAM Function including separate Condition entry for each penetration flow path for the PCIV Position Function.

<u>A.1</u>

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 (or, in the case of a Function that has only one required channel, other non-Regulatory Guide 1.97 instrument channels to monitor the Function), 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.

(continued)

BASES

ACTIONS (continued)

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 5.6.6, which requires a written report to be submitted to the NRC. This action is appropriate in lieu of a shutdown requirement, since alternative actions are identified before loss of functional capability, 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. The 7 day Completion Time for two inoperable hydrogen monitoring channels or two inoperable oxygen monitoring channels is based on the low probability of the occurrence of a LOCA that would generate hydrogen in amounts capable of exceeding the flammability limit, the length of time after the event that operator action would be required to prevent hydrogen accumulation from exceeding this limit, and the availability of the Nitrogen Purge System and the Containment Atmosphere Dilution System. 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.

(continued)

BASES

ACTIONS (continued)

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 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 does 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 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 5.6.6. These alternate means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allotted time.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

The following SRs apply to the PAM instrumentation Functions in Table 3.3.3.1-1, except as noted.

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. The HPCI and RCIC steam line instrumentation of the reactor steam dome pressure Function are compared to each other to satisfy this SR. For PCIV position indication, the CHANNEL CHECK consists of a comparison of the open and closed position lights with the expected position of the PCIV.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. The Frequency 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, but more frequent, checks of channels during normal operational use of those displays associated with the required channels of this LCO.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.3.1.2

A CHANNEL CALIBRATION is performed every 24 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies the channel responds to the measured parameter with the necessary range and accuracy. For PCIV position indication, the CHANNEL CALIBRATION is a comparison of a local visual check to remote control room indication to verify the PCIV's indicated position agrees with the actual position. If an indication does not agree with the actual position, adjustments are made to the PCIV's indication channel. For Primary Containment Area Radiation instrumentation (Drywell and Suppression Chamber), the CHANNEL CALIBRATION shall consist of an electronic calibration of the channel for ranges above 10 R/hr and a one point calibration check of the detector below 10 R/hr with a portable gamma source. For Drywell and Suppression Chamber H_2 and O_2 Analyzers, monitors shall be tested for OPERABILITY using standard bottled H_2 and O_2.

The Frequency is based on operating experience and consistency with the typical industry refueling cycles.

REFERENCES

1. Regulatory Guide 1.97, Revision 2, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," December 1980.

2. R. M. Pulsifer (NRC) to L. Liu (IELP), "Duane Arnold Energy Center - Conformance to Regulatory Guide 1.97, Revision 2 (TAC M84788)," dated August 4, 1993.

B 3.3 INSTRUMENTATION

B 3.3.3.2 Remote Shutdown System

BASES

BACKGROUND

The Remote Shutdown System provides the control room operator with sufficient instrumentation and controls to place and maintain the plant in a safe shutdown condition from a location other than the control room. This capability is necessary to protect against the possibility of the control room becoming inaccessible. A safe shutdown condition is defined as MODE 3. With the plant in MODE 3, the safety relief valves and the Residual Heat Removal Shutdown Cooling System can be used to remove core decay heat and meet all safety requirements. The ability to operate shutdown cooling from outside the control room allows operation in MODE 3 and the capability to place the plant in MODE 4 provides long term core decay heat removal.

In the event that the control room becomes inaccessible, the operators can establish control at the remote shutdown panel and place and maintain the plant in MODE 3. Once in MODE 3, additional controls and indication necessary to transition to MODE 4 are available. Not all controls and necessary transfer switches are located at the remote shutdown panel. Some controls and transfer switches will have to be operated locally at the switchgear, motor control panels, or other local stations. The plant is in MODE 3 following a plant shutdown and can be maintained safely in MODE 3 for a period of time.

The OPERABILITY of the Remote Shutdown System control and instrumentation Functions ensures that there is sufficient information available on selected plant parameters to place and maintain the plant in MODE 3 should the control room become inaccessible.

APPLICABLE SAFETY ANALYSES

The Remote Shutdown System is required to provide equipment at appropriate locations outside the control room with a design capability to promptly shut down the reactor to MODE 3, including the necessary instrumentation and controls, to maintain the plant in a safe condition in MODE 3.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The criteria governing the design and the specific system requirements of the Remote Shutdown System are located in the UFSAR, Section 7.4.2 (Ref. 1).

The Remote Shutdown System is considered an important contributor to reducing the risk of accidents; as such, it has been retained in the Technical Specifications (TS). The Remote Shutdown System satisfies criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The Remote Shutdown System LCO provides the requirements for the OPERABILITY of the instrumentation and controls necessary to place and maintain the plant in MODE 3 from a location other than the control room. The instrumentation and transfer/control Functions required are listed in Bases Table B 3.3.3.2-1. For the Transfer/Control circuits, the number shown in the table is the number of transfer switch circuits required.

The controls, instrumentation, and transfer switches are those required for:

- Reactor Pressure Vessel (RPV) pressure control;
- Decay heat removal;
- RPV inventory control; and
- Safety support systems for the above functions, including service water, component cooling water, and onsite power, including the diesel generators.

Pump ammeters, reactor head vents, and Reactor Feed Pump RPV High Level Trip Function, while installed in the Remote Shutdown Panels, are not required in order to consider the Remote Shutdown System OPERABLE. The Remote Shutdown System is OPERABLE if all instrument channels and transfer/control circuits needed to support the remote shutdown function are OPERABLE.

(continued)

BASES

LCO (continued)

The Remote Shutdown System instruments and transfer/control circuits covered by this LCO do not need to be energized to be considered OPERABLE. This LCO is intended to ensure that the instruments and transfer/control circuits will be OPERABLE if plant conditions require that the Remote Shutdown System be placed in operation.

APPLICABILITY

The Remote Shutdown System LCO is applicable in MODES 1 and 2. This is required so that the plant can be placed and maintained in MODE 3 for a period of time from a location other than the control room.

This LCO is not applicable in MODES 3, 4, and 5. In these MODES, the plant is already subcritical and in a condition of reduced Reactor Coolant System energy. Under these conditions, considerable time is available to restore necessary instrument control Functions if control room instruments or control becomes unavailable. Consequently, the TS do not require OPERABILITY in MODES 3, 4, and 5.

ACTIONS

A Note is included that excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into an applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require a plant shutdown. This exception is acceptable due to the low probability of an event requiring this system.

Note 2 has been provided to modify the ACTIONS related to Remote Shutdown System Functions. 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 Remote Shutdown System Functions provide appropriate compensatory measures for separate Functions.

(continued)

BASES

ACTIONS (continued)

As such, a Note has been provided that allows separate Condition entry for each inoperable Remote Shutdown System Function.

A.1

Condition A addresses the situation where one or more required Functions of the Remote Shutdown System is inoperable. This includes any instrument channel Function listed in Table B 3.3.3.2-1, as well as the transfer/control circuit Functions.

The Required Action is to restore the Function to OPERABLE status within 30 days. The Completion Time is based on operating experience and the low probability of an event that would require evacuation of the control room.

B.1

If the Required Action and associated Completion Time of Condition A are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.3.2.1

SR 3.3.3.2.1 verifies each required Remote Shutdown System transfer switch and control circuit performs the intended function. This verification is performed from the remote shutdown panel and locally, as appropriate. Operation of the equipment from the remote shutdown panel is not necessary. The Surveillance can be satisfied by performance of a continuity check. This will ensure that if the control room becomes inaccessible, the plant can be placed and maintained in MODE 3 from the remote shutdown panel and the local control stations. Operating experience demonstrates that Remote Shutdown System control channels usually pass the Surveillance when performed at the 24 month Frequency.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.3.2.2

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor for each of the instrument channel functions. The test verifies the channel responds to measured parameter values with the necessary range and accuracy.

The 24 month Frequency is based upon operating experience and consistency with the typical industry refueling cycle.

REFERENCES

1. UFSAR, Section 7.4.2.

(continued)

TABLE B 3.3.3.2-1 (PAGE 1 OF 2)

REMOTE SHUTDOWN SYSTEM INSTRUMENTATION

FUNCTION	REQUIRED NUMBER OF CHANNELS
Instrument Parameter	
1. Reactor Pressure Vessel Pressure Control	
a) Reactor Pressure	1
2. Reactor Pressure Vessel Inventory Control	
a) Reactor Level (Yarway)	1
b) Reactor Level (Floodup)	1
c) B Core Spray Discharge Flow	1
3. Decay Heat Removal	
a) B RHR Loop Flow	1
b) B RHRSW Loop Flow	1
c) B RHR Heat Exchanger Shell Pressure	1
4. Safety Support Systems	
a) Torus Level	1
b) Torus Temperature	1
Transfer/Control Functions	
1. Reactor Pressure Vessel Pressure Control	
a) SRVs (Manual and Self-actuating)	2
b) SRVs (Self-actuating only)	(a)
c) SRVs (Self-actuating only)	1
d) Reactor Pressure Instrumentation	(b)
e) Emergency Fuses	(c)
2. Reactor Pressure Vessel Inventory Control	
a) B Core Spray MOVs	3
b) B Core Spray Pump	1
c) RCIC RPV High Level Shutdown/Restart	(d)
d) Outboard MSIVs	4
e) Closes Outboard MSL Drain MO-4424	1
f) HPCI RPV High Level Shutdown/Restart	1
g) Reactor Level Floodup Instrumentation	(b)
h) Reactor Level (Yarway) and B Core Spray Discharge Flow Instrumentation	(e)
i) Emergency Fuses	(c)
3. Decay Heat Removal	
a) B RHR MOVs	7
b) B RHR Pumps	1
c) B RHRSW MOV and Closes RHR Radwaste Isolation MO-1937	1
d) B RHRSW Pumps	1
e) Closes RHR Cross-tie MO-2010	(d)

(continued)

TABLE B 3.3.3.2-1 (PAGE 2 OF 2)

REMOTE SHUTDOWN SYSTEM INSTRUMENTATION

FUNCTION	REQUIRED NUMBER OF CHANNELS
3. Decay Heat Removal (continued)	
f) RHR SDC 135 psig Permissive	1
g) A RHR Pump Suction MOVs and Emergency Fuses	1
h) B RHR Loop Flow and B RHRSW Loop Flow Instrumentation	1
i) B RHR Heat Exchanger Shell Pressure Instrumentation	(e)
j) Emergency Fuses	(c)
4. Safety Support Systems	
a) B ESW Pump	1
b) B RWS Pumps	1
c) Aligns B RWS Makeup Valves	(a)
d) B DG Controls and Emergency Fuses	2
e) B Essential 4160 Instrumentation	1
f) B Essential 4160 and 480 Breakers	6
g) Starts B RHR/CS and DG Room Cooling Fans	1
h) Starts B Battery Exhaust Fan Enables B DG Fuel Oil Transfer Pump	1
i) Opens Containment N_2 Supply CV-4371A	1
j) Torus Water Level and Torus Water Temperature Instrumentation	(b)
k) Emergency Fuses	(c)

(a) One switch is required and transfers both functions annotated with this note.

(b) One switch is required and transfers all three functions annotated with this note.

(c) One switch is required and transfers fuses for all four functions annotated with this note.

(d) One switch transfers both functions annotated with this note.

(e) One switch transfers both functions annotated with this note.

B 3.3 INSTRUMENTATION

B 3.3.4.1 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

BASES

BACKGROUND

The EOC-RPT instrumentation initiates a Recirculation Pump Trip (RPT) to reduce the peak reactor pressure and power resulting from turbine trip or generator load rejection transients to provide additional margin to core thermal MCPR Safety Limits (SLs).

The need for the additional negative reactivity in excess of that normally inserted on a scram reflects end of cycle reactivity considerations. The scram reactivity depends on the ability of the control rods to be in the high flux regions of the core. The minimum scram reactivity occurs at the end of cycle when control rods are fully withdrawn from the core. In this situation, it takes a longer time for the control rods to travel to a high importance region in the core. For this reason at the end of cycle the control rods may not be able to ensure that thermal limits are maintained by inserting sufficient negative reactivity during the first few feet of rod travel upon a scram caused by Turbine Control Valve (TCV) Fast Closure, Trip Oil Pressure – Low or Turbine Stop Valve (TSV) – Closure. The physical phenomenon involved is that the positive reactivity feedback due to a pressurization transient (i.e., void collapse) can add positive reactivity at a faster rate than the control rods can add negative reactivity.

The EOC-RPT instrumentation, is composed of sensors that detect initiation of closure of the TSVs or fast closure of the TCVs, combined with relays, logic circuits, and fast acting circuit breakers that interrupt power from the recirculation pump Motor Generator (MG) set generators to each of the recirculation pump motors. The channels include electronic equipment (e.g., limit switches or pressure switches) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs an EOC-RPT signal to the trip logic. When the RPT breakers trip open, the recirculation pumps coast down under their own inertia. The EOC-RPT has two identical trip systems, either of which can actuate an RPT in both recirculation loops.

(continued)

BASES

BACKGROUND (continued)

Each EOC-RPT trip system is a two-out-of-two logic for each Function; thus, either two TSV – Closure or two TCV Fast Closure, Trip Oil Pressure – Low signals are required for a trip system to actuate. If either trip system actuates, both recirculation pumps will trip. There are two EOC-RPT breakers in series per recirculation pump. One trip system trips one of the two EOC-RPT breakers for each recirculation pump, and the second trip system trips the other EOC-RPT breaker for each recirculation pump.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The TSV – Closure and the TCV Fast Closure, Trip Oil Pressure – Low Functions are designed to trip the recirculation pumps in the event of a turbine trip or generator load rejection to mitigate the neutron flux, heat flux, and pressurization transients, and to increase the margin to the MCPR SL. The analytical methods and assumptions used in evaluating the turbine trip and generator load rejection, as well as other safety analyses that ensure EOC-RPT, are summarized in References 1, 2, and 3.

To mitigate pressurization transient effects, the EOC-RPT must trip the recirculation pumps after initiation of closure movement of either the TSVs or the TCVs. The combined effects of this trip and a scram reduce fuel bundle power more rapidly than a scram alone, resulting in an increased margin to the MCPR SL. Alternatively, MCPR limits for an inoperable EOC-RPT, as specified in the COLR, are sufficient to mitigate pressurization transient effects. The EOC-RPT function is automatically disabled when turbine first stage pressure is < 30% RTP.

EOC-RPT instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

The OPERABILITY of the EOC-RPT is dependent on the OPERABILITY of the individual instrumentation channel Functions. Each Function must have a required number of OPERABLE channels in each trip system, with their setpoints within the specified Allowable Value of SR 3.3.4.1.2. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Channel OPERABILITY also includes the associated EOC-RPT breakers. Each channel

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

(including the associated EOC-RPT breakers) must also respond within its assumed response time.

Allowable Values are specified for each EOC-RPT Function specified in the LCO. Nominal trip setpoints are specified in the setpoint calculations. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The nominal setpoints are selected to ensure that the 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. Each Allowable Value specified is more conservative than the Analytical Limit assumed in the transient and accident analysis in order to account for instrument uncertainties appropriate to the Function. 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., TSV position), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., limit switch) changes state. Analytical Limits, where established, are the limiting values of the process parameters used in safety analysis to define the margin to unacceptable consequences. Margin is provided between the Allowable Value and the Analytical Limits to allow for process, calibration (i.e., M&TE) and some instrument uncertainties. Additional margin is provided between the Allowable Value and the trip setpoint to allow for the remaining instrument uncertainties (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 specific Applicable Safety Analysis, LCO, and Applicability discussions are listed below on a Function by Function basis.

Alternatively, since this instrumentation protects against a MCPR SL violation, with the instrumentation inoperable, modifications to the MCPR limits (LCO 3.2.2) may be applied to allow this LCO to be met. The MCPR penalty for the EOC-RPT inoperable condition is specified in the COLR.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Turbine Stop Valve – Closure

Closure of the TSVs and a main turbine trip result in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, an RPT is initiated on TSV – Closure in anticipation of the transients that would result from closure of these valves. EOC-RPT decreases reactor power and aids the reactor scram in ensuring that the MCPR SL is not exceeded during the worst case transient.

Closure of the TSVs is determined by measuring the position of each valve. There is one position switch associated with each stop valve. The signals from two of the four valves are assigned to one trip system, while the signals from the other two valves are assigned to the other trip system. The logic for the TSV – Closure Function is such that either TSV 1 and TSV 2 must be less than fully open or TSV 3 and TSV 4 must be less than fully open to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ 30% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure. Because an increase in the main turbine bypass flow can affect this Function nonconservatively (THERMAL POWER is derived from the turbine first stage pressure), the main turbine bypass valves must not cause the trip Functions to be bypassed in order to maintain this Function OPERABLE when THERMAL POWER ≥ 30% RTP. In addition, other steam loads, such as second stage reheaters in operation below 30% RTP, must be accounted for in establishing this setpoint. Otherwise, turbine first stage pressure would be non-conservative with respect to the 30% RTP RPS bypass. Four channels of TSV – Closure, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TSV – Closure Allowable Value is selected to detect imminent TSV closure.

This protection is required, consistent with the safety analysis assumptions, whenever THERMAL POWER is ≥ 30% RTP. Below 30% RTP, the Reactor Vessel Steam Dome Pressure – High and the Average Power Range Monitor (APRM) Fixed Neutron Flux – High Functions of the Reactor Protection System (RPS) are adequate to maintain the necessary safety margins.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Turbine Control Valve Fast Closure, Trip Oil Pressure - Low

Fast closure of the TCVs during a generator load rejection results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, an RPT is initiated on TCV Fast Closure, Trip Oil Pressure - Low in anticipation of the transients that would result from the closure of these valves. The EOC-RPT decreases reactor power and aids the reactor scram in ensuring that the MCPR SL is not exceeded during the worst case transient.

Fast closure of the TCVs is determined by measuring the electrohydraulic control fluid pressure at each control valve. There is one pressure switch associated with each control valve. The signal from two of the four valves are assigned one trip system, while the signals from the other two valves are assigned to the other trip system. The logic for the TCV Fast Closure, Trip Oil Pressure - Low Function is such that either TCV 1 and TCV 2 must be closed (pressure switch trips) or TCV 3 and TCV 4 must be closed to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ 30% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure. Because an increase in the main turbine bypass flow can affect this Function nonconservatively (THERMAL POWER is derived from turbine first stage pressure), the main turbine bypass valves must not cause the trip Functions to be bypassed in order to maintain this Function OPERABLE when THERMAL POWER is ≥ 30% RTP. In addition, other steam loads, such as second stage reheaters in operation below 30% RTP, must be accounted for in establishing this setpoint. Otherwise, turbine first stage pressure would be non-conservative with respect to the 30% RTP RPS bypass. Four channels of TCV Fast Closure, Trip Oil Pressure - Low, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TCV Fast Closure, Trip Oil Pressure - Low Allowable Value is selected high enough to detect imminent TCV fast closure. The transient analysis assumes a response time of 30 msec (i.e., start of TCV closure from full open to switch actuation). The switch setting is selected to support this response time.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Turbine Control Valve Fast Closure, Trip Oil Pressure-Low (continued)

This protection is required consistent with the safety analysis whenever THERMAL POWER is ≥ 30% RTP. Below 30% RTP, the Reactor Vessel Steam Dome Pressure-High and the APRM Fixed Neutron Flux-High Functions of the RPS are adequate to maintain the necessary safety margins.

ACTIONS

A Note has been provided to modify the ACTIONS related to EOC-RPT 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 EOC-RPT 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 EOC-RPT instrumentation channel.

A.1 and A.2

With one or more channels inoperable, but with EOC-RPT trip capability maintained (refer to Required Actions B.1 and B.2 Bases), the EOC-RPT System is capable of performing the intended function. However, the reliability and redundancy of the EOC-RPT instrumentation is reduced such that a single failure in the remaining trip system could result in the inability of the EOC-RPT System to perform the intended function. Therefore, only a limited time is allowed to restore compliance with the LCO. Because of the diversity of sensors available to provide 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 an EOC-RPT, 72 hours is provided to restore the inoperable channels (Required Action A.1). Additionally, applying the EOC-RPT inoperable MCPR limit satisfies the LCO and is thus acceptable.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Alternately, the inoperable channels may be placed in trip (Required Action A.2) since this would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. As noted, placing the channel in trip with no further restrictions is not allowed if the inoperable channel is the result of an inoperable breaker, since this may not adequately compensate for the inoperable breaker (e.g., the breaker may be inoperable such that it will not open). If it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an RPT, or if the inoperable channel is the result of an inoperable breaker), Condition C must be entered and its Required Actions taken.

B.1 and B.2

Required Actions B.1 and B.2 are intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in the Function not maintaining EOC-RPT trip capability. A Function is considered to be maintaining EOC-RPT trip capability when sufficient channels are OPERABLE or in trip, such that the EOC-RPT System will generate a trip signal from the given Function on a valid signal and both recirculation pumps can be tripped. This requires two channels of the Function in the same trip system, to each be OPERABLE or in trip, and the associated EOC-RPT breakers to be OPERABLE or in trip. Alternately, Required Action B.2 requires the MCPR limit for inoperable EOC-RPT, as specified in the COLR, to be applied. This also restores the margin to MCPR assumed in the safety analysis.

The 2 hour Completion Time is sufficient time for the operator to restore EOC-RPT trip capability or apply the MCPR penalty, and takes into account the likelihood of an event requiring actuation of the EOC-RPT instrumentation during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR violation. Once the MCPR penalty is applied, Required Action A.1 of LCO 3.2.2 allows 2 hours for MCPR to be restored within limits, if necessary.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

With any Required Action and associated Completion Time not met, THERMAL POWER must be reduced to < 30% RTP within 4 hours. Alternately, the associated recirculation pump may be removed from service, since this performs the intended function of the instrumentation. The allowed Completion Time of 4 hours is reasonable, based on operating experience, to reduce THERMAL POWER to < 30% RTP from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

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 EOC-RPT 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 (Ref. 4) 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 recirculation pumps will trip when necessary.

SR 3.3.4.1.1

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. The CHANNEL FUNCTIONAL TEST verifies acceptable response by verifying the change of state of at least one contact on the relay which inputs into the trip logic. The required contacts not tested during the CHANNEL FUNCTIONAL TEST are tested under the LOGIC SYSTEM FUNCTIONAL TEST. This is acceptable because operating experience shows that the contacts not tested during the CHANNEL FUNCTIONAL TEST normally pass the LOGIC SYSTEM FUNCTIONAL TEST. The RPT breaker is excluded from this testing.

The Frequency of 92 days is based on reliability analysis of Reference 4.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.4.1.2

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies 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.

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 3.3.4.1.3

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The functional test of the pump breakers is included as a part of this test, overlapping the LOGIC System Functional Test, to provide complete testing of the associated safety function. Therefore, if a breaker is incapable of operating, the associated instrument channel(s) would also be inoperable.

The 24 month Frequency 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 these components usually pass the Surveillance when performed at the 24 month Frequency.

SR 3.3.4.1.4

This SR ensures that an EOC-RPT initiated from the TSV–Closure and TCV Fast Closure, Trip Oil Pressure–Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ 30% RTP. This involves calibration of the bypass channels. The 30% RTP is the analytical limit. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint. Because main turbine bypass flow can affect this setpoint nonconservatively (THERMAL POWER is derived from first stage pressure) the main turbine bypass valves must remain closed at THERMAL

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1.4 (continued)

POWER ≥ 30% RTP to ensure that the calibration remains valid. If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ 30% RTP, either due to open main turbine bypass valves or other reasons), the affected TSV—Closure and TCV 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 with the channel considered OPERABLE.

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

SR 3.3.4.1.5

This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the transient analysis. The EOC-RPT SYSTEM RESPONSE TIME acceptance criteria are documented in Reference 3.

EOC-RPT SYSTEM RESPONSE TIME tests are conducted on a 24 month STAGGERED TEST BASIS. Response times cannot be determined at power because operation of final actuated devices is required. Therefore, 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 that cause serious response time degradation, but not channel failure, are infrequent occurrences.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 5.2.2.
2. UFSAR, Sections 15.2.1 and 15.2.2.
3. UFSAR, Section 7.2.1.2.3.
4. GENE-770-06-1, "Bases For Changes To Surveillance Test Intervals And Allowed Out-Of-Service Times For Selected Instrumentation Technical Specifications," February 1991.

B 3.3 INSTRUMENTATION

B 3.3.4.2 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation

BASES

BACKGROUND

The ATWS-RPT System initiates an RPT, adding negative reactivity, following events in which a scram does not (but should) occur, to lessen the effects of an ATWS event. Tripping the recirculation pumps adds negative reactivity from the increase in steam voiding in the core area as core flow decreases. When Reactor Vessel Water Level – Low Low or Reactor Steam Dome Pressure – High setpoint is reached, the recirculation pump motor breakers trip.

The ATWS-RPT System (Ref. 1) includes sensors, relays, bypass capability circuit breakers, and switches that are necessary to cause initiation of an RPT. The channels include electronic equipment (e.g., on-off sensors) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs an ATWS-RPT signal to the trip logic.

The ATWS-RPT consists of two independent trip systems, with two channels of Reactor Steam Dome Pressure – High and two channels of Reactor Vessel Water Level – Low Low in each trip system. Each ATWS-RPT trip system is a two-out-of-two logic for each Function. Thus, either two Reactor Water Level – Low Low or two Reactor Pressure – High signals are needed to trip a trip system. The outputs of the channels in a trip system are combined in a logic so that either trip system will trip both recirculation pumps (by tripping the associated pump motor breakers). Since redundancy in the ATWS-RPT is not required, only one of the two trip systems is required to be OPERABLE.

Each pump motor has two breakers in series between the pump motor and the recirculation motor generator set. One pump motor breaker in each recirculation loop is associated with one trip system, while the other pump motor breaker in each recirculation loop is associated with the other trip system, for a total of four breakers. The output of each trip system is provided to both associated recirculation pump breakers.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The ATWS-RPT is not assumed to function during or after a design basis accident. The ATWS-RPT initiates an RPT to aid in preserving the integrity of the fuel cladding following events in which a scram does not, but should, occur. Based on its contribution to the reduction of overall plant risk, however, the instrumentation is included as required by Criterion 4 of 10 CFR 50.36(c)(2)(ii).

The OPERABILITY of the ATWS-RPT is dependent on the OPERABILITY of the individual instrumentation channel Functions. Each Function must have two OPERABLE channels in a trip system, with their setpoints within the specified Allowable Value of SR 3.3.4.2.3. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Channel OPERABILITY also includes the associated recirculation pump motor breakers. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Allowable Values are specified for each ATWS-RPT Function specified in the LCO. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between 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 vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., on-off sensors) changes state. Analytical Limits are the limiting values of the process parameters for use in safety analysis. Margin is provided between the Allowable Value and the Analytical Limits to allow for process, calibration (i.e., M&TE) and some instrument uncertainties. Additional margin is provided between the Allowable Value and the trip setpoint to allow for the remaining instrument uncertainties (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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The individual Functions are required to be OPERABLE in MODE 1 to protect against common mode failures of the Reactor Protection System by providing a diverse trip to mitigate the consequences of a postulated ATWS event. The Reactor Steam Dome Pressure – High and Reactor Vessel Water Level – Low Low Functions are required to be OPERABLE in MODE 1, since the reactor is producing significant power and the recirculation system could be at high flow. During this MODE, the potential exists for pressure increases or low water level, assuming an ATWS event. In MODE 2, the reactor is at low power and the recirculation system is at low flow; thus, the potential is low for a pressure increase or low water level, assuming an ATWS event. Therefore, the ATWS-RPT is not necessary. In MODES 3 and 4, the reactor is shut down with all control rods inserted; thus, an ATWS event is not significant and the possibility of a significant pressure increase or low water level is negligible. In MODE 5, the one rod out interlock ensures that the reactor remains subcritical; thus, an ATWS event is not significant. In addition, the Reactor Pressure Vessel (RPV) head is not fully tensioned and no pressure transient threat to the Reactor Coolant Pressure Boundary (RCPB) exists.

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

a. Reactor Vessel Water Level – Low Low

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, the ATWS-RPT System is initiated at Reactor Vessel Water Level-Low Low to aid in maintaining level above the top of the active fuel. The reduction of core flow reduces the neutron flux and THERMAL POWER and, therefore, the rate of coolant boiloff.

Reactor vessel water level signals are initiated from four level switches 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Reactor Vessel Water Level - Low Low (continued)

Since redundancy is not required in the ATWS-RPT, two channels of Reactor Vessel Water Level - Low Low in a trip system are available and required to be OPERABLE. The Reactor Vessel Water Level - Low Low Allowable Value is chosen for convenience with the reactor core isolation cooling initiation. A 9 second time delay is incorporated into the RPT initiating signal of Reactor Vessel Water Level - Low Low. This delay is to allow the Low Pressure Coolant Injection System Loop Selection Logic to complete its function.

b. Reactor Steam Dome Pressure - High

Excessively high RPV pressure may rupture the RCPB. An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This increases neutron flux and THERMAL POWER, which could potentially result in fuel failure and overpressurization. The Reactor Steam Dome Pressure - High Function initiates an RPT for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power generation. For the overpressurization event, the RPT aids in the termination of the ATWS event and, along with the safety/relief valves, limits the peak RPV pressure to less than the ASME Section III Code Service Level C limits (1500 psig).

The Reactor Steam Dome Pressure - High signals are initiated from four pressure switches that monitor reactor steam dome pressure. However, since redundancy is not required in the ATWS-RPT, two channels of Reactor Steam Dome Pressure - High in a trip system are required to be OPERABLE. The Reactor Steam Dome Pressure - High Allowable Value is chosen to provide an adequate margin to the ASME Section III Code Service Level C allowable Reactor Coolant System pressure.

(continued)

BASES (continued)

ACTIONS

A Note has been provided to modify the ACTIONS related to ATWS-RPT 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 ATWS-RPT 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 ATWS-RPT instrumentation channel.

A.1

Required Action A.1 is intended to ensure that appropriate actions are taken if either a single channel is inoperable and untripped or if multiple, untripped channels within the same Function are inoperable such that the Function does not maintain ATWS-RPT trip capability. A Function is considered to be maintaining ATWS-RPT trip capability when both channels in a trip system are OPERABLE or in trip, and both recirculation pumps can be tripped. This requires two channels of the Function in the same trip system to each be OPERABLE or in trip, and the recirculation pump motor breakers to be OPERABLE or in trip. However, placing the channel in trip with no further restrictions is not allowed if the inoperable channel is the result of an inoperable breaker, since this may not adequately compensate for the inoperable breaker (e.g., the breaker may be inoperable such that it will not open). If it is not desired to place the channel in trip (e.g. as in the case where placing the inoperable channel in trip would result in an RPT), or if the inoperable channel is the result of an inoperable breaker, Condition C must be entered and its Required Actions taken.

The 72 hour Completion Time is sufficient for the operator to take corrective action (i.e., place the channel in trip) and takes into account the likelihood of an event requiring actuation of the ATWS-RPT instrumentation during this period and that one instrumentation Function is still maintaining ATWS-RPT trip capability.

(continued)

BASES

ACTIONS (continued)

B.1

With one or more channels for both Functions in a trip system inoperable, the ATWS-RPT System is not capable of performing the intended function. Required Action B.1 is intended to ensure that appropriate Actions are taken if multiple, inoperable, untripped channels within both Functions result in both Functions not maintaining ATWS-RPT trip capability. Therefore, a 1 hour Completion Time is provided to restore the inoperable channel. Alternately, the inoperable channel may be placed in trip since this would conservatively compensate for the inoperability and allow operation to continue. As noted, placing the channel in trip with no further restrictions is not allowed if the inoperable channel is the result of an inoperable breaker, since this may not adequately compensate for the inoperable breaker (e.g., the breaker may be inoperable such that it will not open). If it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an RPT), or if the inoperable channel is the result of an inoperable breaker, Condition C must be entered and its Required Actions taken.

The 1 hour Completion Time is sufficient for the operator to take corrective action and takes into account the likelihood of an event requiring actuation of the ATWS-RPT instrumentation during this period.

C.1 and C.2

With any Required Action and associated Completion Time not met, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 8 hours (Required Action C.2). Alternately, the associated recirculation pump may be removed from service since this performs the intended function of the instrumentation (Required Action C.1). The allowed Completion Times of 8 hours to reach MODE 2 from full power conditions and of 8 hours to remove a recirculation pump from service in an orderly manner and without challenging plant systems, are reasonable, based on operating experience.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.4.2.1

Performance of the CHANNEL CHECK for the Reactor Vessel Water Level-Low Low Function once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the 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 are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

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

SR 3.3.4.2.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. The CHANNEL FUNCTIONAL TEST verifies acceptable response by verifying the change of state of at least one contact on the relay which inputs into the trip logic. The required contacts not tested during the CHANNEL FUNCTIONAL TEST are tested under the LOGIC SYSTEM FUNCTIONAL TEST. This is acceptable because operating experience shows that the contacts not tested during the CHANNEL FUNCTIONAL TEST normally pass the LOGIC SYSTEM FUNCTIONAL TEST. The RPT breaker itself is excluded from this testing.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.4.2.2 (continued)

The Frequency of 12 months is based on the fact that ATWS is considered a very low probability event and is outside the normal design basis. Therefore, the surveillance frequency is less stringent than for safety-related instrumentation.

SR 3.3.4.2.3

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies 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.

The Frequency is based upon the assumption of a 12 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.4.2.4

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The system functional test of the pump breakers is included as part of this Surveillance and overlaps the LOGIC SYSTEM FUNCTIONAL TEST to provide complete testing of the assumed safety function. Therefore, if a breaker is incapable of operating, the associated instrument channel(s) would be inoperable.

The 24 month Frequency 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 these components usually pass the Surveillance when performed at the 24 month Frequency.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Figure 7.2-10, ATWS-RPT Logic Diagram.

2. GENE-770-06-1, "Bases for Changes To Surveillance Test Intervals and Allowed Out-of-Service Times For Selected Instrumentation Technical Specifications," February 1991.

B 3.3 INSTRUMENTATION

B 3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation

BASES

BACKGROUND

The purpose of the ECCS instrumentation is to initiate appropriate responses from the systems to ensure that the fuel is adequately cooled in the event of a design basis accident or transient.

For most Abnormal Operational Transients and Design Basis Accidents (DBAs), a wide range of dependent and independent parameters are monitored.

The ECCS instrumentation actuates Core Spray (CS), Low Pressure Coolant Injection (LPCI), High Pressure Coolant Injection (HPCI), Automatic Depressurization System (ADS), and the Diesel Generators (DGs). The equipment involved with each of these systems is described in the Bases for LCO 3.5.1, "ECCS - Operating" or LCO 3.8.1, "AC Sources-Operating."

<u>Core Spray System</u>

The CS System may be initiated by either automatic or manual means, although manual initiation requires manipulation of individual pump and valve control switches. Automatic initiation occurs for conditions of Reactor Vessel Water Level - Low Low Low or Drywell Pressure - High. Each of these diverse variables is monitored by four redundant instruments, which are, in turn, connected to four relays whose contacts are arranged in a one-out-of-two taken twice logic (i.e., two trip systems) for each Function.

Both initiation signals are sealed in signals and must be manually reset. The CS initiation logic can be reset when both the drywell pressure and the reactor water level have been restored. The CS pumps can be secured even with an initiation signal present. A "disagreement" light will illuminate if this situation occurs. Upon receipt of an initiation signal, the CS pumps are started after 5 seconds (nominal) if power is available. If power is unavailable, the CS pump will start 5 seconds (nominal) after power is restored to the associated essential bus.

(continued)

BASES

BACKGROUND

Core Spray System (continued)

The CS test bypass valve, which is also a Primary Containment Isolation Valve (PCIV), is normally closed to maintain primary containment integrity, and also closes, if open, on a CS initiation signal to allow the assumed system flow to the Reactor Pressure Vessel (RPV). This design feature is not assumed in any accident analyses. The accident analyses are performed assuming the ECCS systems are in the standby readiness mode except that accidents are not assumed to initiate during unit testing.

The CS pump discharge flow is monitored by a flow transmitter. The minimum flow return line valve is normally open and is automatically closed if flow is above the minimum flow setpoint to allow the system flow assumed in the accident analysis.

The CS initiation logic also monitors the pressure in the RPV to ensure that, before the injection valves open, the RPV pressure has fallen to a value below the CS System's maximum design pressure. The variable is monitored by four redundant pressure switches, which are, in turn, connected to four relays whose contacts are arranged in a one-out-of-two taken twice logic.

Low Pressure Coolant Injection System

LPCI is an operating mode of the Residual Heat Removal (RHR) System. The LPCI System may be initiated by automatic or manual means, although manual initiation requires manipulation of individual pump and valve control switches. Automatic initiation occurs for conditions of Reactor Vessel Water Level – Low Low Low or Drywell Pressure – High. Each of these diverse variables is monitored by four redundant instruments, which, in turn, are connected to four relays whose contacts are arranged in a one-out-of-two taken twice logic (i.e., two trip systems) for each Function. Once an initiation signal is received by the LPCI control circuitry, the signal is sealed in until both initiation signals clear and the logic is manually reset. The LPCI pumps can be secured even with an initiation signal present. A "disagreement" light will illuminate if this situation occurs.

(continued)

BASES

BACKGROUND

Low Pressure Coolant Injection System (continued)

Upon receipt of an initiation signal, the A and B LPCI pumps start after a 10 second (nominal) delay if power is available.The C and D LPCI pumps are started after a 15 second (nominal) delay if power is available. The respective time delays are intended to limit the loading of the standby power sources. If power is unavailable, the respective LPCI pumps will start after power is restored to the respective emergency bus, and after the applicable time delay.

The LPCI System initiation logic also contains LPCI Loop Select Logic whose purpose is to identify and direct LPCI flow to the unbroken recirculation loop if a Design Basis Accident (DBA) occurs. The LPCI Loop Select Logic is initiated upon the receipt of either a Reactor Vessel Water Level - Low Low signal or a Drywell Pressure - High signal. Each of these variables is monitored by four redundant instruments which are, in turn, connected to four relays whose contacts are arranged in a one-out-of-two taken twice logic. When initiated, the LPCI Loop Select Logic first determines recirculation pump operation by sensing the differential pressure (dp) between the suction and discharge of each pump. There are four dp switches in each recirculation loop which are, in turn, connected to relays whose contacts are arranged in a one-out-of-two taken twice logic for each recirculation pump. If the logic senses that either pump is not running, i.e., single loop operation, then a trip signal is sent to both recirculation pumps to eliminate the possibility of pipe breaks being masked by the operating recirculation pump pressure. If a pump trip signal is generated, RPV pressure must drop to a specified value before the logic will continue. This adjusts the selection time to optimize sensitivity and still ensure that LPCI injection is not unnecessarily delayed. The RPV pressure is sensed by four pressure switches that are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic. After the satisfaction of this pressure requirement or if both recirculation pumps indicate they are running, a time delay is provided to remove initial pressure perturbations and allow momentum effects of the pump coastdown to settle out. Selection of the unbroken recirculation loop is then initiated. This is done by comparing the absolute pressure of the two

(continued)

BACKGROUND

Low Pressure Coolant Injection System (continued)

recirculation loops. The broken loop is indicated by a lower pressure than the unbroken loop. The loop with the higher pressure is then used for LPCI injection. If, after a small time delay, the pressure in loop A is not indicating higher than loop B, the logic will provide a signal to close the B recirculation loop discharge and discharge bypass valves, open the LPCI injection valve to the B recirculation loop and close the LPCI injection valve to the A recirculation loop. This is the "default" choice in the LPCI Loop Select Logic. (However, the discharge bypass valves are not environmentally - qualified and the DAEC accident analysis (Ref. 4) assumes that they remain open and divert some flow out of the pipe break.) If recirculation loop A pressure indicates higher than loop B pressure, the recirculation valves in loop A are closed, the LPCI injection valve to loop A is signaled to open and the LPCI injection valve to loop B is signaled to close. The four dp switches which provide input to this portion of the logic detect the pressure difference between the corresponding risers to the jet pumps in each recirculation loop. The four dp switches are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic. There are two redundant trip systems in the LPCI Loop Select Logic.

The LPCI System's two discharge flow paths are monitored by flow transmitters which provide input to the two minimum flow return line valves. The two Division 1 pumps are associated with one minimum flow return line, and the two Division 2 pumps are associated with the other minimum flow return line. When a pump is running and discharge flow is low enough so that pump overheating may occur, the respective minimum flow return line valve is opened. If flow is above the minimum flow setpoint, the valve is automatically closed to allow the system flow assumed in the analyses.

The RHR test line suppression pool cooling isolation valve, suppression pool spray isolation valves, and containment spray isolation valves (which are also PCIVs) are normally closed to maintain primary containment integrity, and also close, if open, on a LPCI initiation signal to allow the assumed system flow to the RPV. These design features are not assumed in any accident analyses. The accident analyses

(continued)

BASES

BACKGROUND

Low Pressure Coolant Injection System (continued)

are performed assuming the ECCS systems are in the standby readiness condition when the accident occurs.

The LPCI initiation logic monitors the pressure in the RPV to ensure that, before an injection valve opens, the RPV pressure has fallen to a value below the LPCI System's maximum design pressure. The variable is monitored by four redundant pressure switches, which are, in turn, connected to four relays whose contacts are arranged in a one-out-of-two taken twice logic.

Low reactor water level in the shroud is detected by four additional instruments to allow the use of other modes of RHR (e.g., suppression pool cooling) when a LPCI initiation signal is present. These instruments allow containment cooling to be performed if the level in the shroud is above the prescribed setpoint.

High Pressure Coolant Injection System

The HPCI System may be initiated by either automatic or manual means although manual initiation requires manipulation of individual component control switches. Automatic initiation occurs for conditions of Reactor Vessel Water Level – Low Low or Drywell Pressure – High. Each of these variables is monitored by four redundant instruments, which are, in turn, connected to four relays whose contacts are arranged in a one-out-of-two taken twice logic for each Function.

The HPCI pump discharge flow is monitored by a flow transmitter. When the pump is running and discharge flow is low enough so that pump overheating may occur, the minimum flow return line valve is opened. The valve is automatically closed if flow is above the minimum flow setpoint to allow the system flow assumed in the accident analysis.

The HPCI test return valve and redundant shutoff valve are closed upon receipt of a HPCI initiation signal to allow the assumed system flow to the RPV. This design feature is not assumed in any accident analyses. The accident analyses are performed assuming the ECCS are in the standby readiness

(continued)

BASES

BACKGROUND

High Pressure Coolant Injection System (continued)

condition when the accident occurs. These valves are also closed if either suppression pool suction valve is full open. This prevents pumping suppression pool water to the Condensate Storage Tank (CST).

The HPCI System also monitors the water levels in the Condensate Storage Tank (CST) and the suppression pool because these are the two sources of water for HPCI operation. Reactor grade water in the CST is the preferred source although this source would not necessarily be available under accident conditions. Upon receipt of a HPCI initiation signal, the CST suction valve is automatically signaled to open (it is normally in the open position) unless both suppression pool suction valves are open. If the water level in the CST falls below a preselected level, first the suppression pool suction valves automatically open, and then the CST suction valve automatically closes. Two level switches are used to detect low water level in the CST. Either switch can cause the suppression pool suction valves to open and the CST suction valve to close. The suppression pool suction valves also automatically open and the CST suction valve closes if high water level is detected in the suppression pool. To prevent losing suction to the pump, the suction valves are interlocked so that one suction path must be open before the other automatically closes.

The HPCI System provides makeup water to the reactor until the reactor vessel water level reaches the Reactor Vessel Water Level – High trip, at which time the HPCI turbine trips, which causes the turbine's stop valves, injection valve and minimum flow valve to close. The logic is two-out-of-two once to provide high reliability of the HPCI System. The HPCI System automatically restarts if a Reactor Vessel Water Level – Low Low signal is subsequently received.

Automatic Depressurization System

The ADS may be initiated either automatically or by manually opening the individual valves using the corresponding handswitches. Automatic initiation occurs when signals indicating Reactor Vessel Water Level – Low Low Low; confirmed Reactor Vessel Water Level – Low, and CS or LPCI Pump Discharge Pressure – High are all present and the ADS

(continued)

BASES

BACKGROUND

Automatic Depressurization System (continued)

Timer has timed out. There are two level switches for Reactor Vessel Water Level – Low Low and one level switch for confirmed Reactor Vessel Water Level – Low, in each of the two ADS trip logics. Each of these level switches connects to a relay whose contacts form the initiation logic.

Each ADS trip logic includes a time delay between satisfying the initiation logic and the actuation of the ADS valves. The ADS Timer time delay setpoint chosen is long enough that the HPCI has sufficient operating time to recover to a level above Reactor Vessel Water Level-Low Low, yet not so long that the LPCI and CS Systems are unable to adequately cool the fuel if the HPCI fails to maintain that level. An alarm in the control room is annunciated when either of the timers is timing. Resetting the ADS initiation signals resets the ADS Timers. In addition, these timers can be manually overridden to preclude an unnecessary RPV depressurization.

The ADS also monitors the discharge pressures of the four LPCI pumps and the two CS pumps. Each ADS trip logic includes two discharge pressure permissive switches from each of the following: one CS pump and two LPCI pumps in the associated Division (i.e., Division 1 LPCI pumps A and C input to ADS trip logic A, and Division 2 LPCI pumps B and D input to ADS trip logic B). The signals are used as a permissive for ADS actuation, indicating that there is a source of core coolant available once the ADS has depressurized the vessel. Any one of the six low pressure pumps is sufficient to provide adequate core cooling for a small break LOCA and permit automatic depressurization.

The logic scheme used for initiating the ADS System is a single trip system that contains two logics (the A logic and the B logic). Each logic can initiate opening all four ADS valves to effect automatic depressurization. The A logic is powered from Division 1 of the 125 V DC System, while logic B can be powered from either Division of the 125 V DC System. Also, each ADS solenoid valve and its associated portion of the actuation logic can be powered from either Division of the 125 V DC System. Each logic is arranged such that all of the following must occur to cause ADS actuation:

(continued)

BACKGROUND

Automatic Depressurization System (continued)

1) two-out-of-two Reactor Vessel Water Level - Low Low switches must actuate, 2) the single Reactor Vessel Water Level - Low switch must actuate, 3) one-out-of-three taken twice Pump Discharge Pressure switches must actuate, and 4) the single ADS Timer must time out. The ADS Timer begins its timing sequence when the associated Reactor Vessel Water Level - Low Low Low and confirmatory Reactor vessel Water Level - Low signals are present. Once the ADS initiation signal is present, it is sealed in until manually reset. A reset pushbutton permits the ADS Timer to begin the timing sequence anew anytime before the Timer has timed out. ADS initiation can also be prevented by placing the reset handswitch in the Override position. Neither the reset pushbutton nor the reset handswitch is required for ADS OPERABILITY.

Diesel Generators

The DGs may be initiated by either automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level – Low Low Low or Drywell Pressure – High. The DGs are also initiated upon loss of voltage signals. (Refer to the Bases for LCO 3.3.8.1, "Loss of Power (LOP) Instrumentation," for a discussion of these signals.) Each of these diverse variables is monitored by four redundant instruments, which are, in turn, connected to four relays whose contacts are connected to a one-out-of-two taken twice logic to initiate both DGs 1G-31 and 1G-21. The DGs receive their initiation signals from the CS System initiation logic. The DGs can also be started manually from the control room and locally from the associated DG room. The DG initiation signal is a sealed in signal and can be manually reset. The DG initiation logic is reset by resetting the associated CS System initiation logic. Upon receipt of a Loss of Coolant Accident (LOCA) initiation signal, each DG is automatically started, is ready to load in approximately 10 seconds, and will run in standby conditions (rated voltage and speed, with the DG output breaker open). The DGs will only energize their respective Engineered Safety Feature buses if a loss of offsite power occurs (Refer to Bases for LCO 3.3.8.1.).

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The actions of the ECCS are explicitly assumed in the safety analyses of References 1, 2 and 3. The ECCS is initiated to preserve the integrity of the fuel cladding by limiting the post LOCA peak cladding temperature to less than the 10 CFR 50.46 limits.

ECCS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.

The OPERABILITY of the ECCS instrumentation is dependent upon the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.5.1-1. Each Function must have a required number of OPERABLE channels, with their setpoints within the specified Allowable Values, where appropriate. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Table 3.3.5.1-1 is modified by two footnotes. Footnote (a) is added to clarify that the associated Functions are required to be OPERABLE in MODES 4 and 5 only when their supported ECCS are required to be OPERABLE per LCO 3.5.2, ECCS-Shutdown. Footnote (b), is added to show that certain ECCS instrumentation Functions also perform DG initiation.

Allowable Values are specified for each ECCS Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between 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 waterlevel), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g.,on-off sensor or bi-stable trip circuit) changes state. Analytical Limits, where established, are the limiting values of the process parameters used in safety analysis to define the margin to unacceptable consequences. Margin is provided between the Allowable Value and the Analytical Limits to allow for process, calibration (i.e., M&TE) and some instrument uncertainties. Additional margin

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

is provided between the Allowable Value and the trip setpoint to allow for the remaining instrument uncertainties (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.

In general, the individual Functions are required to be OPERABLE in the MODES or other specified conditions that may require ECCS (or DG) initiation to mitigate the consequences of an abnormal operational transient or design basis accident. To ensure reliable ECCS and DG function, a combination of Functions is required to provide primary and secondary initiation signals.

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

Core Spray and Low Pressure Coolant Injection Systems

1.a, 2.a. Reactor Vessel Water Level – Low Low Low

Low Reactor Pressure Vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. The low pressure ECCS and associated DGs are initiated at Reactor Vessel Water Level-Low Low Low to ensure that core spray and flooding functions are available to prevent or minimize fuel damage. Reactor Vessel Water Level – Low Low Low is one of the Functions assumed to be OPERABLE and capable of initiating the ECCS during the transients analyzed in References 1 and 3. In addition, the Reactor Vessel Water Level – Low Low Low Function is directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

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BASES

APPLICABLE SAVETY ANALYSIS, LCO, and APPLICABILITY

1.a, 2.a Reactor Vessel Water Level - Low Low Low (continued)

Reactor Vessel Water Level – Low Low Low signals are initiated from four level switches 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. The Reactor Vessel Water Level – Low Low Low Allowable Value is chosen to allow time for the low pressure core flooding LCO, andsystems to activate and provide adequate cooling.

Four channels of Reactor Vessel Water Level – Low Low Low Function are only required to be OPERABLE when the ECCS are required to be OPERABLE to ensure that no single instrument failure can preclude ECCS initiation. Per Footnote (a) to Table 3.3.5.1-1, the Reactor Vessel Water Level-Low, Low, Low, Level 1 Function is only required to be OPERABLE in MODES 4 and 5 whenever the associated ECCS is required to be OPERABLE per LCO 3.5.2. Refer to LCO 3.5.1 and LCO 3.5.2, "ECCS – Shutdown," for Applicability Bases for the low pressure ECCS subsystems; LCO 3.8.1, "AC Sources – Operating"; and LCO 3.8.2, "AC Sources – Shutdown," for Applicability Bases for the DGs.

1.b, 2.b. Drywell Pressure – High

High pressure in the drywell could indicate a break in the Reactor Coolant Pressure Boundary (RCPB). The low pressure ECCS and associated DGs are initiated upon receipt of the Drywell Pressure – High Function in order to minimize the possibility of fuel damage. The Drywell Pressure – High Function, along with the Reactor Water Level – Low Low Low Function, is directly assumed in the analysis of the recirculation line break (Ref. 4). The core cooling function of the ECCS, along with the scram action of the RPS, 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 switches that sense drywell pressure. The Allowable Value was selected to be as low as possible and be indicative of a LOCA inside primary containment.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.b, 2.b. Drywell Pressure-High (continued)

The Drywell Pressure-High Function is required to be OPERABLE when the ECCS or DG is required to be OPERABLE in conjunction with times when the primary containment is required to be OPERABLE. Thus, four channels of the CS Drywell Pressure-High Function are required to be OPERABLE in MODES 1, 2, and 3 to ensure that no single instrument failure can preclude CS and DG initiation. Also, the LPCI Drywell Pressure - High Function is required to be OPERABLE in MODES 1, 2, and 3 to ensure no single instrument failure can preclude LPCI initiation. In MODES 4 and 5, the Drywell Pressure-High Function is not required, since there is insufficient energy in the reactor to pressurize the primary containment to Drywell Pressure-High setpoint. Refer to LCO 3.5.1 for Applicability Bases for the low pressure ECCS subsystems and to LCO 3.8.1 for Applicability Bases for the DGs.

1.c, 2.c. Reactor Steam Dome Pressure-Low (Injection Permissive)

Low reactor steam dome pressure signals are used as permissives for the low pressure ECCS subsystems. This ensures that, prior to opening the injection valves of the low pressure ECCS subsystems, the reactor pressure has fallen to a value below these subsystems' maximum design pressure. The Reactor Steam Dome Pressure-Low is one of the Functions assumed to be OPERABLE and capable of permitting initiation of the ECCS during the transients analyzed in References 1 and 3. In addition, the Reactor Steam Dome Pressure-Low Function is directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

The Reactor Steam Dome Pressure-Low signals are initiated from four pressure switches that sense the reactor dome pressure.

The Allowable Value is low enough to prevent over-pressurizing the equipment in the low pressure ECCS, but high enough to ensure that the ECCS injection prevents the fuel peak cladding temperature from exceeding the limits of 10 CFR 50.46.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.c, 2.c. Reactor Steam Dome Pressure—Low (Injection Permissive) (continued)

Four channels of Reactor Steam Dome Pressure—Low Function are only required to be OPERABLE when the ECCS is required to be OPERABLE to ensure that no single instrument failure can preclude ECCS initiation. Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the low pressure ECCS subsystems.

1.d, 2.f. Core Spray and Low Pressure Coolant Injection Pump Discharge Flow—Low (Bypass)

The minimum flow instruments are provided to protect the associated low pressure ECCS pump(s) from overheating when the pump is operating and the associated injection valve is not fully open. The minimum flow line valves (normally open for the CS System and normally closed for the LPCI System) receive an open signal when low flow is sensed, and automatically close when the flow rate is adequate to protect the associated pump. The LPCI and CS Pump Discharge Flow—Low Functions are assumed to be OPERABLE and capable of closing the minimum flow valves to ensure that the low pressure ECCS flow rates assumed during the transients and accidents analyzed in References 1, 2, and 3 are met. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

One flow switch per CS pump and one differential pressure switch for the two RHR pumps in each division are used to detect the associated subsystems' flow rates. The logic is arranged such that each differential pressure switch or flow switch causes its associated minimum flow valve to receive an open signal. The logic will close the minimum flow valve once the closure setpoint is exceeded. The LPCI minimum flow valves are time delayed such that the valves will not open for 10 seconds after the switches detect low flow. The time delay is provided by design to limit reactor vessel inventory loss during the startup of the RHR shutdown cooling mode although, typically, the minimum flow valves are prevented from opening when operating in the shutdown cooling mode. The Pump Discharge Flow—Low Allowable Values are high enough to ensure that the pump flow rate is sufficient to protect the pump, yet low enough to ensure

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.d, 2.f. Core Spray and Low Pressure Coolant Injection Pump Discharge Flow-Low (Bypass) (continued)

that the closure of the minimum flow valve is initiated to allow the assumed flow into the core. Each channel of Pump Discharge Flow – Low Function (two CS channels, one per pump and two LPCI channels, one per loop) is only required to be OPERABLE when the associated ECCS is required to be OPERABLE to ensure that no single instrument failure can preclude the ECCS function. Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the low pressure ECCS subsystems.

2.d. Reactor Vessel Shroud Level – Low

The Reactor Vessel Shroud Level-Low Function is provided as a permissive to allow the RHR System to be manually aligned from the LPCI mode to the suppression pool cooling/spray or drywell spray modes with a LPCI initiation signal still present. This function ensures: 1) that the permissive is removed prior to reaching two thirds core height when vessel level is decreasing, and 2) that the permissive is not restored until two thirds core height is reached when vessel level is increasing. This ensures that LPCI is available to prevent or minimize fuel damage. This function may be overridden during accident conditions as allowed by plant procedures to allow containment cooling/spray regardless of the level present in the shroud.

Reactor Vessel Shroud Level – Low signals are initiated from four level transmitters 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. The Reactor Vessel Shroud Level – Low Allowable Value is chosen to allow the low pressure core flooding systems to activate and provide adequate cooling before allowing a manual transfer.

Four channels of the Reactor Vessel Shroud Level – Low Function are only required to be OPERABLE in MODES 1, 2, and 3. In MODES 4 and 5, the specified initiation time of the LPCI subsystems is not assumed, and other administrative controls are adequate to control the valves that this Function isolates (since the containment cooling mode of RHR is not required to be OPERABLE in MODES 4 and 5 and is normally not used).

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.e, 2.e Core Spray Pump Startup - Time Delay Relay and Low Pressure Coolant Injection Pump Start - Time Delay Relay

The purpose of these time delay relays is to stagger the start of the CS pumps and LPCI pumps that are in each of Divisions 1 and 2, thus limiting the starting transients on the 4.16 kV emergency buses. This Function is only necessary when power is being supplied from the standby power sources (DG). However, since the time delay does not degrade ECCS operation, it remains in the pump start logic at all times. The CS Pump and LPCI Pump Start – Time Delay Relays are assumed to be OPERABLE in the accident and transient analyses requiring ECCS initiation. That is, the analyses assume that the pumps will initiate when required and excess loading will not cause failure of the standby power sources.

There are two CS Pump Start - Time Delay Relays and four LPCI Pump Start - Time Delay Relays, one in each of the CS and RHR pump start logic circuits. While each time delay relay is dedicated to a single pump start logic, a single failure of a CS Pump Start - Time Delay Relay or of a LPCI Pump Start - Time Delay Relay could result in the failure of the three low pressure ECCS pumps, powered from the same emergency bus if the emergency bus is being powered by its associated DG, to perform their intended function within the assumed time (e.g., as in the case where two or more ECCS pumps on one emergency bus start simultaneously due to an inoperable time delay relay which, in turn cause the associated DG output breaker to trip open due to undervoltage conditions). This still leaves three of the six low pressure ECCS pumps OPERABLE; thus, the single failure criterion is met (i.e., loss of one instrument does not preclude ECCS initiation). The Allowable Value for the CS Pump Start-Time Delay Relay and each LPCI Pump Start – Time Delay Relay is chosen to be long enough so that most of the starting transient of one pump is complete before starting another pump on the same 4.16 kV emergency bus and short enough so that ECCS operation is not degraded.

Each CS Pump Start - Time Delay Relay and each LPCI Pump Start - Time Delay Relay Function is required to be OPERABLE only when the associated CS or LPCI System is required to be OPERABLE. Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the CS and LPCI Systems.

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BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY (continued)

1.f, 2.k 4.16 kV Emergency Bus Sequential Loading Relay

An undervoltage condition on an emergency bus indicates that sufficient power (from either the offsite or onsite sources) is not available to allow starting of the low pressure ECCS pumps. Therefore, if this condition exists, the start permissive signal is withheld from the circuits that start both the Core Spray (CS) and Low Pressure Coolant Injection (LPCI) pumps during accident conditions, and the CS and LPCI pumps powered from the respective emergency bus are prevented from starting. This ensures that the low pressure ECCS pumps are not started during accident conditions unless adequate power is available.

Each emergency bus is monitored by a single relay, which inputs into a one-out-of-two once logic for each division. Each logic channel supports one CS and two LPCI pumps. An instrument channel consists of the common bus monitoring relay and the associated relay contacts for each ECCS pump.

The 4.16 kV Emergency Bus Sequential Loading Relay Allowable Values are low enough to prevent low pressure ECCS pump starting unless adequate power is available, but high enough so that low pressure ECCS pump starting is not unnecessarily prohibited or delayed and is within the maximum adjustable range of the relay.

To ensure that no single failure can prevent successful operation of the combined low pressure ECCS, two channels of the 4.16 kV Emergency Bus Sequential Loading Relay Function are required to be OPERABLE whenever the LPCI System is required to be OPERABLE (i.e., MODES 1, 2, and 3), and one channel is required to be OPERABLE whenever the associated LPCI pump(s) is required to be OPERABLE by LCO 3.5.2, "ECCS - Shutdown." One channel of 4.16 kV Emergency Bus Sequential Loading Relay Function is required to be OPERABLE whenever the associated CS subsystem (i.e., pump) is required to be OPERABLE. To ensure this Function is available when required, the 4.16 kV Emergency Bus Sequential Loading Relay is required to be OPERABLE in MODES 1, 2, and 3, and when the associated low pressure ECCS is required to be OPERABLE by LCO 3.5.2, "ECCS - Shutdown." Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the CS and LPCI Systems.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.g. LPCI Loop Select - Reactor Vessel Water Level - Low Low

The purpose of the LPCI Loop Select Reactor Vessel Water Level - Low Low Function is to initiate the LPCI Loop Select Logic at a level slightly higher than the level that starts the low pressure ECCS pumps (the LPCI Loop Select Logic can also be initiated simultaneously with low pressure ECCS pump starting if the first indication of a LOCA is high drywell pressure). This Function is only required to be OPERABLE for the DBA LOCA analysis, i.e., if the break location is in the recirculation system suction piping (Ref. 2). For a DBA LOCA, the analysis assumes that the LPCI Loop Select Logic successfully identifies and directs LPCI flow to the unbroken recirculation loop so that core reflooding is accomplished in time to ensure that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. For other LOCA events, (i.e., non-DBA recirculation system pipe breaks or other RPV pipe breaks), the success of the Loop Select Logic is less critical than for the DBA.

LPCI Loop Select Reactor Vessel Water Level - Low Low signals are initiated from four level switches that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water (variable leg) in the vessel.

The LPCI Loop Select Reactor Vessel Water Level - Low Low Allowable Value is chosen to be slightly higher than the level that starts the low pressure ECCS pumps so that certain actions associated with the LPCI function are accomplished in time to effect rapid core reflooding when vessel level is dropping rapidly, i.e., during a DBA LOCA.

Four channels of LPCI Loop Select Reactor Vessel Water Level -Low Low Function are only required to be OPERABLE in Modes 1, 2, and 3 to ensure that no single failure can preclude LPCI Loop Select Logic initiation. This Function is not required to be OPERABLE in MODES 4 and 5 because, in those MODES, the LPCI System is considered to consist of two independent subsystems (Refer to BASES for LCO 3.5.2).

2.h LPCI Loop Select - Reactor Steam Dome Pressure - Low

The purpose of the LPCI Loop Select Reactor Steam Dome Pressure - Low Function is to optimize the LPCI Loop Select Logic sensitivity if the logic previously actuated

(continued)

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.h LPCI Loop Select - Reactor Steam Dome Pressure - Low (continued)

recirculation pump trips. This is accomplished by preventing the logic from continuing on to the unbroken loop selection activity until reactor steam dome pressure has dropped below a specified value. This Function is only required to be OPERABLE for the DBA LOCA analysis, i.e., if the break location is in the recirculation system suction piping (Ref. 2). For a DBA LOCA, the analysis assumes that the LPCI Loop Select Logic successfully identifies and directs LPCI flow to the unbroken recirculation loop so that core reflooding is accomplished in time to ensure that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. For other LOCA events, (i.e., non-DBA recirculation system pipe breaks), or other RPV pipe breaks the success of the Loop Select Logic is less critical than for the DBA.

LPCI Loop Select Reactor Steam Dome Pressure - Low signals are initiated from four pressure switches that sense the reactor steam dome pressure.

The Allowable Value is chosen to optimize the sensitivity of the portion of the LPCI Loop Select Logic that selects the unbroken recirculation loop for LPCI injection, while ensuring that LPCI injection is not unnecessarily delayed.

Four channels of LPCI Loop Select Reactor Steam Dome Pressure - Low Function are only required to be OPERABLE in MODES 1, 2, and 3 to ensure that no single failure can prevent the LPCI Loop Select Logic from successfully selecting the unbroken recirculation loop for LPCI injection. This Function is not required to be OPERABLE in MODES 4 and 5 because, in those MODES, the LPCI System is considered to consist of two independent subsystems (Refer to BASES for LCO 3.5.2).

2.i. LPCI Loop Select - Recirculation Pump Differential Pressure

Recirculation Pump Differential Pressure signals are used by the LPCI Loop Select Logic to determine if either recirculation pump is running. If either pump is not running, i.e., Single Loop Operation, the logic sends a trip signal to both recirculation pumps. This is necessary to

(continued)

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APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY

2.i. LPCI Loop Select - Recirculation Pump Differential Pressure (continued)

eliminate the possibility of small pipe breaks being masked by a running recirculation pump. This Function is only required to be OPERABLE for the DBA LOCA analysis, i.e., if the break location is in the recirculation system suction piping (Ref. 2). For a DBA LOCA, the analysis assumes that the LPCI Loop Select Logic successfully identifies and directs LPCI flow to the unbroken recirculation loop so that core reflooding is accomplished in time to ensure that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. For other LOCA events (i.e., non-DBA recirculation system pipe breaks or other RPV pipe breaks), the success of the Loop Select Logic is less critical than for the DBA.

LPCI Loop Select Recirculation Pump Differential Pressure signals are initiated from eight differential pressure switches, four of which sense the pressure differential between the suction and discharge of each recirculation pump.

The Allowable Value is chosen to be as low as possible, while still maintaining the ability to differentiate between a running and non-running recirculation pump.

Eight channels of LPCI Loop Select - Recirculation Pump Differential Pressure Function are only required to be OPERABLE in MODES 1, 2, and 3 to ensure that no single failure can prevent the LPCI Loop Select Logic from successfully determining if either recirculation pump is running. This Function is not required to be OPERABLE in MODES 4 and 5 because, in those MODES, the LPCI System is considered to consist of two independent subsystems. (Refer to BASES for LCO 3.5.2).

2.j. LPCI Loop Select - Recirculation Riser Differential Pressure

Recirculation Riser Differential Pressure signals are used by the LPCI Loop Select Logic to determine which, if any, recirculation loop is broken. This is accomplished by comparing the absolute pressure of the two recirculation loops. A broken loop will be indicated by a lower pressure

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APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY

2.j. LPCI Loop Select - Recirculation Riser Differential Pressure (continued)

than an unbroken loop. The loop with the higher pressure is then selected for LPCI injection. If neither loop is broken, the logic defaults to injecting into the "B" recirculation loop. This Function is only required to be OPERABLE for the DBA LOCA analysis, i.e., if the break location is in the recirculation system suction piping (Ref. 2). For a DBA LOCA, the analysis assumes that the LPCI Loop Select Logic successfully identifies and directs LPCI flow to the unbroken recirculation LOCA, the analysis assumes that the LPCI Loop Select Logic successfully identifies and directs LPCI flow to the unbroken recirculation loop so that core reflooding is accomplished in time to ensure that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. For other LOCA events, (i.e., non-DBA recirculation system pipe breaks), or other RPV pipe breaks, the success of the Loop Select Logic is less critical than for the DBA.

LPCI Loop Select Recirculation Riser Differential Pressure signals are initiated from four differential pressure switches that sense the pressure differential between the A recirculation loop riser and the B recirculation loop riser. If, after a small time delay, the pressure in loop A is not indicating higher than loop B pressure, the logic will select the B loop for injection. If recirculation loop A pressure is indicating higher than loop B pressure, the logic will select the A loop for LPCI injection.

The Allowable Value is chosen to be as low as possible, while still maintaining the ability to differentiate between a broken and unbroken recirculation loop.

Four channels of LPCI Loop Select Recirculation Riser Differential Pressure Function are only required to be OPERABLE in MODES 1, 2, and 3 to ensure that no single failure can prevent the LPCI Loop Select Logic from successfully selecting the unbroken recirculation loop for LPCI injection. This Function is not required to be OPERABLE in MODES 4 and 5 because, in those MODES, the LPCI System is considered to consist of two independent subsystems. (Refer to BASES for LCO 3.5.2).

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY (continued)

HPCI System

3.a. Reactor Vessel Water Level - Low Low

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, the HPCI System is initiated at Reactor Vessel Water Level - Low Low to maintain level above the top of the active fuel. Reactor Vessel Water Level - Low Low is one of the Functions assumed to be OPERABLE and capable of initiating HPCI during the transients analyzed in References 1 and 3. Additionally, the Reactor Vessel Water Level - Low Low Function associated with HPCI is directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Vessel Water Level – Low Low signals are initiated from four level switches 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.

The Reactor Vessel Water Level – Low Low Allowable Value is high enough such that for a small break LOCA, the HPCI System will be sufficient to avoid initiation of the ADS System at Reactor Vessel Water Level – Low Low Low.

Four channels of Reactor Vessel Water Level – Low Low Function are required to be OPERABLE only when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI initiation. Refer to LCO 3.5.1 for HPCI Applicability Bases.

3.b. Drywell Pressure – High

High pressure in the drywell could indicate a break in the RCPB. The HPCI System is initiated upon receipt of the Drywell Pressure – High Function in order to minimize the possibility of fuel damage. The Drywell Pressure – High Function, along with the Reactor Water Level – Low Low Function, is directly assumed in the analysis of the recirculation line break (Ref. 4). The core cooling

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY

3.b. Drywell Pressure - High (continued)

function of the ECCS, along with the scram action of the RPS, 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 switches that sense drywell pressure. The Allowable Value was selected to be as low as possible to be indicative of a LOCA inside primary containment.

Four channels of the Drywell Pressure – High Function are required to be OPERABLE when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI initiation. Refer to LCO 3.5.1 for the Applicability Bases for the HPCI System.

3.c. Reactor Vessel Water Level – High

High RPV water level indicates that sufficient cooling water inventory exists in the reactor vessel such that there is no danger to the fuel. Therefore, the Reactor Vessel Water Level - High signal is used to trip the HPCI turbine to prevent overflow into the Main Steam Lines (MSLs). The Reactor Vessel Water Level – High Function is not assumed in the accident and transient analyses. It was retained since it is a potentially significant contributor to risk, thus it meets Criterion 4 of 10 CFR 50.36(c)(2)(ii).

Reactor Vessel Water Level – High signals for HPCI are initiated from two level switches. Both Reactor Vessel Water Level - High signals are required in order to trip the HPCI turbine. This ensures that no single instrument failure can preclude HPCI initiation. The Reactor Vessel Water Level – High Allowable Value is chosen to prevent flow from the HPCI System from overflowing into the MSLs.

Two channels of Reactor Vessel Water Level – High Function are required to be OPERABLE only when HPCI is required to be OPERABLE. Refer to LCO 3.5.1 for HPCI Applicability Bases.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3.d. Condensate Storage Tank Level - Low

Low level in the CST indicates the unavailability of an adequate supply of makeup water from this preferred source. Normally the suction valves between HPCI and the CST are open and, upon receiving a HPCI initiation signal, water for HPCI injection would be taken from the CST. However, if the water level in the CST falls below a preselected level, first the suppression pool suction valves automatically open, and then the CST suction valve automatically closes. This ensures that an adequate supply of makeup water from a safety related source is available to the HPCI pump. To prevent losing suction to the pump, the suction valves are interlocked so that the suppression pool suction valves must be open before the CST suction valve automatically closes. The Function is implicitly assumed in the accident and transient analyses (which take credit for HPCI) since the analyses assume that the HPCI suction source is the suppression pool.

Condensate Storage Tank Level - Low signals are initiated from two level switches. The logic is arranged such that either level switch can cause the suppression pool suction valves to open and the CST suction valve to close. The Condensate Storage Tank Level - Low Function Allowable Value is high enough to ensure adequate pump suction head while water is being taken from the CST.

Two channels of the Condensate Storage Tank Level - Low Function are required to be OPERABLE only when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI suction transfer to suppression pool source. Refer to LCO 3.5.1 for HPCI Applicability Bases.

3.e. Suppression Pool Water Level - High

Excessively high suppression pool water could result in the loads on the suppression pool exceeding design values should there be a blowdown of the reactor vessel pressure through the safety relief valves. Therefore, signals indicating high suppression pool water level are used to transfer the suction source of HPCI from the CST to the suppression pool to eliminate the possibility of HPCI continuing to provide additional water from a source outside containment. To prevent losing suction to the pump, the suction valves are

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.e. Suppression Pool Water Level - High (continued)

interlocked so that the suppression pool suction valves must be open before the CST suction valve automatically closes. This Function is implicitly assumed in the accident and transient analyses (which take credit for HPCI) since the analyses assume that the HPCI suction source is the suppression pool.

Suppression Pool Water Level - High signals are initiated from two level switches. The logic is arranged such that either switch can cause the suppression pool suction valves to open and the CST suction valve to close. The Allowable Value for the Suppression Pool Water Level - High Function is chosen to ensure that HPCI will be aligned for suction from the suppression pool before the water level reaches the point at which suppression pool design loads would be exceeded.

Two channels of Suppression Pool Water Level - High Function are required to be OPERABLE only when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI suction transfer to suppression pool source. Refer to LCO 3.5.1 for HPCI Applicability Bases.

3.f. High Pressure Coolant Injection Pump Discharge Flow - Low (Bypass)

The minimum flow instruments are provided to protect the HPCI pump from overheating when the pump is operating and the associated injection valve is not fully open. The minimum flow line valve is opened when low flow is sensed, and the valve is automatically closed when the flow rate is adequate to protect the pump. The High Pressure Coolant Injection Pump Discharge Flow - Low Function is assumed to be OPERABLE and capable of closing the minimum flow valve to ensure that the ECCS flow assumed during the transients and accidents analyzed in References 1, 2, 3 and 4 are met. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

One flow switch is used to detect the HPCI System's flow rate. The logic is arranged such that the switch causes the minimum flow valve to open if flow is low and pump discharge

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY

3.f High Pressure Coolant Injection Pump Discharge Flow - Low (Bypass) (continued)

pressure is above a specified valve. The logic will close the minimum flow valve once the flow rate closure setpoint is exceeded. The High Pressure Coolant Injection Pump Discharge Flow—Low Allowable Value is high enough to ensure that pump flow rate is sufficient to protect the pump, yet low enough to ensure that the closure of the minimum flow valve is initiated to allow the assumed flow into the core.

One channel is required to be OPERABLE when HPCI is required to be OPERABLE. Refer to LCO 3.5.1 for HPCI Applicability Bases.

Automatic Depressurization System

4.a, 5.a. Reactor Vessel Water Level—Low Low Low

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, ADS receives one of the signals necessary for initiation from this Function. The Reactor Vessel Water Level—Low Low Low is one of the Functions assumed to be OPERABLE and capable of initiating the ADS during the accident analyzed in Reference 2. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Vessel Water Level—Low Low Low signals are initiated from four level switches 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 Low Low Function are required to be OPERABLE only when ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. Two channels input to ADS trip logic A, while the other two channels input to ADS trip logic B. Refer to LCO 3.5.1 for ADS Applicability Bases.

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY

4.a, 5.a. Reactor Vessel Water Level – Low Low Low (continued)

The Reactor Vessel Water Level – Low Low Low Allowable Value is chosen to allow time for the low pressure core flooding systems to initiate and provide adequate cooling.

4.b, 5.b. Automatic Depressurization System Timer

The purpose of the Automatic Depressurization System Timer is to delay depressurization of the reactor vessel to allow the HPCI System time to maintain reactor vessel water level. Since the rapid depressurization caused by ADS operation is one of the most severe transients on the reactor vessel, its occurrence should be limited. By delaying initiation of the ADS Function, the operator is given the chance to monitor the success or failure of the HPCI System to maintain water level, and then to decide whether or not to allow ADS to initiate, to delay initiation further by recycling the timer, or to inhibit initiation permanently. The Automatic Depressurization System Timer Function is assumed to be OPERABLE for the accident analyses of Reference 2 that require ECCS initiation and assume failure of the HPCI System.

There are two Automatic Depressurization System Timer relays, one in each of the two ADS trip logics. The Allowable Value for the Automatic Depressurization System Timer is based upon a nominal setting of 2 minutes with an allowance for drift. The nominal setting of 2 minutes is chosen so that there is still time after depressurization for the low pressure ECCS subsystems to provide adequate core cooling.

Two channels of the Automatic Depressurization System Timer Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. (One channel inputs to ADS trip logic A, while the other channel inputs to ADS trip logic B.) Refer to LCO 3.5.1 for ADS Applicability Bases.

4.c, 5.c. Reactor Vessel Water Level - Low (Confirmatory)

The Reactor Vessel Water Level - Low Function is used by the ADS only as a confirmatory low water level signal. ADS receives one of the signals necessary for initiation from

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY

4.c, 5.c. Reactor Vessel Water Level - Low (Confirmatory) (continued)

Reactor Vessel Water Level - Low Low Low signals. In order to prevent spurious initiation of the ADS due to a single instrument line break, a Reactor Vessel Water Level - Low signal must also be received before ADS initiation commences.

Reactor Vessel Water Level - Low signals are initiated from two level switches 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. The Allowable Value for Reactor Vessel Water Level - Low is selected at the RPS Reactor Vessel Water Level - Low scram Allowable Value for convenience. Refer to LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," for the Bases discussion of this Function.

Two channels of Reactor Vessel Water Level - Low Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. One channel inputs to ADS trip logic system A, while the other channel inputs to ADS trip logic B. Refer to LCO 3.5.1 for ADS Applicability Bases.

4.d, 4.e, 5.d, 5.e. Core Spray and Low Pressure Coolant Injection Pump Discharge Pressure - High

The Pump Discharge Pressure - High signals from the CS and LPCI pumps are used as permissives for ADS initiation, indicating that there is a source of low pressure cooling water available once the ADS has depressurized the vessel. Pump Discharge Pressure - High is one of the Functions assumed to be OPERABLE and capable of permitting ADS initiation during the events analyzed in Reference 2 with an assumed HPCI failure. For these events the ADS depressurizes the reactor vessel so that the low pressure ECCS can perform the core cooling functions. This core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Pump discharge pressure signals are initiated from twelve pressure switches, two on the discharge side of each of the six low pressure ECCS pumps. In order to generate an ADS permissive in one trip logic, it is necessary that one of

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

<u>4.d, 4.e, 5.d, 5.e. Core Spray and Low Pressure Coolant Injection Pump Discharge Pressure—High</u> (continued)

the channels from the discharge of the three low pressure ECCS pumps in the same division indicate high discharge pressure, and one of the redundant channels from the same three pumps also indicate high discharge pressure (i.e. one-out-of three taken twice). The Pump Discharge Pressure—High Allowable Value is less than the pump discharge pressure when one pump is operating in minimum bypass flow mode and high enough to avoid any condition that results in a discharge pressure permissive when the CS and LPCI pumps are aligned for injection and the pumps are not running. The actual operating point of this function is not assumed in any transient or accident analysis.

Twelve channels of Core Spray and Low Pressure Coolant Injection Pump Discharge Pressure—High Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. Two CS channels associated with CS pump A and four LPCI channels associated with LPCI pumps A and C are required for trip logic A. Two CS channels associated with CS pump B and four LPCI channels associated with LPCI pumps B and D are required for trip logic B. Refer to LCO 3.5.1 for ADS Applicability Bases.

ACTIONS

A Note has been provided to modify the ACTIONS related to ECCS 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 ECCS instrumentation channels provide appropriate compensatory measures for separate inoperable Condition entry for each inoperable ECCS instrumentation channel.

(continued)

BASES

ACTIONS (continued)

A.1

Required Action A.1 directs entry into the appropriate Condition referenced in Table 3.3.5.1-1. The applicable Condition referenced in the table is Function dependent. Each time a channel is discovered inoperable, Condition A is entered for that channel and provides for transfer to the appropriate subsequent Condition.

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

Required Actions B.1 and B.2 are intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in redundant automatic initiation capability being lost for the feature(s). Required Action B.1 features would be those that are initiated by Functions 1.a, 1.b, 2.a, and 2.b (e.g., low pressure ECCS). The Required Action B.2 system would be HPCI. For Required Action B.1, redundant automatic initiation capability is lost if certain combinations of two or more Function 1.a, 1.b, 2.a or 2.b channels are inoperable and untripped in more than one trip system. Since there are three redundant low pressure ECCS subsystems (i.e., CS 'A', CS 'B', and the LPCI subsystem), Required Action B.1 is only applicable if initiation capability is lost in either: 1) both CS subsystems, or 2) either of the CS subsystems and the LPCI subsystem. For low pressure ECCS, since each inoperable channel would have Required Action B.1 applied separately (refer to ACTIONS Note), each inoperable channel would only require the affected portion of the associated system of low pressure ECCS and DGs to be declared inoperable. However, since channels that provide input to initiation logics associated with two or more low pressure ECCS subsystems (i.e., both CS subsystems or either CS subsystem in combination with the LPCI subsystem) are inoperable and untripped, and the Completion Times started concurrently for the channels associated with two or more low pressure ECCS subsystems, this results in the affected portions in the associated low pressure ECCS and DGs being concurrently declared inoperable.

(continued)

BASES

ACTIONS

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

For Required Action B.2, automatic initiation capability is lost if certain combinations of Function 3.a or Function 3.b channels are inoperable and untripped. In this situation (loss of automatic initiation capability), the 24 hour allowance of Required Action B.3 is not appropriate and the feature(s) associated with the inoperable, untripped channels must be declared inoperable within 1 hour. As noted (Note 1 to Required Action B.1), Required Action B.1 is only applicable in MODES 1, 2, and 3. In MODES 4 and 5, the specific initiation time of the low pressure ECCS is not assumed and the probability of a LOCA is lower. Thus, a total loss of initiation capability for 24 hours (as allowed by Required Action B.3) is allowed during MODES 4 and 5. There is no similar Note provided for Required Action B.2 since HPCI instrumentation is not required in MODES 4 and 5; thus, a Note is not necessary.

Notes are also provided (Note 2 to Required Action B.1 and the Note to Required Action B.2) to delineate which Required Action is applicable for each Function that requires entry into Condition B if an associated channel is inoperable. This ensures that the proper loss of initiation capability check is performed. Required Action B.1 (the Required Action for certain inoperable channels in the low pressure ECCS subsystems) is not applicable to Function 2.d, since this Function provides backup to administrative controls ensuring that operators do not divert LPCI flow from injecting into the core when needed. Thus, a total loss of Function 2.d capability for 24 hours is allowed, since the LPCI System remains capable of performing its intended function.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowable out of service time "clock." For Required Action B.1, the Completion Time only begins upon discovery that a redundant feature in two or more low pressure ECCS subsystems (i.e., both CS subsystems or either CS subsystem in combination with the LPCI subsystem) cannot automatically initiate their supported features due to inoperable, untripped channels within the same Function as described in the paragraph above.

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BASES

ACTIONS

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

For Required Action B.2, the Completion Time only begins upon discovery that the HPCI System cannot be automatically initiated due to the required combination of two inoperable, untripped channels for the associated Function. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 5) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action B.3. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition I must be entered and its Required Action taken.

C.1, C.2 and C.3

Required Actions C.1 and C.2 are intended to ensure that appropriate actions are taken if multiple, inoperable channels within the same Function result in automatic initiation capability being lost for redundant features either in both divisions of LPCI Loop Select Logic or in two or more low pressure ECCS subsystems. Required Action C.1 features would be those that are initiated by Functions 1.c, 2.c, 1.e, and 2.e (i.e., low pressure ECCS). Required Action C.2 would be the feature affected by Functions 2.g, 2.h, 2.i, and 2.j (i.e., the LPCI Loop Select Logic). Redundant automatic initiation capability is lost if certain combinations of either (a) two Function 1.c or 2.c channels are inoperable in two or more low pressure ECCS trip systems, (b) two Function 2.g channels are inoperable in both trip systems, (c) two Function 2.h channels are inoperable in both trip systems,

(continued)

BASES

ACTIONS

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

(d) two or more Function 2.i channels for the same recirculation pump are inoperable in both trip systems, (e) two Function 2.j channels are inoperable in both trip systems, or (f) two or more Function 1.e or 2.e channels are inoperable in different divisions. In these situations (loss of redundant automatic initiation capability), the 24 hour allowance of Required Action C.3 is not appropriate and the feature(s) associated with the inoperable channels must be declared inoperable within 1 hour. Since each inoperable channel would have Required Action C.1 or C.2, as appropriate, applied separately (refer to ACTIONS Note), each inoperable channel would only require the affected portion of the associated system to be declared inoperable. However, since channels for either both divisions of LPCI loop select logic or for two or more low pressure ECCS subsystems are inoperable (e.g., both CS subsystems or either CS subsystem in combination with the LPCI subsystem), and the Completion Times started concurrently for the channels in both subsystems, this results in the affected portions in both subsystems being concurrently declared inoperable. For Functions 1.c, 1.e, and 2.e, the affected portions are the associated low pressure ECCS pumps. For Functions 2.g, 2.h, 2.i, and 2.j, the affected portion is the LPCI subsystem. As noted (Note 1), Required Actions C.1 and C.2 are only applicable in MODES 1, 2, and 3. In MODES 4 and 5, the specific initiation time of the ECCS is not assumed and the probability of a LOCA is lower. Thus, a total loss of automatic initiation capability for 24 hours (as allowed by Required Action C.3) is allowed during MODES 4 and 5.

Note 2 to Required Action C.1 states that Required Action C.1 is only applicable for Functions 1.c, 2.c, 1.e, and 2.e. Note 2 to Required Action C.2 states that Required Action C.2 is only applicable for Functions 2.g, 2.h, 2.i, and 2.j. Required Actions C.1 and C.2 are not applicable to Function 3.c (which also requires entry into this Condition if a channel in this Function is inoperable), since the loss of one channel results in a loss of the Function (two-out-of-two logic). This loss was considered during the development of Reference 5 and considered acceptable for the 24 hours allowed by Required Action C.3.

(continued)

BASES

ACTIONS

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

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowable out of service time "clock." For Required Actions C.1 and C.2, the Completion Time only begins upon discovery that the same feature, either in both divisions of LPCI Loop Select Logic or in two or more low pressure ECCS subsystems, cannot be automatically initiated due to inoperable channels within the same Function as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 5) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, Condition I must be entered and its Required Action taken. The Required Actions do not allow placing the channel in trip since this action would either cause the initiation or it would not necessarily result in a safe state for the channel in all events.

D.1, D.2.1, and D.2.2

Required Action D.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in a complete loss of automatic suction transfer capability for the HPCI System. Automatic suction transfer capability is lost if two Function 3.d channels or two Function 3.e channels are inoperable and untripped. In this situation (loss of automatic suction swap), the 24 hour allowance of Required Actions D.2.1 and D.2.2 is not appropriate and the HPCI System must be declared inoperable within 1 hour after discovery of loss of HPCI suction transfer capability. As noted, Required Action D.1 is only applicable if the HPCI pump suction is not aligned to the suppression pool, since, if aligned, the Function is already performed.

(continued)

BASES

ACTIONS

D.1, D.2.1 and D.2.2 (continued)

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowable out of service time "clock." For Required Action D.1, the Completion Time only begins upon discovery that the HPCI System cannot be automatically aligned to the suppression pool due to two inoperable, untripped channels in the same Function. The 1 hour Completion Time from discovery of loss of suction transfer capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 5) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action D.2.1 or the suction source must be aligned to the suppression pool per Required Action D.2.2. Placing the inoperable channel in trip performs the intended function of the channel (shifting the suction source to the suppression pool). Performance of either of these two Required Actions will allow operation to continue. If Required Action D.2.1 or D.2.2 is performed, measures should be taken to ensure that the HPCI System piping remains filled with water. Alternately, if it is not desired to perform Required Actions D.2.1 and D.2.2 (e.g., as in the case where shifting the suction source could drain down the HPCI suction piping), Condition I must be entered and its Required Action taken.

(continued)

BASES

ACTIONS (continued)

E.1 and E.2

Required Action E.1 is intended to ensure that appropriate actions are taken if multiple, inoperable channels within the Core Spray and Low Pressure Coolant Injection Pump Discharge Flow—Low Bypass Functions result in redundant automatic initiation capability being lost for the feature(s). For Required Action E.1, the features would be those that are initiated by Functions 1.d and 2.f (e.g., low pressure ECCS). Redundant automatic initiation capability is lost if either the automatic opening or closing function for two or more low pressure ECCS minimum flow valves is inoperable. Since each of the four minimum flow valves is initiated by a corresponding instrument channel, redundant automatic initiation capability is lost if any two of the four Function 1.d and 2.f channels are inoperable. Since each inoperable channel would have Required Action E.1 applied separately (refer to ACTIONS Note), each inoperable channel would only require the affected low pressure ECCS pump(s) to be declared inoperable. However, since channels for two or more minimum flow valves in the low pressure ECCS subsystems are inoperable, and the Completion Times started concurrently for the channels of the low pressure ECCS minimum flow valves, this results in the affected low pressure ECCS pump(s) being concurrently declared inoperable.

In this situation (loss of redundant automatic initiation capability), the 7 day allowance of Required Action E.2 is not appropriate and the system or subsystem associated with each inoperable channel must be declared inoperable within 1 hour. As noted (Note 1 to Required Action E.1), Required Action E.1 is only applicable in MODES 1, 2, and 3. In MODES 4 and 5, the specific initiation time of the ECCS is not assumed and the probability of a LOCA is lower. Thus, a total loss of initiation capability for 7 days (as allowed by Required Action E.2) is allowed during MODES 4 and 5. A Note is also provided (Note 2 to Required Action E.1) to delineate that Required Action E.1 is only applicable to low pressure ECCS Functions. Required Action E.1 is not applicable to HPCI Function 3.f since the loss of one channel results in a loss of the Function (one-out-of-one logic). This loss was considered during the development of Reference 5 and considered acceptable for the 7 days allowed by Required Action E.2.

(continued)

BASES

ACTIONS

E.1 and E.2 (continued)

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowable out of service time "clock."

For Required Action E.1, the Completion Time only begins upon discovery that redundant automatic initiation capability for two or more low pressure ECCS minimum flow valves is inoperable due to inoperable channels as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration of channels.

If the instrumentation that controls the pump minimum flow valve is inoperable, such that the valve will not automatically open, extended pump operation with no injection path available could lead to pump overheating and failure. If there were a failure of the instrumentation, such that the valve would not automatically close, a portion of the subsystem flow could be diverted from the reactor vessel injection path, causing insufficient core cooling. These consequences can be averted by the operator's manual control of the valve, which would be adequate to maintain ECCS pump protection and required flow. However, remote manual valve control can only be accomplished with temporary logic modifications during this period. Furthermore, other ECCS pumps would be sufficient to complete the assumed safety function if no additional single failure were to occur. The 7 day Completion Time of Required Action E.2 to restore the inoperable channel to OPERABLE status is reasonable based on the remaining capability of the associated ECCS subsystems, the redundancy available in the ECCS design, and the low probability of a DBA occurring during the allowed out of service time. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, Condition I must be entered and its Required Action taken. The Required Actions do not allow placing the channel in trip since this action would not necessarily result in a safe state for the channel in all events.

(continued)

BASES

ACTIONS (continued)

F.1

With a Function 1.f (2.k) channel inoperable, the 4.16 kV Emergency Bus Sequential Loading Relay is not cable of providing a start permissive signal for the low pressure ECCS pumps in the affected division, and the associated low pressure ECCS are not capable of performing their intended functions. Placing a channel in the tripped condition makes the associated ECCS pump(s) inoperable, since the pump(s) is prevented from automatically starting. In fact, tripping the bus power monitor relay will cause the associated DG to start, as it is common to the bus undervoltage logic. Consequently, one hour is provided to restore OPERABILITY of the channel. Otherwise, the affected low pressure ECCS (e.g., the CS subsystem in the affected division, the LPCI System if either division is affected in Modes 1, 2 or 3, and the LPCI pump(s) in the affected division for Modes 4 and 5) are declared inoperable immediately. This requires entry into LCO 3.5.1 or LCO 3.5.2, which provide appropriate actions for inoperable low pressure ECCS Systems and subsystems.

G.1 and G.2

Required Action G.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within similar ADS trip logic A and B Functions result in redundant automatic initiation capability being lost for the ADS. Redundant automatic initiation capability is lost if either (a) one Function 4.a channel and one Function 5.a channel are inoperable and untripped, or (b) one Function 4.c channel and one Function 5.c channel are inoperable and untripped.

In this situation (loss of automatic initiation capability), the 96 hour or 8 day allowance, as applicable, of Required Action G.2 is not appropriate and all ADS valves must be declared inoperable within 1 hour after discovery of loss of ADS initiation capability.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowable out of service time "clock."

(continued)

BASES

ACTIONS

G.1 and G.2 (continued)

For Required Action G.1, the Completion Time only begins upon discovery that the ADS cannot be automatically initiated due to inoperable, untripped channels within similar ADS trip system Functions as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 8 days has been shown to be acceptable (Ref. 5) to permit restoration of any inoperable channel to OPERABLE status if both HPCI and RCIC are OPERABLE. If either HPCI or RCIC is inoperable, the time is shortened to 96 hours. If the status of HPCI or RCIC changes such that the Completion Time changes from 8 days to 96 hours, the 96 hours begins upon discovery of HPCI or RCIC inoperability. However, the total time for an inoperable, untripped channel cannot exceed 8 days. If the status of HPCI or RCIC changes such that the Completion Time changes from 96 hours to 8 days, the "time zero" for beginning the 8 day "clock" begins upon discovery of the inoperable, untripped channel. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action G.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition I must be entered and its Required Action taken.

(continued)

BASES

ACTIONS (continued)

H.1 and H.2

Required Action H.1 is intended to ensure that appropriate actions are taken if multiple, inoperable channels within similar ADS trip system Functions result in automatic initiation capability being lost for the ADS. Automatic initiation capability is lost if either (a) one Function 4.b channel and one Function 5.b channel are inoperable or (b) a combination of Function 4.d, 4.e, 5.d, and 5.e channels are inoperable such that both channels associated with three low pressure ECCS pumps in the same division are inoperable.

In this situation (loss of automatic initiation capability), the 96 hour or 8 day allowance, as applicable, of Required Action H.2 is not appropriate, and all ADS valves must be declared inoperable within 1 hour after discovery of loss of ADS initiation capability.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowable out of service time "clock." For Required Action H.1, the Completion Time only begins upon discovery that the ADS cannot be automatically initiated due to inoperable channels within similar ADS trip system Functions as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 8 days has been shown to be acceptable (Ref. 5) to permit restoration of any inoperable channel to OPERABLE status if both HPCI and RCIC are OPERABLE (Required Action H.2). If either HPCI or RCIC is inoperable, the time shortens to 96 hours. If the status of HPCI or RCIC changes such that the Completion Time changes from 8 days to 96 hours, the 96 hours begins upon discovery of HPCI or RCIC inoperability. However, the total time for an inoperable channel cannot exceed 8 days. If the status of HPCI or RCIC changes such that the Completion Time changes from 96 hours to 8 days, the "time zero" for beginning the 8 day "clock" begins upon discovery of the inoperable channel. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of

(continued)

BASES

ACTIONS

H.1 and H.2 (continued)

service time, Condition I must be entered and its Required Action taken. The Required Actions do not allow placing the channel in trip since this action would not necessarily result in a safe state for the channel in all events.

I.1

With any Required Action and associated Completion Time not met, the associated feature(s) may be incapable of performing the intended function, and the supported feature(s) associated with inoperable untripped channels must be declared inoperable immediately.

SURVEILLANCE REQUIREMENTS

As noted in the beginning of the SRs, the SRs for each ECCS instrumentation Function are found in the SRs column of Table 3.3.5.1-1.

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 as follows: (a) for up to 6 hours for Functions 1.d, 2.f, 3.c, 3.d, 3.e, and 3.f; and (b) for up to 6 hours for Functions other than 1.d, 2.f, 3.c, 3.d, 3.e, and 3.f provided the associated Function (or redundant Function in the case of ADS Logics A and B) maintains ECCS initiation or loop selection capability. Loop selection capability is maintained so long as sufficient channels are OPERABLE for the Loop Select Logic to be able to identify the intact recirculation loop for injection. 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 (Ref. 5) 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 ECCS will initiate when necessary.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

Because the Ref. 5 analysis made no assumptions regarding the elapsed time between testing of consecutive channels in the same logic, it is not necessary to remove jumpers/relays blocks or reconnect lifted leads used to prevent actuation of the trip logic during testing of logic channels with instruments in series solely for the purpose of administering the AOT clocks, provided that the AOT allowance is not exceeded on a per instrument channel basis.

SR 3.3.5.1.1

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

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit. The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.5.1.2, SR 3.3.5.1.3, and SR 3.3.5.1.5

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. The CHANNEL FUNCTIONAL TEST verifies acceptable response by verifying the change of state of at least one contact on the relay which inputs into the trip logic.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1.2, SR 3.3.5.1.3, and SR 3.3.5.1.5 (continued)

The required contacts not tested during the CHANNEL FUNCTIONAL TEST are tested under the LOGIC SYSTEM FUNCTIONAL TEST. This is acceptable because operating experience shows that the contacts not tested during the CHANNEL FUNCTIONAL TEST normally pass the LOGIC SYSTEM FUNCTIONAL TEST.

The Frequency of 92 days for SR 3.3.5.1.3 is based on the reliability analyses of Reference 5.

The Frequencies of 31 days and 12 months (SR 3.3.5.1.2 and SR 3.3.5.1.5, respectively) are based upon engineering judgment and the reliability of the components.

SR 3.3.5.1.4, SR 3.3.5.1.6, SR 3.3.5.1.7, and SR 3.3.5.1.8

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies 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.

The Frequency of SR 3.3.5.1.4 is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

The Frequency of SR 3.3.5.1.6 is based upon the assumption of a 12 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

The Frequency of SR 3.3.5.1.7 is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

The Frequency of SR 3.3.5.1.8 is based upon the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.1.9

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic for a specific channel. The system functional testing performed in LCO 3.5.1, LCO 3.5.2, LCO 3.8.1, and LCO 3.8.2 overlaps this Surveillance to complete testing of the assumed safety function.

The 24 month Frequency 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.

REFERENCES

1. UFSAR, Section 5.4.
2. UFSAR, Section 6.3.
3. UFSAR, Chapter 15.
4. NEDC-32915P, "Duane Arnold Energy Center GE12 Fuel Upgrade Project", November 1999.
5. NEDC-30936-P-A, "BWR Owners' Group Technical Specification Improvement Analyses for ECCS Actuation Instrumentation, Part 2," December 1988.

B 3.3 INSTRUMENTATION

B 3.3.5.2 Reactor Core Isolation Cooling (RCIC) System Instrumentation

BASES

BACKGROUND

The purpose of the RCIC System instrumentation is to initiate actions to ensure adequate core cooling when the reactor vessel is isolated from its primary heat sink (the main condenser) and normal coolant makeup flow from the Reactor Feedwater System is unavailable, such that initiation of the low pressure Emergency Core Cooling Systems (ECCS) pumps does not occur. A more complete discussion of RCIC System operation is provided in the Bases of LCO 3.5.3, "RCIC System."

The RCIC System may be initiated by either automatic or manual means, although manual initiation requires manipulation of individual component control switches. Automatic initiation occurs for conditions of reactor vessel low low water level. The variable is monitored by four level switches that are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic arrangement. Once initiated, the RCIC logic seals in and can be reset by the operator only when the reactor vessel water level signals have cleared.

The RCIC test return valve is closed on a RCIC initiation signal to allow full system flow to the Reactor Pressure Vessel (RPV). However, this design feature is not assumed in any transient analyses. Transient analyses are performed assuming the RCIC System is in the standby readiness condition when the transient occurs.

The RCIC System also monitors the water levels in the Condensate Storage Tank (CST) since this is the preferred source of water for RCIC operation. Upon receipt of a RCIC initiation signal, the CST suction valve is automatically signaled to open (it is normally in the open position) unless the pump suction from the suppression pool valves is open. If the water level in the CST falls below a preselected level, first the suppression pool suction valves automatically open, and then the CST suction valve automatically closes. Two level switches are used to detect low water level in the CST. Either switch can cause the suppression pool suction valves to open and the CST suction valve to close (i.e., one-out-of-two once logic).

(continued)

BASES

BACKGROUND (continued)

To prevent losing suction to the pump, the suction valves are interlocked so that one suction path must be open before the other automatically closes.

The RCIC System provides makeup water to the reactor until the reactor vessel water level reaches the RPV high water level trip (two-out-of-two once logic), at which time the RCIC steam supply valve closes (the injection valve also closes due to the closure of the steam supply valve). The RCIC System restarts if vessel level again drops to the low low level initiation point.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The function of the RCIC System to provide makeup coolant to the reactor is used to respond to transient events. The RCIC System is not an Engineered Safety Feature System and no credit is taken in the accident analyses for RCIC System operation. Based on its contribution to the reduction of overall plant risk, however, the system, and therefore its instrumentation meets Criterion 4 of 10 CFR 50.36(c)(2)(ii). Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.

The OPERABILITY of the RCIC System instrumentation is dependent upon the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.5.2-1. Each Function must have a required number of OPERABLE channels with their setpoints within the specified Allowable Values. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

Allowable Values are specified for each RCIC System instrumentation Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

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 exceeds the setpoint, the associated device (e.g., on-off sensor, bi-stable trip circuit, or trip unit) changes state. Margin is provided to allow for process, calibration (i.e., M&TE) and some instrument uncertainties (e.g., drift). Allowable Values 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 individual Functions are required to be OPERABLE in MODE 1, and in MODES 2 and 3 with reactor steam dome pressure > 150 psig since this is when RCIC is required to be OPERABLE. (Refer to LCO 3.5.3 for Applicability Bases for the RCIC System.)

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

1. Reactor Vessel Water Level – Low Low

Low RPV water level indicates that normal feedwater flow is insufficient to maintain reactor vessel water level and that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, the RCIC System is initiated at Reactor Vessel Water Level - Low Low to assist in maintaining water level above the top of the active fuel.

Reactor Vessel Water Level – Low Low signals are initiated from four level switches 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.

The Reactor Vessel Water Level – Low Low Allowable Value is set high enough such that for complete loss of feedwater flow, the RCIC System flow with the High Pressure Coolant Injection assumed to fail will be sufficient to avoid

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Reactor Vessel Water Level - Low Low (continued)

initiation of low pressure ECCS at Reactor Vessel Water Level - Low Low Low.

Four channels of Reactor Vessel Water Level - Low Low Function are available and are required to be OPERABLE when RCIC is required to be OPERABLE to ensure that no single instrument failure can preclude RCIC initiation. Refer to LCO 3.5.3 for RCIC Applicability Bases.

2. Reactor Vessel Water Level - High

High RPV water level indicates that sufficient cooling water inventory exists in the reactor vessel such that there is no danger to the fuel. Therefore, the Reactor Vessel Water Level - High signal is used to close the RCIC steam supply valve to prevent overflow into the Main Steam Lines (MSLs). (The injection valve also closes due to the closure of the steam supply valve.)

Reactor Vessel Water Level - High signals for RCIC are initiated from two level switches 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.

The Reactor Vessel Water Level - High Allowable Value is high enough to preclude isolating the injection valve of the RCIC during normal operation, yet low enough to trip the RCIC System prior to water overflowing into the MSLs.

Two channels of Reactor Vessel Water Level - High Function are available and are required to be OPERABLE when RCIC is required to be OPERABLE to ensure that no single instrument failure can preclude RCIC initiation. Refer to LCO 3.5.3 for RCIC Applicability Bases.

3. Condensate Storage Tank Level - Low

Low level in the CST indicates the unavailability of an adequate supply of makeup water from this preferred source. Normally, the suction valve between the RCIC pump and the CST is open and, upon receiving a RCIC initiation signal,

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. Condensate Storage Tank Level - Low (continued)

water for RCIC injection would be taken from the CST. However, if the water level in the CST falls below a preselected level, first the suppression pool suction valves automatically open, and then the CST suction valve automatically closes. This ensures that an adequate supply of makeup water is available to the RCIC pump. To prevent losing suction to the pump, the suction valves are interlocked so that the suppression pool suction valves must be open before the CST suction valve automatically closes.

Two level switches are used to detect low water level in the CSTs. The Condensate Storage Tank Level - Low Function Allowable Value is set high enough to ensure adequate pump suction head while water is being taken from the CST.

Two channels of Condensate Storage Tank Level - Low Function are available and are required to be OPERABLE when RCIC is required to be OPERABLE to ensure that no single instrument failure can preclude RCIC swap to suppression pool source. Refer to LCO 3.5.3 for RCIC Applicability Bases.

ACTIONS

A Note has been provided to modify the ACTIONS related to RCIC System 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 RCIC System instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, the Note has been provided to allow separate Condition entry for each inoperable RCIC System instrumentation channel.

(continued)

BASES

ACTIONS (continued)

A.1

Required Action A.1 directs entry into the appropriate Condition referenced in Table 3.3.5.2-1. The applicable Condition referenced in the Table is Function dependent. Each time a channel is discovered to be inoperable, Condition A is entered for that channel and provides for transfer to the appropriate subsequent Condition.

B.1 and B.2

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in a complete loss of automatic initiation capability for the RCIC System. In this case, automatic initiation capability is lost if certain combinations of two Function 1 channels are inoperable and untripped. In this situation (loss of automatic initiation capability), the 24 hour allowance of Required Action B.2 is not appropriate, and the RCIC System must be declared inoperable within 1 hour after discovery of loss of RCIC initiation capability.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowable out of service time "clock". For Required Action B.1, the Completion Time only begins upon discovery that the RCIC System cannot be automatically initiated due to the required combination of two inoperable, untripped Reactor Vessel Water Level – Low Low channels. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the redundancy of sensors available to provide initiation signals and the fact that the RCIC System is not assumed in any accident analysis, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 1) to permit restoration of any inoperable channel to OPERABLE status.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action B.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition E must be entered and its Required Action taken.

C.1

A risk based analysis was performed and determined that an allowable out of service time of 24 hours (Ref. 1) is acceptable to permit restoration of any inoperable channel to OPERABLE status (Required Action C.1). A Required Action (similar to Required Action B.1) limiting the allowable out of service time, if a loss of automatic RCIC initiation capability exists, is not required. This Condition applies to the Reactor Vessel Water Level — High Function whose logic is arranged such that any inoperable channel will result in a loss of automatic RCIC shutdown on high level capability. As stated above, this loss of automatic RCIC shutdown capability was analyzed and determined to be acceptable.

D.1, D.2.1, and D.2.2

Required Action D.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in automatic suction transfer capability being lost. The automatic suction transfer capability is lost if two Function 3 channels are inoperable and untripped. In this situation (loss of automatic suction transfer), the 24 hour allowance of Required Actions D.2.1 and D.2.2 is not appropriate, and the RCIC System must be declared inoperable within 1 hour from discovery of loss of RCIC suction transfer capability. As noted, Required Action D.1 is only applicable if the RCIC pump suction is not aligned to the suppression pool since, if aligned, the Function is already performed.

(continued)

BASES

ACTIONS

D.1, D.2.1, and D.2.2 (continued)

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowable out of service time "clock." For Required Action D.1, the Completion Time only begins upon discovery that the RCIC System cannot be automatically aligned to the suppression pool due to two inoperable, untripped channels in the same Function. The 1 hour Completion Time from discovery of loss of suction transfer capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the redundancy of sensors available to provide suction transfer signals and the fact that the RCIC System is not assumed in any accident analysis, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 1) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action D.2.1, which performs the intended function of the channel (shifting the suction source to the suppression pool). Alternatively, Required Action D.2.2 allows the manual alignment of the RCIC suction to the suppression pool, which also performs the intended function. If Required Action D.2.1 or D.2.2 is performed, measures should be taken to ensure that the RCIC System piping remains filled with water. If it is not desired to perform Required Actions D.2.1 and D.2.2 (e.g., as in the case where shifting the suction source could drain down the RCIC discharge piping), Condition E must be entered and its Required Action taken.

E.1

With any Required Action and associated Completion Time not met, the RCIC System may be incapable of performing the intended function, and the RCIC System must be declared inoperable immediately.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

As noted in the beginning of the SRs, the SRs for each RCIC System instrumentation Function are found in the SRs column of Table 3.3.5.2-1.

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 as follows: (a) for up to 6 hours for Functions 2 and 3; and (b) for up to 6 hours for Function 1, provided the associated Function maintains trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 1) 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 RCIC will initiate when necessary. Because the Ref. 1 analysis made no assumptions regarding the elapsed time between testing of consecutive channels in the same logic, it is not necessary to remove jumpers/relay blocks or reconnect lifted leads used to prevent actuation of the trip logic during testing of logic channels with instruments in series solely for the purpose of administering the AOT clocks, provided that the AOT allowance is not exceeded on a per instrument channel basis.

SR 3.3.5.2.1

Performance of the CHANNEL CHECK once every 24 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a parameter on other similar channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or 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 3.3.5.2.1 (continued)

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

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

SR 3.3.5.2.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. The CHANNEL FUNCTIONAL TEST verifies acceptable response by verifying the change of state of at least one contact on the relay which inputs into the trip logic. The required contacts not tested during the CHANNEL FUNCTIONAL TEST are tested under the LOGIC SYSTEM FUNCTIONAL TEST. This is acceptable because operating experience shows that the contacts not tested during the CHANNEL FUNCTIONAL TEST normally pass the LOGIC SYSTEM FUNCTIONAL TEST.

The Frequency of 92 days is based on the reliability analysis of Reference 1.

SR 3.3.5.2.3 and SR 3.3.5.2.4

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies 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.

The Frequency of SR 3.3.5.2.3 is based upon the assumption of a 12 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.2.3 and SR 3.3.5.2.4 (continued)

The Frequency of SR 3.3.5.2.4 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 3.3.5.2.5

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic for a specific channel. The system functional testing performed in LCO 3.5.3 overlaps this Surveillance to provide complete testing of the safety function.

The 24 month Frequency 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.

REFERENCES

1. GENE-770-06-2, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," February 1991.

(continued)

B 3.3 INSTRUMENTATION

B 3.3.6.1 Primary Containment Isolation Instrumentation

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The primary containment isolation instrumentation automatically initiates closure of appropriate Primary Containment Isolation Valves (PCIVs). The function of the PCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs). Primary containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a DBA.

The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of primary containment and Reactor Coolant Pressure Boundary (RCPB) isolation. Most channels include equipment (e.g., on-off sensors or bi-stable trip circuits) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a primary containment isolation signal to the isolation logic. Functional diversity is provided by monitoring a wide range of independent parameters. The input parameters to the isolation logics are (a) reactor vessel water level, (b) area ambient and differential temperatures, (c) Main Steam Line (MSL) flow measurement and high radiation, (d) Standby Liquid Control (SLC) System initiation, (e) condenser vacuum, (f) main steam line pressure, (g) High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) steam line flow, (h) drywell pressure, (i) HPCI and RCIC steam line pressure, (j) HPCI and RCIC turbine exhaust diaphragm pressure, (k) Reactor Water Cleanup (RWCU) differential flow, (l) reactor steam dome pressure, (m) Offgas Vent Stack radiation, (n) Reactor Building Exhaust Shaft radiation, and (o) Refueling Floor Exhaust Duct radiation. Redundant sensor input signals from each parameter are provided for automatic initiation of isolation. The only exceptions are SLC System initiation and RWCU differential flow. In addition, manual isolation of certain logics is provided. Primary containment isolation instrumentation has inputs to

(continued)

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BACKGROUND (continued)

the trip logic of the isolation functions listed below.

1. Main Steam Line Isolation

Most MSL Isolation Functions receive inputs from four channels. The outputs from these channels are combined in a one-out-of-two taken twice logic to initiate isolation of all Main Steam Isolation Valves (MSIVs). The outputs from the same channels are arranged into two two-out-of-two logic trip systems to isolate both MSL drain valves. Both trip systems receive inputs from both Divisions of instrument channels (e.g., channels A1 and B1 or channels A2 and B2). The MSL drain line has two isolation valves with one two-out-of-two logic system associated with each valve.

The exceptions to this arrangement are the Main Steam Line Flow—High Function and Area Temperature Functions. The Main Steam Line Flow—High Function uses 16 flow channels, four for each steam line. Because of the number of instrument channels, they are arranged in a unique manner to achieve both electrical and spatial separation. One instrument channel from each steam line inputs to one of the four logic channels. Two logic channels make up each trip system and both trip systems must trip to cause an MSL isolation. Each logic channel has four inputs (one per MSL), any one of which will trip the logic channel. The logic channels are arranged in a one-out-of-two taken twice logic trip system. This allows one logic channel to be out of service and still provide isolation capability in the event of a steam line break in any MSL. This is effectively a one-out-of-eight taken twice logic arrangement to initiate isolation of the MSIVs. Similarly, the 16 flow channels are connected into two two-out-of-two logic trip systems (effectively, two one-out-of-four twice logic), with each trip system isolating one of the two MSL drain valves.

The Main Steam Tunnel Temperature—High Function receives input from 16 channels. Because of the number of instrument channels, they are arranged in a unique manner to achieve both electrical and spatial separation. The logic is arranged similar to the Main Steam Line Flow—High Function, except that the four channels from each steam line input to one of the four logic channels. The Turbine Building Area Temperature—High Function receives input from 8 channels. The inputs are arranged in a one-out-of-four taken twice logic trip system to isolate all MSIVs. Similarly, the inputs are arranged in two one-out-of-two twice logic trip

(continued)

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BACKGROUND

1. Main Steam Line Isolation (continued)

systems, with each trip system isolating one of the two MSL drain valves.

MSL Isolation Functions isolate the Group 1 valves.

2. Primary Containment Isolation

Most Primary Containment Isolation Functions receive inputs from four channels. The outputs from these channels are arranged into two two-out-of-two logic trip systems. Both trip systems receive inputs from both Divisions of instrument channels (e.g., channels A1 and B1 or channels A2 and B2). One trip system initiates isolation of all inboard primary containment isolation valves, while the other trip system initiates isolation of all outboard primary containment isolation valves. Each logic closes one of the two valves on each penetration, so that operation of either logic isolates the penetration.

An exception to this arrangement is the Offgas Vent Stack High-Radiation Function. This Function has two channels, whose outputs are arranged in two one-out-of-one logic trip systems. Each trip system isolates one valve per associated penetration, similar to the two-out-of-two logic described above.

Primary Containment Isolation Drywell Pressure—High and Reactor Vessel Water Level—Low Functions isolate the Group 2, 3, and 4 valves. Reactor Building Exhaust Shaft and Refueling Floor Exhaust Duct High-Radiation Functions isolate the Group 3 valves. Offgas Vent Stack High-Radiation Function isolates the containment purge and vent valves.

3, 4. High Pressure Coolant Injection System Isolation and Reactor Core Isolation Cooling System Isolation

Most Functions that isolate HPCI and RCIC receive input from two channels, with each channel in one trip system using a one-out-of-one logic. Each of the two trip systems in each isolation group is connected to one of the two valves on each associated penetration.

(continued)

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BACKGROUND

3, 4. High Pressure Coolant Injection System Isolation and Reactor Core Isolation Cooling System Isolation (continued)

The exceptions are the HPCI and RCIC Turbine Exhaust Diaphragm Pressure – High and Steam Supply Line Pressure – Low Functions. These Functions receive inputs from four turbine exhaust diaphragm pressure and four steam supply pressure channels for each system. The outputs from the turbine exhaust diaphragm pressure and steam supply pressure channels are each connected to two two-out-of-two trip systems. Each trip system isolates one valve per associated penetration.

HPCI and RCIC Functions isolate the Group 6A, 6B, and 9 valves.

5. Reactor Water Cleanup System Isolation

The Reactor Vessel Water Level – Low Low Isolation Function receives input from four reactor vessel water level channels. The outputs from the reactor vessel water level channels are connected into two two-out-of-two trip systems. Both trip systems receive inputs from both Divisions of instrument channels (e.g., channels A1 and B1 or channels A2 and B2). The Differential Flow – High and SLC System Initiation Functions receive input from one channel. SLC System initiation only inputs to the outboard trip system using a one-out-of-one logic. Differential Flow - High also uses a one-out-of-one logic but inputs to both trip systems via separate relays. The Area Temperature – High Function receives input from six temperature monitors, three to each trip system. The Area Ventilation Differential Temperature – High Function receives input from six differential temperature monitors, three in each trip system. These are configured so that any one input will trip the associated trip system. Each of the two trip systems is connected to either an inboard valve or an outboard valve on each RWCU penetration.

RWCU Functions isolate the Group 5 valves.

6. Shutdown Cooling System Isolation

The Reactor Vessel Water Level – Low Function receives input from four reactor vessel water level channels. The outputs from the reactor vessel water level channels are connected

(continued)

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BACKGROUND

6. Shutdown Cooling System Isolation (continued)

to two two-out-of-two trip systems. Both trip systems receive inputs from both Divisions of instrument channels (e.g., channels A1 and B1 or channels A2 and B2). The Drywell Pressure - High Function receives input from four drywell pressure channels. The outputs from the drywell pressure channels are connected to two two-out-of-two trip systems. The Reactor Steam Dome Pressure – High Function receives input from two channels, with each channel in one trip system using a one-out-of-one logic. Each of the two trip systems is connected to one of the two shutdown cooling suction valves. In addition, one trip system isolates one LPCI inboard injection valve, while the other trip system isolates the other LPCI inboard injection valve (the inboard LPCI injection valves automatically close if either a Reactor Vessel Water Level - Low signal or a Drywell Pressure - High signal is received while both shutdown cooling suction valves are not fully closed).

Shutdown Cooling System Isolation Functions isolate the Group 4 valves.

7. Containment Cooling System Permissive

In order to initiate Containment Cooling with primary containment pressure greater than the high drywell pressure ECCS initiation signal, the containment cooling permissive logic must be satisfied (upon ECCS initiation, the containment cooling penetration isolation valves receive closing signals to ensure that all Low Pressure Coolant Injection (LCPI) System flow is directed to the reactor vessel, and to ensure that containment spray is automatically isolated at a primary containment pressure greater than the pressure assumed in the analysis used to show that the primary containment design negative pressure is not exceeded during the most rapid containment cooldown transient). The containment cooling permissive logic is satisfied if all of the following conditions exist; 1) Reactor Vessel Shroud level is greater than two thirds core height, 2) a LPCI initiation signal is present, and 3) primary containment pressure is greater than the Containment Pressure - High requirement. The first two conditions support LPCI System operation and are addressed in LCO 3.3.5.1, "ECCS Instrumentation". The Containment Cooling System Permissive Function channels for the Containment Pressure - High signal receive inputs from different sensors

(continued)

BASES

BACKGROUND

7. Containment Cooling System Permissive (continued)

than those that provide input to the Drywell Pressure - High ECCS initiation signals, and are addressed here.

The Containment Pressure - High Function receives input from four primary containment pressure channels. The outputs from the primary containment pressure channels are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic. This common logic outputs into two trip systems for this function, one trip system is associated with one division of containment cooling and the other trip system is associated with the other division of containment cooling.

The Containment Cooling System Permissive Function allows the opening of the Drywell Spray, Suppression Pool Spray and Suppression Pool Cooling valves when the required conditions exist, and also results in an automatic isolation of those valves when containment pressure drops below the Containment Pressure - High setpoint.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The isolation signals generated by the primary containment isolation instrumentation are implicitly assumed in the safety analyses of References 1 and 2 to initiate closure of valves to limit offsite doses. Refer to LCO 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," Applicable Safety Analyses Bases for more detail of the safety analyses.

Primary containment isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.

The OPERABILITY of the primary containment instrumentation is dependent on the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.6.1-1. Each Function must have a required number of OPERABLE channels, with their setpoints within the specified Allowable Values, where appropriate. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Allowable Values are specified for each Primary Containment Isolation Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between 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 exceeds the setpoint, the associated device (e.g., on-off sensor or bi-stable trip circuit) changes state. Analytical Limits, where established, are the limiting values of the process parameters used in safety analysis to define the margin to unacceptable consequences. Margin is provided between the Allowable Value and the Analytical Limits to allow for process, calibration (i.e., M&TE) and some instrument uncertainties. Additional margin is provided between the Allowable Value and the trip setpoint to allow for the remaining instrument uncertainties (e.g., drift). 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.

In general, the individual Functions are required to be OPERABLE in MODES 1, 2, and 3 consistent with the Applicability for LCO 3.6.1.1, "Primary Containment." Functions that have different Applicabilities are discussed below in the individual Functions discussion.

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Main Steam Line Isolation

1.a. Reactor Vessel Water Level – Low Low Low

Low Reactor Pressure Vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of the MSIVs and other interfaces with the reactor vessel occurs to prevent offsite dose limits from being exceeded. The Reactor Vessel Water Level – Low Low Low Function is one of the many Functions assumed to be OPERABLE and capable of providing isolation signals. The Reactor Vessel Water Level – Low Low Low Function associated with isolation is assumed in the analysis of the recirculation line break (Ref. 1). The isolation of the MSLs on Reactor Vessel Water Level - Low Low Low supports actions to ensure that offsite dose limits are not exceeded for a DBA.

Reactor vessel water level signals are initiated from four level switches 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 Low Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level – Low Low Low Allowable Value is chosen to be the same as the ECCS Reactor Vessel Water Level - Low Low Low Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential Loss of Coolant Accident (LOCA) to prevent offsite doses from exceeding 10 CFR 100 limits.

This Function isolates the Group 1 valves.

1.b. Main Steam Line Pressure – Low

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure – Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 2). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.b. Main Steam Line Pressure-Low (continued)

not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four pressure switches that are each connected to an associated MSL. The pressure switches are arranged such that, even though physically separated from each other, each pressure switch is able to detect low MSL pressure. Four channels of Main Steam Line Pressure-Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure-Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 2).

This Function isolates the Group 1 valves.

1.c. Main Steam Line Flow-High

Main Steam Line Flow-High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow-High Function is directly assumed in the analysis of the Main Steam Line Break (MSLB) (Ref. 8). The isolation action, along with the scram function of the Reactor Protection System (RPS) and the mass flow rate limiting function of the MSL flow restrictors, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the 10 CFR 100 limits.

The MSL flow signals are initiated from 16 differential pressure switches that are connected to the four MSLs. The differential pressure switches are arranged such that, even though physically separated from each other, all four

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.c. Main Steam Line Flow - High (continued)

connected to one MSL would be able to detect the high flow. Four channels of Main Steam Line Flow—High Function for each unisolated MSL (two channels per trip system) are available and are required to be OPERABLE so that no single instrument failure will preclude detecting a break in any individual MSL. The Allowable Value is chosen to ensure that offsite dose limits are not exceeded due to the break. While expressed as a percentage of rated steam flow, the Allowable Value is actually displayed in terms of differential pressure across the MSL flow venturies, (i.e., psid).

This Function isolates the Group 1 valves.

1.d. Condenser Backpressure - High

The Condenser Backpressure - High Function is provided to prevent overpressurization of the main condenser in the event of a loss of the main condenser vacuum. Since the integrity of the condenser is an assumption in offsite dose calculations, the Condenser Backpressure - High Function is assumed to be OPERABLE and capable of initiating closure of the MSIVs. The closure of the MSIVs is initiated to prevent the addition of steam that would lead to additional condenser pressurization and possible rupture of the diaphragm installed to protect the turbine exhaust hood, thereby preventing a potential radiation leakage path following an accident.

Condenser vacuum pressure (backpressure) signals are derived from four pressure transmitters that sense the pressure in the condenser. Four channels of Condenser Backpressure - High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value is chosen to prevent damage to the condenser due to pressurization, thereby ensuring its integrity for offsite dose analysis. As noted (footnote (a) to Table 3.3.6.1-1), the channels are not required to be OPERABLE in MODES 2 and 3 when all Turbine Stop Valves (TSVs) are closed, since the potential for condenser overpressurization is minimized. Switches are provided to manually bypass the channels when all TSVs are closed.

This Function isolates the Group 1 valves.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.e and 1.g. Area Temperature - High

Area temperature is provided to detect a leak in the RCPB and provides diversity to the high flow instrumentation. The isolation occurs when a very small leak has occurred. If the small leak is allowed to continue without isolation, offsite dose limits may be reached. However, credit for these instruments is not taken in any transient or accident analysis in the UFSAR, since bounding analyses are performed for large breaks, such as MSLBs.

Area temperature signals are initiated from Resistance Temperature Detectors (RTDs) located in the area being monitored. Sixteen channels of Main Steam Tunnel Temperature - High Function are available and 8 channels (2 per main steam line) are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function for a break of the size for which protection is necessary. Eight channels of Turbine Building Area Temperature - High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. Each channel consists of a RTD and its contacts in the corresponding temperatures indicating switch.

The Area Temperature-High Allowable Value is set far enough above the temperature expected during operations at rated power to avoid spurious isolation, yet low enough to provide early indication of a steamline break.

These Functions isolate the Group 1 valves.

1.f. Main Steam Line Radiation - High

The Main Steam Line Radiation - High isolation signal has been removed from the MSIVs (Ref. 9); however, this isolation Function has been retained for other valves (e.g., Main Steam Line (MSL) Drains) to ensure that the assumptions utilized to determine that acceptable offsite doses resulting from a CRDA are maintained.

Main Steam Line Radiation - High signals are generated from four radiation elements and associated monitors, each of which is located near one of the MSLs in the steam tunnel.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.f. Main Steam Line Radiation - High (continued)

The Main Steam Line Radiation - High Allowable Value is chosen to be low enough so that the assumptions utilized to determine that acceptable offsite doses resulting from a CRDA are maintained.

Four Main Steam Line Radiation - High channels are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

This function isolates the MSL drains and the recirculation sample valves, and causes the mechanical vacuum pump to trip, if operating, and then the suction valves to the vacuum pump to close.

Primary Containment Isolation

2.a. Reactor Vessel Water Level – Low

Low RPV water level indicates that the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Reactor Vessel Water Level-Low supports actions to ensure that offsite dose limits of 10 CFR 100 are not exceeded. The Reactor Vessel Water Level – Low Function associated with isolation is implicitly assumed in the UFSAR analysis as these leakage paths are assumed to be isolated post LOCA.

Reactor Vessel Water Level – Low signals are initiated from level switches 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 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level – Low Allowable Value was chosen to be the same as the RPS Reactor Vessel Water Level - Low scram Allowable Value (LCO 3.3.1.1), since isolation of these valves is not critical to orderly plant shutdown.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Reactor Vessel Water Level – Low (continued)

This Function isolates the Group 2 and 3 valves.

2.b. Drywell Pressure – High

High drywell pressure can indicate a break in the RCPB inside the primary containment. The isolation of some of the primary containment isolation valves on high drywell pressure supports actions to ensure that offsite dose limits of 10 CFR 100 are not exceeded. The Drywell Pressure – High Function, associated with isolation of the primary containment, is implicitly assumed in the UFSAR accident analysis as these leakage paths are assumed to be isolated post LOCA.

High drywell pressure signals are initiated from pressure switches that sense the pressure in the drywell. Four channels of Drywell Pressure – High per Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be approximately the same as the ECCS Drywell Pressure – High Allowable Value (LCO 3.3.5.1), since this may be indicative of a LOCA inside primary containment.

This Function isolates the Group 2 and 3 valves.

2.c. Offgas Vent Stack - High Radiation

High Offgas Vent Stack radiation indicates possible gross failure of the fuel cladding. Therefore, when Offgas Vent Stack High-Radiation is detected, an isolation is initiated to limit the release of fission products should a LOCA occur while the primary containment is undergoing vent or purge operations (Ref. 10). However, this Function is not assumed in any accident or transient analysis in the UFSAR because other leakage paths (e.g., MSIVs) are more limiting.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.c. Offgas Vent Stack - High Radiation (continued)

The offgas vent stack radiation signals are initiated from radiation detectors that are located in the offgas vent stack. Two channels of Offgas Vent Stack High-Radiation Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value is low enough to ensure primary containment isolation prior to a maximum allowable 10 CFR 20 release, and is determined in accordance with the Offsite Dose Assessment Manual.

This Function is only required to isolate the containment vent and purge valves. This Function also starts the Standby Gas Treatment System.

2.d., 2.e. Reactor Building Exhaust Shaft and Refueling Floor Exhaust Duct - High Radiation

High secondary containment exhaust radiation is an indication of possible gross failure of the fuel cladding. The release may have originated from the primary containment due to a break in the RCPB. When a Reactor Building Exhaust Shaft-High Radiation or Refueling Floor Exhaust Duct-High Radiation is detected, valves whose penetrations communicate with the primary containment atmosphere and certain other valves are isolated to limit the release of fission products. Additionally, the Refueling Floor Exhaust Duct High-Radiation Function is assumed to initiate isolation of the secondary containment during a fuel handling accident (Ref. 2).

These high radiation signals are initiated from radiation detectors that are located either in the reactor building exhaust shaft or in the exhaust duct coming from the refueling floor, respectively. The signal from each detector is input to an individual monitor whose trip outputs are assigned to an isolation channel. Two channels of Reactor Building Exhaust Shaft – High Radiation Function and two channels of Refueling Floor Exhaust Duct – High Radiation Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.d., 2.e. Reactor Building Exhaust Shaft and Refueling Floor Exhaust Duct - High Radiation (continued)

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding.

These Functions isolate the Group 3 valves. These Functions also start the Standby Gas Treatment System.

High Pressure Coolant Injection and Reactor Core Isolation Cooling Systems Isolation

3.a., 4.a. HPCI and RCIC Steam Line Flow-High

Steam Line Flow-High Functions are provided to detect a break of the RCIC or HPCI steam lines and initiate closure of the steam line isolation valves of the appropriate system. If the steam is allowed to continue flowing out of the break, the reactor will depressurize and the core can uncover. Therefore, the isolations are initiated on high flow to prevent or minimize core damage and to ensure that potential fission product releases are maintained within applicable limits. The isolation action, along with the scram function of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. Specific credit for these Functions is not assumed in any UFSAR accident analyses since the bounding analysis is performed for large breaks such as recirculation and MSL breaks. However, these instruments prevent the RCIC or HPCI steam line breaks from becoming bounding.

The HPCI and RCIC Steam Line Flow-High signals are initiated from differential pressure switches (two for HPCI and two for RCIC) that are connected to the system steam lines. Two channels of both HPCI and RCIC Steam Line Flow-High Functions are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are chosen to be low enough to ensure that the trip occurs to prevent fuel damage and maintains the MSLB event as the bounding event.

These Functions isolate the Group 6A and 6B valves, as appropriate.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3.b., 4.b. HPCI and RCIC Steam Supply Line Pressure – Low

Low MSL pressure indicates that the pressure of the steam in the HPCI or RCIC turbine may be too low to continue operation of the associated system's turbine. These isolations are for equipment protection and are not assumed in any transient or accident analysis in the UFSAR. However, they also provide a diverse signal to indicate a possible system break and the HPCI Steam Supply Line Pressure - Low provides a permissive for the Drywell Pressure - High isolation of the HPCI and RCIC turbine exhaust vacuum breakers. These instruments are included in Technical Specifications (TS) because of the potential for risk due to possible failure of the instruments (i.e., the measured variable failing low or the setpoint failing high) preventing HPCI and RCIC initiations (Ref. 3). Therefore, they meet Criterion 4 of 10 CFR 50.36(c)(2)(ii).

The HPCI and RCIC Steam Supply Line Pressure – Low signals are initiated from pressure switches (four for HPCI and four for RCIC) that are connected to the system steam line. Four channels of both HPCI and RCIC Steam Supply Line Pressure – Low Functions are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are selected to be high enough to prevent damage to the system's turbine. Also, the upper Allowable Value for HPCI Steam Supply Line Pressure-Low is selected to ensure HPCI is available in the pressure range assumed in the LOCA analysis.

These Functions isolate the Group 6A and 6B valves, as appropriate.

3.c., 4.c. HPCI and RCIC Turbine Exhaust Diaphragm Pressure – High

High turbine exhaust diaphragm pressure indicates that the pressure may be too high to continue operation of the associated system's turbine. That is, the inner diaphragm has developed a leak and pressure is reaching a point where the outer diaphragm may be challenged. These isolations are for equipment and personnel safety protection and are not assumed in any transient or accident analysis in the UFSAR.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.c., 4.c. HPCI and RCIC Turbine Exhaust Diaphragm Pressure - High (continued)

These instruments are included in the TS because of the potential for risk due to possible failure of the instruments preventing HPCI and RCIC initiations (Ref. 3). In addition, the failure of these instruments to detect a degraded inner diaphragm, increases the risk of an unwarranted release of steam to the equipment space. Therefore, they meet Criterion 4 of 10 CFR 50.36(c)(2)(ii).

The HPCI and RCIC Turbine Exhaust Diaphragm Pressure—High signals are initiated from pressure switches (four for HPCI and four for RCIC) that are connected to the area between the rupture diaphragms on each system's turbine exhaust line. Four channels of both HPCI and RCIC Turbine Exhaust Diaphragm Pressure—High Functions are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are low enough to detect a degraded inner diaphragm to prevent extended operation with only the outer diaphragm acting as a system pressure boundary.

These Functions isolate the Group 6A and 6B valves, as appropriate.

3.d., 4.d. Drywell Pressure—High

High drywell pressure can indicate a break in the RCPB. The isolation of the HPCI and RCIC turbine exhaust vacuum breakers is provided to prevent communication with the suppression pool atmosphere when high primary containment pressure exists. A potential leakage path exists via the turbine exhaust line. The isolation is delayed until the system becomes unavailable for injection (i.e., low steam line pressure). The isolation of the HPCI and RCIC turbine exhaust vacuum breakers by Drywell Pressure—High is accomplished by common Motor Operated Valves (MOVs) in series, and is indirectly assumed in the UFSAR accident analysis because the turbine exhaust leakage path is not assumed to contribute to offsite doses. Following system shutdown after a LOCA, a closure of the MOVs results in suppression pool pressure forcing water to the turbine exhaust line check valves, thereby precluding gaseous outleakage through this path. The isolation logic ensures

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.d., 4.d. Drywell Pressure-High (continued)

the availability of the vacuum breaker feature following HPCI or RCIC operation while at the same time providing the desired containment isolation capabiity following a LCOA (Ref. 1).

High drywell pressure signals are initiated from pressure switches that sense the pressure in the drywell. Two channels of Drywell Pressure-High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be the same as the ECCS Drywell Pressure-High Allowable Value (LCO 3.3.5.1), since this is indicative of a LOCA inside primary containment.

This Function isolates the Group 9 valves.

3.e., 3.g., 3.h., 3.i., 4.e., 4.g., 4.h., 4.i. Area and Differential Temperature-High

Area and differential temperatures are provided to detect a leak from the associated system steam piping. The isolation occurs when a very small leak has occurred and is diverse to the high flow instrumentation. If the small leak is allowed to continue without isolation, offsite dose limits may be reached. These Functions are not assumed in any UFSAR transient or accident analysis, since bounding analyses are performed for large breaks such as recirculation or MSL breaks.

Area and Ambient Temperature-High signals are initiated from temperature elements that are located in two areas for HPCI and in two areas for RCIC to protect each system. Two instruments monitor each area. Two channels for each HPCI and RCIC Area, and Suppression Pool Area Ambient Temperature-High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

Eight temperature elements provide input to the HPCI and Suppression Pool Area Ventilation Differential Temperature-High Functions and eight temperature elements provide input to the RCIC and Suppression Pool Area Ventilation Differential Temperature - High Functions. The output of these temperature elements is used to determine the differential temperature. Each channel consists of a

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.e., 3.g., 3.h., 3.i., 4.e., 4.g., 4.h., 4.i. Area and Differential Temperature - High (continued)

differential temperature instrument that receives inputs from temperature elements that are located in the inlet and outlet of the area cooling system for a total of four available channels (two for RCIC and two for HPCI).

The Allowable Values are set low enough to detect a leak equivalent to 5 gpm. These Functions isolate the Group 6A and 6B valves, as appropriate.

3.f., and 4.f. HPCI and RCIC Leak Detection Time Delay

The HPCI Leak Detection and RCIC Leak Detection Time Delay Functions are provided to allow all the other systems that may be leaking into the pool area (as indicated by the high temperature) to be isolated before HPCI and/or RCIC are automatically isolated. This ensures maximum HPCI and RCIC System availability by preventing isolations due to leaks in other systems. These Functions are not assumed in any UFSAR transient or accident analysis.

There are four time delay relays (two for HPCI and two for RCIC). Two channels each for both HPCI and RCIC Leakage Detection Time Delay Functions are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

No Allowable Values are specified in Table 3.3.6.1-1 for these Functions, as Analytical Limits are not established for this variable, only Nominal Trip Setpoints (NTSPs), which are controlled by plant procedures. The NTSPs are based on ensuring that the HPCI and RCIC systems are isolated in a sufficiently short time after a steam leak is detected so as to ensure that any potential release from such a leak remains bounded by the larger releases of the Design Basis Accidents (DBAs). In addition, the HPCI and RCIC time delays are staggered to allow the operator to determine which of these two systems is the source of leakage, allowing it to isolate, while maintaining the other system for core cooling. The time delay for the HPCI isolation is set shorter than that for RCIC as it has the larger steam line and hence the larger reactor vessel inventory loss if broken. Margin has been added to the NTSPs to account for instrument drift to specify acceptable as-found tolerances. These Functions isolate the Group 6A and 6B valves, as appropriate.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3.j, and 4.j. HPCI and RCIC Manual Initiation

The Manual Initiation push button channels introduce signals into the HPCI and RCIC systems' isolation logics that are redundant to the automatic protective instrumentation and provide manual isolation capability if a system initiation signals is present. There is no specific UFSAR safety analysis that takes credit for thes Functions. They are retained for overall redundancy and diversity of the isolation function.

There is one push button for each system (HPCI and RCIC), which actuates the respective outboard trip system. There is no Allowable Value for these Functions, since the channels are mechanically actuated based solely on the position of the push buttons.

One channel of both HPCI and RCIC Manual Initiation Function is available and is required to be OPERABLE in MODES 1, 2, and 3 since these are the MODES in which the HPCI and RCIC systems' Isolation automatic Functions are required to be OPERABLE.

Reactor Water Cleanup System Isolation

5.a Differential Flow-High

The high differential flow signal is provided to detect a break in the RWCU System. This will detect leaks in the RWCU System when area or differential temperature would not provide detection (i.e., a cold leg break). Should the reactor coolant continue to flow out of the break, offsite dose limits may be exceeded. Therefore, isolation of the RWCU System is initiated when high differential flow is sensed to prevent exceeding offsite doses. A time delay is provided to prevent spurious trips during most RWCU operational transients. This Function is not assumed in any UFSAR transient or accident analysis, since bounding analyses are performed for large breaks such as MSLBs.

The high differential flow signals are initiated from flow transmitters that are connected to the inlet (from the reactor vessel) and outlets (to the condenser (or radwaste) and to feedwater) of the RWCU System. The outputs of the transmitters are compared (in a common summer) and the

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Reactor Water Cleanup System Isolation

5.a. Differential Flow-High (continued)

resulting output is sent to two flow switches that trip when the sensed differential flow exceeds a predetermined value. If the difference between the inlet and outlets flow is too large, each flow switch generates an isolation signal. Two channels of Differential Flow-High Function are available and are required to be OPERABLE to ensure that no single instrument failure downstream of the common summer can preclude the isolation function.

The Differential Flow-High Allowable Value ensures that a "critical crack" in the RWCU piping is detected.

This Function isolates the Group 5 valves.

5.b., 5.c., and 5.f Area, Area Near Tip Room Ambient, and Area Ventilation Differential Temperature-High

RWCU area, area near TIP Room ambient, and area ventilation differential temperatures are provided to detect a leak from the RWCU System. The isolation occurs even when very small leaks have occurred and is diverse to the high differential flow instrumentation for the hot portions of the RWCU System. If the small leak continues without isolation, offsite dose limits may be reached. In addition, the Area Near TIP Room Ambient Temperature - High Function isolates the RWCU System to ensure that the assumed environmental conditions are maintained in the first floor of the Reactor Building for equipment qualification purposes. Credit for these instruments is not taken in any transient or accident analysis in the UFSAR, since bounding analyses are performed for large breaks such as recirculation or MSL breaks.

Area, area near TIP Room ambient, and area ventilation differential temperature signals are initiated from temperature elements that are located in the area that is being monitored. Six temperature elements provide input to the Area Temperature-High Function (four in the RWCU heat exchanger area and two in the RWCU pump area). However, only two channels are required to be OPERABLE (one in each trip system) to ensure that no single instrument failure can preclude the isolation function since the Area Differential

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY

5.b, 5.c, and f.5 Area, Area Near Tip Room Ambient, and Area Ventilation Differential Temperature-High (continued)

Temperature High Function is able to detect breaks in the same areas as the Area Temperature - High Functions. As noted in footnote (d) to Table 3.3.6.1-1, in order to maintain coverage in both areas, each area (i.e., the heat exchanger area and the pump area) must have either an OPERABLE Area Temperature - High channel or an OPERABLE Area Ventilation Differential Temperature - High channel in each trip system.

Twelve temperature elements provide input to the Area Ventilation Differential Temperature – High Function. The output of these temperature elements is used to determine the differential temperature. Each channel consists of a differential temperature instrument that receives inputs from temperature elements that are located in the inlet and outlet of the area cooling system and for a total of six available channels (three in the RWCU heat exchanger area and three in the RWCU pump area). However, only two channels are required to be OPERABLE (one in each trip system) to ensure that no single instrument failure can preclude the isolation function since the Area Temperature - High Function is able to detect breaks in the same area as the Area Ventilation Differential Temperature - High Function. As noted in footnote (d) to Table 3.3.6.1-1, in order to maintain coverage in both areas, each area (i.e., the heat exchanger area and the pump area) must have either an OPERABLE Area Temperature - High channel or an OPERABLE Area Ventilation Differential Temperature - High channel in each trip system.

Four temperature elements provide input to the Area Near TIP Room Ambient Temperature - High Function. However, only two channels are required to be OPERABLE (one in each trip system) to ensure that no single instrument failure can preclude the isolation function.

The Area, Area Near TIP Room Ambient, and Area Ventilation Differential Temperature–High Allowable Values are set low enough to detect a leak equivalent to 5 gpm.

These Functions isolate the Group 5 valves.

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY (continued)

5.d. SLC System Initiation

The isolation of the RWCU System is required when the SLC System has been initiated to prevent dilution and removal of the boron solution by the RWCU System (Ref. 4). The SLC System initiation signal is initiated from the SLC System initiation switch. There is no Allowable Value associated with this Function since the channels are mechanically actuated based solely on the position of the SLC System initiation switch. One channel of the SLC System Initiation Function is available and is required to be OPERABLE only in MODES 1 and 2, since these are the only MODES where the reactor can be critical, and these MODES are consistent with the Applicability for the SLC System (LCO 3.1.7). As noted (footnote (e) to Table 3.3.6.1-1), this Function is only required to close the valves associated with one trip system, since the signal only provides input into one of the two trip systems. This results in isolating two of the three RWCU PCIVs.

5.e. Reactor Vessel Water Level - Low Low

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some interfaces with the reactor vessel occurs to isolate the potential sources of a break. The isolation of the RWCU System on Reactor Vessel Water Level-Low Low supports actions to ensure that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. The Reactor Vessel Water Level - Low Low Function associated with RWCU isolation is not directly assumed in the UFSAR safety analyses because the RWCU System line break is bounded by breaks of larger systems (recirculation and MSL breaks are more limiting).

Reactor Vessel Water Level - Low Low signals are initiated from four level switches 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 Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY (continued)

5.e. Reactor Vessel Water Level - Low Low (continued)

The Reactor Vessel Water Level - Low Low Allowable Value was chosen to be the same as the ECCS Reactor Vessel Water Level - Low Low Allowable Value (LCO 3.3.5.1), since the capability to cool the fuel may be threatened.

This Function isolates the Group 5 valves.

Shutdown Cooling System Isolation

6.a. Reactor Steam Dome Pressure - High

The Reactor Steam Dome Pressure - High Function is provided to isolate the shutdown cooling portion of the Residual Heat Removal (RHR) System (i.e., the shutdown cooling suction valves). This interlock is provided only for equipment protection to prevent an intersystem LOCA scenario (i.e., a break of the low pressure RHR suction piping caused by exposure to relatively high pressure RPV fluid), and credit for the interlock is not assumed in the accident or transient analysis in the UFSAR.

The Reactor Steam Dome Pressure - High signals are initiated from two pressure switches that are connected to different taps on the suction piping of the "B" Recirculation System. Two channels of Reactor Steam Dome Pressure - High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. The Function is only required to be OPERABLE in MODES 1, 2, and 3, since these are the only MODES in which the reactor can be pressurized; thus, over pressure protection is needed. The Allowable Value was chosen to be low enough to protect the RHR System piping from overpressurization (even with a time delay present), yet high enough to preclude spurious isolations of shutdown cooling during system startup and operation and to provide sufficient overlap with the low pressure isolations of the HPCI and RCIC turbines to allow the transition to shutdown cooling during plant shutdowns.

This Function isolates the Group 4 valves.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

6.b. Reactor Vessel Water Level – Low

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some reactor vessel interfaces occurs to begin isolating the potential sources of a break. The Reactor Vessel Water Level – Low Function associated with RHR Shutdown Cooling System isolation is not directly assumed in safety analyses because a break of the RHR Shutdown Cooling System is bounded by breaks of the Recirculation Suction and MSL. The RHR Shutdown Cooling System isolation on Reactor Vessel Water Level-Low supports actions to ensure that the RPV water level does not drop below the top of the active fuel during a vessel draindown event caused by a leak (e.g., pipe break or inadvertent valve opening) in the RHR Shutdown Cooling System when the system is in operation (i.e., the shutdown cooling suction valves are automatically isolated, and if both of the RHR shutdown cooling suction valves are not fully closed and reactor steam dome pressure is less than 135 psig (nominal), then the two inboard LPCI injection valves are also automatically isolated if a low reactor vessel water level signal is received).

Reactor Vessel Water Level – Low signals are initiated from four level indicating switches 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 (two channels per trip system) of the Reactor Vessel Water Level – Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. As noted (footnote (f) to Table 3.3.6.1-1), only two channels of the Reactor Vessel Water Level – Low Function are required to be OPERABLE in MODES 4 and 5 (and must be capable of providing input to initiate the isolation of the same division of isolation valves, i.e., both the A1 and B1 channels or both the A2 and B2 channels are required to be OPERABLE), provided the RHR Shutdown Cooling System integrity is maintained. System integrity is maintained provided the piping is intact and no operations with the potential for draining the reactor vessel through the system are being performed.

The Reactor Vessel Water Level – Low Allowable Value was chosen to be the same as the RPS Reactor Vessel Water

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABIITY

6.b. Reactor Vessel Water Level - Low (continued)

Level - Low Allowable Value (LCO 3.3.1.1), since the capability to cool the fuel may be threatened.

The Reactor Vessel Water Level - Low Function is only required to be OPERABLE in MODES 3, 4, and 5 to prevent this potential flow path from lowering the reactor vessel level to the top of the fuel. In MODES 1 and 2, another isolation (i.e., Reactor Steam Dome Pressure - High) and administrative controls ensure that this flow path remains isolated to prevent unexpected loss of inventory via this flow path.

This Function isolates the Group 4 valves.

6.c. Drywell Pressure - High

High drywell pressure indicates that the RHR Shutdown Cooling System piping downstream of the inboard isolation valve located in the drywell may have experienced a break. In order to prevent the level in the RPV from dropping below the top of active fuel if this were to occur, this Function will cause the RHR Shutdown Cooling System to isolate if the system is in use (i.e., the shutdown cooling suction valves are automatically isolated, and if both of the RHR shutdown cooling suction valves are not fully closed and reactor steam dome pressure is less than 135 psig (nominal), then the two inboard LPCI injection valves are also automatically isolated if a high drywell pressure signal is received). The Drywell Pressure - High Function associated with the RHR Shutdown Cooling System isolation is not directly assumed in safety analyses because a break of the RHR Shutdown Cooling system is bounded by breaks of the Recirculation System and MSL.

Drywell Pressure - High signals are initiated from four pressure switches that are connected to the primary containment via four different penetrations. Four channels (two channels per trip system) of the Drywell Pressure - High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

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BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY

6.c. Drywell Pressure - High (continued)

The Drywell Pressure - High Function is only required to be OPERABLE in Modes 1, 2, and 3 since these are the only MODES in which a LOCA could occur to cause the drywell to pressurize.

The Allowable Value was selected to be approximately the same as the ECCS Drywell Pressure-High Allowable Value (LCO 3.3.5.1), since this may be indicative of a LOCA inside primary containment.

This Function isolates the Group 4 valves.

Containment Cooling System Isolation

7.a Containment Pressure - High

High containment pressure indicates that a primary system break inside the containment has occurred. In order to maintain long term primary containment integrity under these conditions, the primary containment must be cooled. However, before Residual Heat Removal (RHR) System flow can be diverted from the LPCI flow path to the containment cooling flow path (e.g., suppression pool cooling), adequate core cooling must be assured. This assurance is provided by the Containment Cooling System Permissive Function, which receives inputs from LPCI initiation signals, reactor vessel shroud level signals and primary containment pressure signals. The LPCI initiation inputs and reactor vessel shroud level inputs to the Containment Cooling System Permissive logic can be overridden, if desired. Therefore, suppression pool cooling is still capable of performing the post accident containment cooling function if either of these two functions are inoperable. However, in order to prevent challenging the containment negative design pressure limit when spraying the containment, the Containment Pressure - High Function cannot be overridden (suppression pool spray valves are located on the same primary containment penetrations as the suppression pool cooling valves). Therefore, if certain combinations of Function 7.a channels are inoperable and untripped (such that the one-out-of-two taken twice logic is inhibited), then the logic will not allow the suppression pool cooling valves to be opened under post accident conditions, and the

(continued)

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APPLICABLE SAFETY ANALYSES LCO, and APPLICABIITY

7.a Containment Pressure - High (continued)

Suppression Pool Cooling System would be unavailable to perform the post accident containment cooling function. Consequently, this Function must be OPERABLE to support long term primary containment integrity by allowing containment cooling under post-LOCA conditions.

Additionally, if certain combinations of Function 7.a channels are in the tripped condition (such that the one-out-of-two taken twice logic is fulfilled), then the logic is not capable of automatically isolating the containment sprays to prevent containment pressure from dropping below the negative design pressure limit when the containment is being sprayed. Therefore, in order to protect primary containment integrity if the correct combination of Function 7.a channels are in the tripped condition such that automatic isolation capability is lost, the containment sprays should be inhibited.

Containment Pressure - High signals are initiated from four pressure switches that are connected to the primary containment via four different penetrations. Four channels of the Containment Pressure - High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the containment cooling function under post accident conditions.

The Containment Pressure - High Function is only required to be OPERABLE in MODES 1, 2 and 3, since these are the only MODES in which a LOCA could cause pressurization of the primary containment. In MODES 4 and 5, the probability and consequences of such events are reduced due to the pressure and temperature limitations in these MODES.

The Allowable Value was selected to be high enough to prevent the primary containment from exceeding the negative design pressure limit when spraying the containment.

This Function contributes to maintaining primary containment integrity under post accident conditions by: 1) allowing the use of the Suppression Pool Cooling System to fulfill the containment cooling function, and 2) preventing the containment negative design pressure limit from being exceeded by automatically isolating the sprays as pressure drops to the Allowable Value when spraying the containment.

(continued)

BASES (continued)

ACTIONS

A Note has been provided to modify the ACTIONS related to primary containment isolation 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 primary containment isolation instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, the Note has been provided to allow separate Condition entry for each inoperable primary containment isolation instrumentation channel.

A.1 and A.2

Because of the diversity of sensors available to provide isolation signals and the redundancy of the isolation design, an allowable out of service time of 12 hours for Functions 2.a, 2.b, 6.b, and 6.c and 24 hours for Functions other than Functions 2.a, 2.b, 6.b, and 6.c has been shown to be acceptable (Refs. 5 and 6) to permit restoration of any inoperable channel to OPERABLE status. This out of service time is only acceptable provided the associated Function is still maintaining isolation capability (refer to Required Action B.1 Bases). If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action A.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue with no further restrictions. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an isolation), Condition C must be entered and its Required Action taken.

An additional Required Action is provided for Function 7.a. Required Action A.2 requires that within 24 hours the containment sprays be inhibited. If a Function 7.a channel is placed in trip per Required Action A.1, containment

(continued)

BASES (continued)

ACTIONS

A.1. and A.2 (continued)

integrity could be threatened due to exceeding the containment negative design pressure limit. Inhibiting the containment sprays from spraying both the drywell and the suppression chamber removes this threat. One acceptable method of inhibiting containment spray operation is to verify that either the inboard or outboard containment spray Motor Operated Valve (MOV) on each applicable penetration is closed, and then to open the circuit breakers that supply electrical power to those MOVs (Note: if the inboard suppression chamber MOVs are closed, suppression pool cooling is not affected). Refer to LCO 3.6.2.4 for the Required Actions and associated Completion Times for inoperable RHR Suppression Pool Spray Subsystem(s). The 24 hour completion time is consistent with the time allowed to place the channel in trip.

B.1

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in redundant automatic isolation capability being lost for the associated penetration flow path(s). The MSL Isolation Functions are considered to be maintaining isolation capability when sufficient channels are OPERABLE or in trip, such that both trip systems will generate a trip signal from the given Function on a valid signal. The other isolation functions are considered to be maintaining isolation capability when sufficient channels are OPERABLE or in trip, such that one trip system will generate a trip signal from the given Function on a valid signal. This ensures that one of the two PCIVs in the associated penetration flow path can receive an isolation signal from the given Function. Note that Condition B can not exist for penetrations that are

(continued)

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ACTIONS

B.1 (continued)

isolated by a single automatic isolation valve in series with a check valve (e.g., recirculation pump mini-purge penetrations and drywell pneumatics nitrogen supply penetration) since isolation capability is considered to be maintained by the check valve regardless of the ability of the automatic isolation functions to affect isolation of the penetration. For one-out-of-two taken twice logics, this would require both trip systems to have one channel OPERABLE or in trip. For the MSL High Flow Function, this would require both trip systems to have one channel, associated with each MSL, OPERABLE or in trip. For Functions that consist of channels that monitor several locations within a given area (e.g., different locations for monitoring MSL temperatures in the turbine building), this would require both trip systems to have one channel per location OPERABLE or in trip. For the Main Steam Line Tunnel Temperature - High Function, two channels per steam line, either OPERABLE or in trip, are required in either trip system. With this degree of coverage available in the steam tunnel, the logic is adequate to detect and isolate the MSIVs for a break of the size for which protection is necessary. For two-out-of-two once logics, this would require the trip system logic associated with either the inboard components or the outboard components to have two channels, each OPERABLE or in trip. For one-out-of-one once logics this would require the trip system logic associated with either the inboard components or the outboard components to have one channel OPERABLE or in trip. For the RWCU Area High Temperature and Area Ventilation High Differential Temperature Functions, each Function consists of channels that monitor several different locations. Therefore, this would require one channel per location (i.e., either one OPERABLE RWCU Area High Temperature channel or one OPERABLE Area Ventilation High Differential Temperature channel per location) to be OPERABLE or in trip (the channels are not required to be in the same trip system). The Condition does not include the Manual Initiation Functions, since they are not assumed in any accident or transient analysis. Thus, a total loss of manual initiation capability for 24 hours (as allowed by Required Action A.1) is allowed. Specific guidance for when Condition B exists is contained in plant procedures.

(continued)

BASES

ACTIONS

B.1 (continued)

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

C.1

Required Action C.1 directs entry into the appropriate Condition referenced in Table 3.3.6.1-1. The applicable Condition specified in Table 3.3.6.1-1 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 or B and the associated Completion Time has expired, Condition C will be entered for that channel and provides for transfer to the appropriate subsequent Condition.

D.1, D.2.1, and D.2.2

If the channel is not restored to OPERABLE status or 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 placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours (Required Actions D.2.1 and D.2.2). Alternately, the associated MSLs may be isolated (Required Action D.1), and, if allowed (i.e., plant safety analysis allows operation with an MSL isolated), operation with that MSL isolated may continue. Isolating the affected MSL accomplishes the safety function of the inoperable channel. The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must

(continued)

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ACTIONS

E.1 (continued)

be placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 2 within 8 hours.

The allowed Completion Time of 8 hours is reasonable, based on operating experience to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

F.1

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, plant operations may continue if the affected penetration flow path(s) is isolated. Isolating the affected penetration flow path(s) accomplishes the safety function of the inoperable channels.

For the RWCU Differential Flow—High Function, if the flow element/transmitter monitoring RWCU flow to radwaste and condensate is the only portion of the channel inoperable, then the affected penetration flow path(s) may be considered isolated by isolating the RWCU return to radwaste and condensate.

Alternately, if it is not desired to isolate the affected penetration flow path(s), Condition H must be entered and its Required Actions taken.

The 1 hour Completion Time is acceptable because it minimizes risk while allowing sufficient time for plant operations personnel to isolate the affected penetration flow path(s).

G.1

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, plant operations may continue if the affected penetration flow path(s) is isolated. Isolating the affected penetration flow path(s) accomplishes the safety function of the inoperable channels. The 24 hour Completion Time is acceptable due to the fact that these Functions (Manual Initiation) are not assumed in

(continued)

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G.1 (continued)

any accident or transient analysis in the UFSAR. Alternately, if it is not desired to isolate the affected penetration flow path(s), Condition H must be entered and its Required Actions taken.

H.1 and H.2

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, or any Required Action of Condition F or G is not met and the associated Completion Time has expired, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

I.1 and I.2

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the SLC System is declared inoperable or the RWCU System is isolated. Since this Function is required to ensure that the SLC System performs its intended function, sufficient remedial measures are provided by declaring the SLC System inoperable or isolating the RWCU System.

The 1 hour Completion Time is acceptable because it minimizes risk while allowing sufficient time for personnel to isolate the RWCU System.

J.1 and J.2

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the associated penetration flow path should be closed. However, if the shutdown cooling function is needed to provide core cooling, these Required Actions allow the penetration flow path to

(continued)

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J.1 and J.2 (continued)

remain unisolated provided action is immediately initiated to restore the channel to OPERABLE status or to isolate the RHR Shutdown Cooling System (i.e., provide alternate decay heat removal capabilities so the penetration flow path can be isolated). Actions must continue until the channel is restored to OPERABLE status or the RHR Shutdown Cooling System is isolated.

K.1 and K.2

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the containment cooling system permissive logic may not allow the Suppression Pool Cooling System to be placed in service under post accident conditions (i.e., with containment pressure greater than 2 psig). Therefore, the affected Suppression Pool Cooling subsystem(s) must be declared inoperable immediately per Required Action K.1. Refer to LCO 3.6.2.3 for the Required Actions and associated Completion Times for inoperable RHR Suppression Pool Cooling subsystem(s). As noted earlier, due to the common values between the two functions, this Action also makes the associated loop of Suppression Pool Spray inoperable and LCO 3.6.2.4 must also be entered. Alternatively, if the channel is placed in trip but the containment sprays are not inhibited, containment integrity could be threatened due to exceeding the negative design pressure limit. Therefore, the primary containment must be declared inoperable immediately per Required Action K.2.

(continued)

BASES

ACTIONS (continued)

L.1 and L.2

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, either the primary containment vent and purge penetration flow paths must be isolated or administrative control of the primary containment vent and purge valves using continuous monitoring of alternate instrumentation must be established. These actions are required because the ability of the Function 2.c isolation signals to limit releases to less than 10 CFR 20 limits (if a LOCA were to occur during primary containment venting operations) is threatened. When Function 2.c is inoperable, the activity for which the isolation was intended must be terminated or administrative controls must be implemented. Acceptable administrative controls are implemented and venting or purging of primary containment may proceed, provided the following conditions are met: 1) an operator is stationed at the valve controls, and 2) that operator is instructed to terminate venting or purging when procedures direct valve closure.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each Primary Containment Isolation instrumentation Function are found in the SRs column of Table 3.3.6.1-1.

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 for Functions other than 3.j, 4.j, and 5.a may be delayed for up to 6 hours provided the associated Function maintains isolation capability and for up to 6 hours for Functions 3.j and 4.j, and 5.a. For Functions 1.c, 1.e, and 1.g, the Allowed Outage Time (AOT) is applied at the instrument channel level, since the associated trip function and isolation capability are maintained via the companion logic channel. This is consistent with the "normal" trip arrangements with one instrument channel feeding each trip logic. Thus, a six hour AOT is applied to each instrument channel undergoing required testing. Upon completion of the Surveillance, or expiration of the applicable 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

This Note is based on the reliability analysis (Refs. 5 and 6) 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 PCIVs will isolate the penetration flow path(s) when necessary. Because the Ref. 5 and 6 analyses made no assumptions regarding the elapsed time between testing of consecutive channels in the same logic, it is not necessary to remove jumpers/relay blocks or reconnect lifted leads used to prevent actuation of the trip logic during testing of logic channels with instruments in series solely for the purpose of administering the AOT clocks, provided that the AOT allowance is not exceeded on a per instrument channel basis.

SR 3.3.6.1.1 and SR 3.3.6.1.2

Performance of the CHANNEL CHECK once every 12 hours or once every 24 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

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

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.1.3, SR 3.3.6.1.4 and SR 3.3.6.1.10

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. The CHANNEL FUNCTIONAL TEST verifies acceptable response by verifying the change of state of at least one contact on the relay which inputs into the trip logic. The required contacts not tested during the CHANNEL FUNCTIONAL TEST are tested under the LOGIC SYSTEM FUNCTIONAL TEST. This is acceptable because operating experience shows that the contacts not tested during the CHANNEL FUNCTIONAL TEST normally pass the LOGIC SYSTEM FUNCTIONAL TEST.

The 92 day Frequency of SR 3.3.6.1.4 is based on the reliability analyses described in References 6 and 7. The 31 day Frequency of SR 3.3.6.1.3 and the 24 month Frequency of SR 3.3.6.1.10 is based on engineering judgment and the reliability of the components.

SR 3.3.6.1.5, SR 3.3.6.1.6, SR 3.3.6.1.7 and SR 3.3.6.1.8

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies 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.

The Frequency of SR 3.3.6.1.5 is based on the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

The Frequency of SR 3.3.6.1.6 is based on the assumption of a 184 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

The Frequency of SR 3.3.6.1.7 is based on the assumption of a 12 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

The Frequency of SR 3.3.6.1.8 is based on the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.1.9

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required isolation logic for a specific channel. The system functional testing performed on PCIVs in LCO 3.6.1.3 overlaps this Surveillance to provide complete testing of the assumed safety function. The 24 month Frequency 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 these components usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. UFSAR, Section 6.3.
2. UFSAR, Chapter 15.
3. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment," November 1987.
4. UFSAR, Section 9.3.4.2.
5. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.
6. NEDC-30851P-A Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.
7. UFSAR, Section 7.3.
8. UFSAR, Section 15.6.5.
9. Amendment 182 to Facility Operating License No. DPR-49 and NEDO-31400A, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor," October 1992.
10. NUREG-0737, Section II.E.4.2.(7).

B 3.3 INSTRUMENTATION

B 3.3.6.2 Secondary Containment Isolation Instrumentation

BASES

BACKGROUND

The secondary containment isolation instrumentation automatically initiates closure of appropriate Secondary Containment Isolation Valves and/or Dampers (SCIV/Ds) and starts the Standby Gas Treatment (SBGT) System. The function of these systems, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) (Ref. 1). Secondary containment isolation and establishment of vacuum with the SBGT System ensures that fission products that leak from primary containment following a DBA, or are released outside primary containment, or are released during certain operations when primary containment is not required to be OPERABLE are maintained within applicable limits.

The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of secondary containment isolation. Most channels include equipment (e.g., on-off sensors or bi-stable trip circuits) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a secondary containment isolation signal to the isolation logic. Functional diversity is provided by monitoring a wide range of independent parameters. The input parameters to the isolation logic are (1) reactor vessel water level, (2) drywell pressure, (3) reactor building exhaust shaft high radiation, and (4) refueling floor exhaust duct high radiation. In addition, an offgas ventilation stack high radiation signal will cause the secondary containment to isolate and the SBGT System to start. However, this signal is not relied upon to isolate secondary containment or start SBGT for the mitigation of any transients or accidents. Therefore, this Function is not included in this LCO. Redundant sensor input signals from each parameter are provided for initiation of isolation.

(continued)

BASES

BACKGROUND (continued)	The outputs of the logic channels are arranged into two two-out-of-two trip system logics (for the reactor vessel water level and drywell pressure functions) or into two one-out-of-one trip system logics (for the reactor building exhaust shaft and refueling floor exhaust duct functions). One trip system logic initiates isolation of one automatic isolation valve (damper) and starts one SBGT subsystem while the other trip system logic initiates isolation of the other automatic isolation valve in the penetration and starts the other SBGT subsystem. Each trip system logic closes one of the two valves/dampers on each penetration and starts one SBGT subsystem, so that operation of either logic isolates the secondary containment and provides for the necessary filtration of fission products.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The isolation signals generated by the secondary containment isolation instrumentation are implicitly assumed in the safety analyses of References 1 and 2 to initiate closure of valves/dampers and start the SBGT System to limit offsite doses.

Refer to LCO 3.6.4.2, "Secondary Containment Isolation Valves/Dampers (SCIV/Ds)," and LCO 3.6.4.3, "Standby Gas Treatment (SBGT) System," Applicable Safety Analyses Bases for more detail of the safety analyses.

The secondary containment isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.

The OPERABILITY of the secondary containment isolation instrumentation is dependent on the OPERABILITY of the individual instrumentation channel Functions. Each Function must have the required number of OPERABLE channels with their setpoints set within the specified Allowable Values, as shown in Table 3.3.6.2-1. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

Allowable Values are specified for each Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

to ensure that the setpoints do not exceed the Allowable Value between 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 exceeds the setpoint, the associated device (e.g., on-off sensor or bi-stable trip circuit) changes state. Analytical Limits, where established, are the limiting values of the process parameters used in safety analysis to define the margin to unacceptable consequences. Margin is provided between the Allowable Value and the Analytical Limits to allow for process, calibration (i.e., M&TE) and some instrument uncertainties. Additional margin is provided between the Allowable Value and the trip setpoint to allow for the remaining instrument uncertainties (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.

In general, the individual Functions are required to be OPERABLE in the MODES or other specified conditions when SCIV/Ds and the SBGT System are required.

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

1. Reactor Vessel Water Level - Low

Low Reactor Pressure Vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. An isolation of the secondary containment and actuation of the SBGT System are initiated in order to minimize the

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Reactor Vessel Water Level – Low (continued)

potential of an offsite dose release. The Reactor Vessel Water Level – Low Function is one of the Functions assumed to be OPERABLE and capable of providing isolation and initiation signals. The isolation and initiation systems on and Reactor Vessel Water Level – Low support actions to ensure that any offsite releases are within the limits calculated in the safety analysis.

Reactor Vessel Water Level – Low signals are initiated from level switches 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 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level-Low Allowable Value was chosen to be the same as the Reactor Protection System Reactor Vessel Level-Low Allowable Value (LCO 3.3.1.1), since this provides an early indication that the capability to cool the fuel is being threatened.

The Reactor Vessel Water Level – Low Function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists in the Reactor Coolant System (RCS); thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, this Function is not required. In addition, the Function is also required to be OPERABLE during Operations with a Potential for Draining the Reactor Vessel (OPDRVs) because the capability of isolating potential sources of leakage must be provided to ensure that offsite dose limits are not exceeded if core damage occurs.

2. Drywell Pressure – High

High drywell pressure can indicate a break in the Reactor Coolant Pressure Boundary (RCPB). An isolation of the secondary containment and actuation of the SBGT System are initiated in order to minimize the potential of an offsite

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2. Drywell Pressure-High (continued)

dose release. The isolation on high drywell pressure supports actions to ensure that any offsite releases are within the limits calculated in the safety analysis. However, the Drywell Pressure-High Function associated with the isolation is not assumed in any UFSAR accident or transient analyses. It is retained for the overall redundancy and diversity of the secondary containment isolation instrumentation.

High drywell pressure signals are initiated from pressure switches that sense the pressure in the drywell. Four channels of Drywell Pressure-High Functions are available and are required to be OPERABLE to ensure that no single instrument failure can preclude performance of the isolation function.

The Allowable Value was chosen to be the same as the ECCS drywell Pressure-High Function Allowable Value (LCO 3.3.5.1) since this is indicative of a Loss of Coolant Accident (LOCA).

The Drywell Pressure-High Function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists in the RCS; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. This Function is not required in MODES 4 and 5 because the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES.

3, 4. Reactor Building Exhaust Shaft and Refueling Floor Exhaust Duct - High Radiation

High secondary containment exhaust radiation is an indication of possible gross failure of the fuel cladding. The release may have originated from the primary containment due to a break in the RCPB or the refueling floor due to a fuel handling accident. When Exhaust - High Radiation is detected, secondary containment isolation and actuation of the SBGT System are initiated to limit the release of fission products as assumed in the UFSAR safety analyses (Ref. 3).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3, 4. Reactor Building Exhaust Shaft and Refueling Floor Exhaust Duct - High Radiation (continued)

The Exhaust - High Radiation signals are initiated from radiation detectors that are located either in the reactor building exhaust shaft or in the exhaust duct coming from the refueling floor, respectively. The signal from each detector is input to an individual monitor whose trip outputs are assigned to an isolation channel. Two channels of Reactor Building Exhaust Shaft - High Radiation Function and two channels of Refueling Floor Exhaust Duct - High Radiation Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding.

The Reactor Building Exhaust Shaft and Refueling Floor Exhaust Duct - High Radiation Functions are required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of a pipe break is low due to the RCS pressure and temperature limitations of these MODES; thus, these Functions are not required. In addition, the Functions are also required to be OPERABLE during CORE ALTERATIONS, OPDRVs, and movement of irradiated fuel assemblies in the secondary containment, because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded.

ACTIONS

A Note has been provided to modify the ACTIONS related to secondary containment isolation 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,

(continued)

BASES (continued)

ACTIONS (continued)

with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable secondary containment isolation instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, the Note has been provided to allow separate Condition entry for each inoperable secondary containment isolation instrumentation channel.

A.1

Because of the diversity of sensors available to provide isolation signals and the redundancy of the isolation design, an allowable out of service time of 12 hours for Functions 1 and 2, and 24 hours for Functions 3 and 4 has been shown to be acceptable (Refs. 4 and 5) to permit restoration of any inoperable channel to OPERABLE status. This out of service time is only acceptable provided the associated Function is still maintaining isolation capability (refer to Required Action B.1 Bases). If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action A.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an isolation), Condition C must be entered and its Required Actions taken.

B.1

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in a complete loss of automatic isolation capability for the associated penetration flow path(s) or a complete loss of automatic initiation capability for the SBGT System. A Function is considered to be maintaining secondary containment isolation capability when sufficient channels are OPERABLE or in trip, such that one trip system will generate a trip signal from the given Function on a valid signal. This ensures that one of the two SCIV/Ds in the associated penetration flow path and one SBGT subsystem can be initiated on an isolation signal from the given Function.

(continued)

BASES

ACTIONS

B.1 (continued)

For the Functions with two two-out-of-two logic trip systems (Functions 1 and 2), this would require one trip system to have two channels either OPERABLE or in trip. For the Functions with two one-out-of-one logic trip systems (Functions 3 and 4), this would require one trip system to have one channel either OPERABLE or in trip.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

C.1.1, C.1.2, C.2.1, and C.2.2

If any Required Action and associated Completion Time of Condition A or B are not met, the ability to isolate the secondary containment and start the SBGT System cannot be ensured. Therefore, further actions must be performed to ensure the ability to maintain the secondary containment function. Isolating the secondary containment (closing the ventilation supply and exhaust automatic isolation dampers) and starting the associated SBGT subsystem (Required Actions C.1.1 and C.2.1) performs the intended function of the instrumentation and allows operation to continue.

Alternately, declaring the associated SCIV/Ds or SBGT subsystem(s) inoperable (Required Actions C.1.2 and C.2.2) is also acceptable since the Required Actions of the respective LCOs (LCO 3.6.4.2 and LCO 3.6.4.3) provide appropriate actions for the inoperable components.
One hour is sufficient for plant operations personnel to establish required plant conditions or to declare the associated components inoperable without unnecessarily challenging plant systems.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each Secondary Containment Isolation instrumentation Function are located in the SRs column of Table 3.3.6.2-1.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 secondary containment isolation 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. 4 and 5) assumption of the average time required to perform channel surveillance. That analysis demonstrated the 6 hour testing allowance does not significantly reduce the probability that the SCIV/Ds will isolate the associated penetration flow paths and that the SBGT System will initiate when necessary.

SR 3.3.6.2.1 and SR 3.3.6.2.2

Performance of the CHANNEL CHECK either once every 12 hours or once every 24 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the 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 are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.2.3

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. The CHANNEL FUNCTIONAL TEST verifies acceptable response by verifying the change of state of at least one contact on the relay which inputs into the trip logic. The required contacts not tested during the CHANNEL FUNCTIONAL TEST are tested under the LOGIC SYSTEM FUNCTIONAL TEST. This is acceptable because operating experience shows that the contacts not tested during the CHANNEL FUNCTIONAL TEST normally pass the LOGIC SYSTEM FUNCTIONAL TEST.

The Frequency of 92 days is based on the reliability analysis of References 4 and 5.

SR 3.3.6.2.4

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies 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.

The Frequency of SR 3.3.6.2.4 is based on the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.6.2.5

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required isolation logic for a specific channel. The system functional testing performed on SCIV/Ds and the SBGT System in LCO 3.6.4.2 and LCO 3.6.4.3, respectively, overlaps this Surveillance to provide complete testing of the assumed safety function.

The 24 month Frequency 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.2.5 (continued)

Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. UFSAR, Section 6.2.3
2. UFSAR, Chapter 15.
3. UFSAR, Sections 15.6.6 and 15.7.1.
4. NEDC-30851P-A Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.
5. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.

B 3.3 INSTRUMENTATION

B 3.3.6.3 Low-Low Set (LLS) Instrumentation

BASES

BACKGROUND

The LLS logic and instrumentation is designed to mitigate the effects of postulated thrust loads on the Safety Relief Valve (SRV) discharge lines by preventing subsequent actuations with an elevated water leg in the SRV discharge line. It also mitigates the effects of postulated pressure loads on the torus shell or suppression pool by preventing multiple actuations in rapid succession of the SRVs subsequent to their initial actuation.

Upon initiation, the LLS logic will electrically control the opening and closing of the two LLS SRVs. The LLS setpoints are selected such that the LLS SRVs will stay open longer; thus, releasing more steam (energy) to the suppression pool, and hence more energy (and time) will be required for repressurization and subsequent SRV openings. The LLS logic increases the time between (or prevents) subsequent actuations to allow the vacuum breakers on the SRV discharge line to open and allow the high water leg created from the initial SRV opening to return to (or fall below) its normal water level; thus, reducing thrust loads from subsequent actuations to within their design limits. In addition, the LLS is designed to limit SRV subsequent actuations to one valve, so torus loads will also be reduced. However, this design feature is not critical since simultaneous valve opening is not a concern for the Mark I containment, and since simultaneous opening will not cause water swell problems for a BWR (Ref. 2).

The LLS instrumentation logic is arranged in two divisions with Logic channel A in one division and Logic channel B in the other division (Ref. 1). Each LLS logic channel (e.g., Logic A channel) controls one LLS valve. The LLS logic channels will not actuate their associated LLS valves at their LLS setpoints until the arming portion of the associated LLS logic is satisfied. Arming occurs when any one of the six SRVs opens as indicated by a signal from two of the three pressure switches located on its tailpipe coincident with a high reactor pressure scram signal. Each logic receives tailpipe arming signals from dedicated tailpipe pressure switches on each of the six SRVs, three in Logic A and three in Logic B. Each LLS logic (e.g.,

(continued)

BASES

BACKGROUND (continued)

Logic A) receives its reactor pressure arming signal from the corresponding Reactor Protection System (RPS) trip system (e.g., LLS Logic A receives its reactor pressure arming signal from either RPS channel A_1 or A_2). Either of the two RPS reactor pressure inputs is capable of arming the associated LLS logic. The arming signal seals in until reset. The arming signal from one logic is sent to the other logic and performs the same function as the tailpipe arming signal (i.e., Logic A will arm if it has received a high reactor pressure signal and Logic B has armed).

After arming, opening and closing of each LLS valve is controlled by two reactor pressure switches, one of which causes the LLS SRV to open when its setpoint is exceeded, while the other causes the LLS SRV to reclose when reactor pressure has decreased to the reclose setpoint.

This logic arrangement prevents single instrument failures from precluding the LLS SRV function, since each logic controls one LLS SRV and one LLS SRV is sufficient to perform the LLS SRV function.

APPLICABLE SAFETY ANALYSES

The LLS instrumentation and logic function ensures that the containment loads remain within the primary containment design basis (Ref. 2).

The LLS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires OPERABILITY of sufficient LLS instrumentation channels to ensure successfully accomplishing the LLS function assuming any single instrumentation channel failure within the LLS logic. Therefore, the OPERABILITY of the LLS instrumentation is dependent on the OPERABILITY of the instrumentation channel Function specified in Table 3.3.6.3-1. Each Function must have a required number of OPERABLE channels, with their setpoints within the specified Allowable Value. A channel is inoperable if any of the trip setpoints of the instruments providing input to the channel are not within the respective required Allowable Value. The actual setpoint is calibrated consistent with applicable setpoint

(continued)

BASES

LCO (continued)

methodology assumptions.

Allowable Values are specified for each LLS actuation Function in Table 3.3.6.3-1. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure the setpoints do not exceed the Allowable Value between 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 steam dome pressure), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., on-off sensor or bi-stable trip circuit) changes state. Analytical Limits, where established, are the limiting values of the process parameters used in safety analysis to define the margin to unacceptable consequences. Margin is provided between the Allowable Value and the Analytical Limits to allow for process, calibration (i.e., M&TE) and some instrument uncertainties. Additional margin is provided between the Allowable Value and the trip setpoint to allow for the remaining instrument uncertainties (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 Tailpipe High Pressure Allowable Value is based on ensuring that a proper arming signal is sent to the LLS logic. That is, the pressure switch is at a pressure that is sufficiently low so that an open SRV is detectable.

The Reactor Steam Dome Pressure - High signal is provided from RPS, and the same Reactor Steam Dome Pressure Allowable Value (LCO 3.3.1.1) is specified because it would be expected that LLS would be needed for pressurization events that result in a high pressure scram. Providing LLS after a scram has been initiated would prevent false initiations of LLS at 100% power. The LLS valve open and close Allowable Values are based on the safety analysis performed in Reference 2.

(continued)

BASES (continued)

APPLICABILITY

The LLS instrumentation is required to be OPERABLE in MODES 1, 2, and 3 since considerable energy is in the nuclear system and the SRVs may be needed to provide pressure relief. If the SRVs are needed, then the LLS function is required to ensure that the primary containment design basis is maintained. In MODES 4 and 5, the reactor pressure is low enough that the overpressure limit cannot be approached by assumed operational transients or accidents. Thus, LLS instrumentation and associated pressure relief is not required.

ACTIONS

A.1

The failure of any reactor steam dome pressure instrument channel to provide the arming, SRV opening or closing pressure setpoints for an individual LLS valve does not affect the ability of the other LLS SRV to perform its LLS function. A LLS valve is OPERABLE if the associated logic, (e.g., Logic A), has one Function 1 channel, two Function 2 channels, and two-out-of-three Function 3 channels associated with one of the three SRVs in the same division OPERABLE. Therefore, 24 hours is provided to restore the inoperable channel(s) to OPERABLE status (Required Action A.1). If the inoperable channel(s) cannot be restored to OPERABLE status within the allowable out of service time, Condition D must be entered and its Required Action taken. The Required Actions do not allow placing the channel in trip since this action could result in an instrumented LLS valve actuation. The 24 hour Completion Time is considered appropriate because of the redundancy in the design (two LLS valves are provided and either LLS valve alone can perform the LLS function) and the very low probability of multiple LLS instrumentation channel failures, which render the remaining LLS SRVs inoperable, occurring together with an event requiring the LLS function during the 24 hour Completion Time. The 24 hour Completion Time is also based on the reliability analysis of Reference 3.

(continued)

BASES

ACTIONS (continued)

B.1

Since each LLS logic normally receives inputs from three SRV pressure switch channels per SRV (and also receives inputs from the other SRV pressure switch channels from the other logic by an arming signal), the LLS logic and instrumentation remains capable of performing its safety function if any SRV tailpipe pressure switch becomes inoperable since each SRV LLS arming logic can be actuated by any two out of the three pressure switches installed on the associated SRV tailpipe. Therefore, it is acceptable for plant operation to continue with only two tailpipe pressure switches OPERABLE on each SRV.

Required Action B.1 requires restoration of the tailpipe pressure switches to OPERABLE status prior to entering MODE 2 or 3 from MODE 4 to ensure that all switches are OPERABLE at the beginning of a reactor startup (this is because the switches are not accessible during plant operation). The Required Actions do not allow placing the channel in trip since this action could result in a LLS valve actuation. As noted, LCO 3.0.4 is not applicable, thus allowing entry into MODE 1 from MODE 2 or into MODE 2 from MODE 3 with inoperable channels. This allowance is acceptable since the channels may require MODE 4 operation to effect repair.

C.1

A failure of two or more pressure switches associated with one SRV tailpipe could result in the loss of the LLS function (i.e., actuation of the SRV would go undetected by the LLS logic). However, both the limiting transient (MSIV closure event) and the limiting accident (small break LOCA with early isolation due to loss of offsite power) that the LLS System is designed to mitigate would result in all six SRVs lifting (Ref. 2). Therefore, it would be very unlikely that a single SRV would be required to arm both LLS logics. Therefore, it is acceptable to allow 14 days to restore two pressure switches to each of the associated SRVs to OPERABLE status (Required Action C.1). If at least two tailpipe pressure switches to each associated SRV cannot be restored to OPERABLE status within the allowable out of service time, Condition D must be entered and its Required Action taken. The Required Actions do not allow placing the channels in trip since this action could result in a LLS valve actuation.

(continued)

BASES

ACTIONS

C.1 (continued)

A Note has been provided in the Condition to modify the Required Actions and Completion Times conventions related to LLS Function 3 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 LLS Function 3 channels provide appropriate compensatory measures for separate inoperable Condition entry for each SRV with inoperable tailpipe pressure switches.

D.1

If any Required Action and associated Completion Time of Conditions A, B, or C are not met, or if both LLS valves are inoperable due to inoperable channels, the LLS valves may be incapable of performing their intended function. Therefore, the associated LLS valve(s) must be declared inoperable immediately. A LLS valve is OPERABLE if the associated logic (e.g., Logic A) has one Function 1 channel, two Function 2 channels, and two out of three Function 3 channels associated with one of the three SRVs in the same division OPERABLE.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each LLS instrumentation Function are located in the SRs column of Table 3.3.6.3-1.

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 LLS initiation 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

the reliability analysis (Ref. 3) 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 LLS valves will initiate when necessary.

SR 3.3.6.3.1 and SR 3.3.6.3.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. The CHANNEL FUNCTIONAL TEST verifies acceptable response by verifying the change of state of at least one contact on the relay which inputs into the trip logic. The required contacts not tested during the CHANNEL FUNCTIONAL TEST are tested under the LOGIC SYSTEM FUNCTIONAL TEST. This is acceptable because operating experience shows that the contacts not tested during the CHANNEL FUNCTIONAL TEST normally pass the LOGIC SYSTEM FUNCTIONAL TEST.

The 92 day Frequency is based on the reliability analysis of Reference 3.

A portion of the SRV tailpipe pressure switch channels is located inside the primary containment and is not available for testing during reactor operation. Therefore, SR 3.3.6.3.1 is only required on that portion of the channel that is outside primary containment.

SR 3.3.6.3.3, SR 3.3.6.3.4, and SR 3.3.6.3.5

CHANNEL CALIBRATION is a complete check of the instrument loop and sensor. This test verifies 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. The Frequency of once every 92 days for SR 3.3.6.3.3, 184 days for SR 3.3.6.3.4, and 24 months for SR 3.3.6.3.5 is based on the assumption of the corresponding calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.3.6

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required actuation logic for a specified channel. The system functional testing performed in LCO 3.4.3, "Safety Relief Valves(SRVs)" and LCO 3.6.1.5, "Low-Low Set (LLS) Safety Relief Valves (SRVs)," for SRVs overlaps this test to provide complete testing of the assumed safety function.

The Frequency of once every 24 months for SR 3.3.6.3.6 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 these components usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. UFSAR, Figure 7.6-31.
2. NEDE-30021-P, Low-Low Set Relief Logic System and Lower MSIV Water Level Trip for the DAEC, January 1983.
3. GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," February 1991.

B 3.3 INSTRUMENTATION

B 3.3.7.1 Standby Filter Unit (SFU) System Instrumentation

BASES

BACKGROUND

The SFU System is designed to provide a radiologically controlled environment to ensure the habitability of the control room for the safety of control room operators under all plant conditions. Two independent SFU subsystems are each capable of fulfilling the stated safety function. The instrumentation and controls for the SFU System automatically initiate action to pressurize the control building envelope to minimize the consequences of radioactive material in the control building envelope.

In the event of a Control Building Intake Area Radiation—High signal, the SFU System is automatically started. The outside air is then passed through the charcoal filter and mixed with recirculated air to maintain the control building envelope slightly pressurized with respect to the atmosphere.

The SFU System instrumentation has two trip systems, with each trip system dedicated to its corresponding SFU. Each trip system receives input from the Control Building Intake Area Radiation - High Function, and is arranged in a one-out-of-one logic. The channels include electronic equipment (e.g., radiation monitors) that compare measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a SFU System initiation signal and a Control Building Ventilation System isolation signal to the initiation logic.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The ability of the SFU System to maintain the habitability of the control building envelope is explicitly assumed for certain accidents as discussed in the UFSAR safety analyses (Refs. 1, 2, and 3). SFU System operation ensures that the radiation exposure of control room personnel, through the duration of any one of the postulated accidents, does not exceed limits consistent with GDC 19 of 10 CFR 50, Appendix A.

SFU System instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY (continued)

The OPERABILITY of the SFU System instrumentation is dependent upon the OPERABILITY of the Control Building Intake Area Radiation - High channel Function. Each SFU must have one OPERABLE channel, with the radiation monitor setpoint within the specified Allowable Value. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

This Allowable Value (AV) is based on a qualitative engineering evaluation. The AV is allowed to be up to 50 mR/hr. Control Building intake area background radiation levels are normally less than 1 mR/hr. Radiation levels just below the AV would be 50 times greater than background and should be detected by other plant radiation monitoring. In addition, radiation levels just below the AV would have to go undetected for over 3 days before a person occupying the Control Room for this entire period would exceed dose in Ref. 2. Dose rates resulting from a LOCA, fuel handling event or vessel draindown event are expected to produce airborne radiation level significantly higher than 50 mR/hr. The trip setpoint is high enough above background to prevent spurious actuations. 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., control building intake area radiation level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., radiation monitor output) changes state.

The Applicable Safety Analyses, LCO, and Applicability discussion for the Control Building Intake Area Radiation - High Function follows.

Control Building Intake Area Radiation—High

The control building intake area radiation monitors measure radiation levels exterior to the intake ducting of the control Building. A high radiation level may pose a threat

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Control Building Intake Area Radiation-High (continued)

to control room personnel; thus, this function automatically initiates the SFU System and places the Control Building Ventilation System in the isolation mode.

The Control Building Intake Area Radiation-High Function consists of two independent monitors. Two channels of Control Building Intake Area Radiation-High are available and are required to be OPERABLE to ensure that no single instrument failure can preclude SFU System initiation. The Allowable Value was selected to ensure protection of the control room personnel.

The Control Building Intake Area Radiation-High Function is required to be OPERABLE in MODES 1, 2, and 3 and during CORE ALTERATIONS, OPDRVs, and movement of irradiated fuel assemblies in the secondary containment, to ensure that control room personnel are protected during a LOCA, fuel handling event, or vessel draindown event. During MODES 4 and 5, when these specified conditions are not in progress (e.g., CORE ALTERATIONS), the probability of a LOCA or fuel damage is low; thus, the Function is not required.

ACTIONS

A Note has been provided to modify the ACTIONS related to SFU System 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 SFU System 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 SFU System instrumentation channel.

(continued)

BASES

ACTIONS (continued)

A.1. and A.2

With a Control Building Intake Area Radiation-High Channel inoperable, the initiation capability of the associated SFU subsystem is lost. Therefore, the associated SFU subsystem(s) must be declared inoperable within 1 hour per Required Action A.1, or must be placed in the isolation mode of operation, (i.e., the SFU in operation and the Control Building Ventilation System isolated, within 1 hour per Required Action A.2. Placing the subsystem in the isolation mode ensures that control room personnel will be protected in the event of a Design Basis Accident. The method used to place the SFU subsystem(s) in operation must provide for automatically re-initiating the subsystem(s) upon restoration of power following a loss of power to the SFU subsystem(s).

The 1 hour Completion Time is intended to allow the operator time to place the SFU subsystem(s) in the isolation mode of operation, and is acceptable because it minimizes risk while allowing time for restoration of channels, or for placing the associated SFU in its safety mode, or for entering the applicable Conditions and Required Actions for the inoperable SFU subsystem(s).

SURVEILLANCE REQUIREMENTS

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 other channel is OPERABLE. 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. 4 and 5) 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 SFU System will initiate when necessary.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.7.1.1

Performance of the CHANNEL CHECK once every 24 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the 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 are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

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

SR 3.3.7.1.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. The CHANNEL FUNCTIONAL TEST verifies acceptable response by verifying the change of state of at least one contact on the relay which inputs into the trip logic. The required contacts not tested during the CHANNEL FUNCTIONAL TEST are tested under the LOGIC SYSTEM FUNCTIONAL TEST. This is acceptable because operating experience shows that the contacts not tested during the CHANNEL FUNCTIONAL TEST normally pass the LOGIC SYSTEM FUNCTIONAL TEST.

The Frequency of 92 days is based on the reliability analyses of References 4 and 5.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.7.1.3

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies 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.

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 3.3.7.1.4

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic for a specific channel. The system functional testing performed in LCO 3.7.4, "Standby Filter Unit (SFU) System," overlaps this Surveillance to provide complete testing of the assumed safety function.

The 24 month Frequency is based on operating experience that has shown these components usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. UFSAR, Section 6.4.1.
2. UFSAR, Section 6.4.4.
3. UFSAR, Table 15.7-5.
4. GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," February 1991.
5. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.

B 3.3 INSTRUMENTATION

B 3.3.8.1 Loss of Power (LOP) Instrumentation

BASES

BACKGROUND

Successful operation of the required safety functions of the Emergency Core Cooling Systems (ECCS) is dependent upon the availability of adequate power sources for energizing the various components such as pump motors, motor operated valves, and the associated control components. The LOP instrumentation monitors the 4.16 kv emergency bus voltages and the Startup and Standby Transformer secondary winding voltages. Offsite power is the preferred source of power for the 4.16 Kv emergency buses. If the monitors determine that insufficient power is available, the buses are disconnected from the offsite power sources and connected to the onsite Diesel Generator (DG) power sources. The monitors also provide start permissive signals for the low pressure ECCS pumps and inputs to various load shedding circuits.

Each 4.16 kV emergency bus has its own independent LOP instrumentation and associated trip logic. Each bus is monitored at two different voltage levels, which can be considered as two different undervoltage Functions: 4.16 kV Emergency Bus Loss of Voltage and 4.16 kV Emergency Bus Degraded Voltage. A third LOP Function monitors the voltages of the secondary windings of both the Startup and Standby Transformers (4.16 kV Emergency Transformer Supply Undervoltage). Each of these Functions provide inputs to various required actions such as: bus transfers, load sheds, or DG starts. Each emergency bus is monitored for the Loss of Voltage Function by a single undervoltage relay. Each bus is monitored for the Degraded Voltage Function by four relays whose contacts form a coincidence logic matrix such that either of the A1 or A2 contacts and either of the B1 or B2 contacts must close to initiate the required actions in the associated division (i.e., one-out-of-two taken twice). For the Emergency Transformer Supply Undervoltage Function, the secondary winding of the Standby Transformer is monitored by two channels. One channel provides an input to initiate the required actions in one division, while the other channel provides input to initiate the required actions in the other division. Each channel of the Emergency Transformer Supply Undervoltage Function includes two relays. Both of the secondary windings

(continued)

BASES

BACKGROUND (continued)

of the Startup Transformer (i.e., a separate secondary winding is associated with each division) are monitored by a single relay for this Function; the relay that monitors the winding associated with each division provides an input to initiate the required actions in that division. The initiation logic for each division receives inputs from both the Startup and Standby Transformers secondary windings (i.e., two-out-of two relay logic). The various channels include monitoring equipment (i.e., current transformers or voltage transformers) and relays that compare measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a LOP signal to its associated trip logic.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The LOP instrumentation is required for Engineered Safety Features (ESF) to function in any accident with a loss of offsite power. The required channels of LOP instrumentation ensure that the ECCS and other assumed systems powered from the DGs, provide plant protection in the event of any of the Reference 1, 2, and 3 analyzed accidents in which a loss of offsite power is assumed. The LOP instrumentation also serves to provide protection for the ESF from damage that could occur due to prolonged degraded voltage conditions on the grid. The initiation of the DGs on a LOP or LOCA signal, and concurrent initiation of the ECCS on a LOCA signal ensure that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. Disconnecting the emergency buses from the offsite sources and starting the DGs when a prolonged degraded voltage condition exists ensures that the ESF (e.g., large pump motors) are not damaged due to starting with insufficient voltage present.

Accident analyses credit the loading of the DG based on the loss of offsite power during a LOCA. The diesel starting and loading times have been included in the delay time associated with each safety system component requiring DG supplied power following a loss of offsite power.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The LOP instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

The OPERABILITY of the LOP instrumentation is dependent upon the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.8.1-1. Each Function must have a required number of OPERABLE channels per 4.16 kV emergency bus, with their setpoints within the specified Allowable Values. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

The Allowable Values are specified for each Function in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within the 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., degraded voltage), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., relay) changes state. Analytical Limits are the limiting values of the process parameters for use in safety analysis. Margin is provided between the Allowable Value and the Analytical Limits to allow for process, calibration (i.e., M&TE) and some instrument uncertainties. Additional margin is provided between the Allowable Value and the trip setpoint to allow for the remaining instrument uncertainties (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 specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY (continued)

1. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)

Loss of voltage on a 4.16 kV emergency bus indicates that offsite power has been completely lost to the respective emergency bus and is unable to supply sufficient power for proper operation of the applicable equipment. Therefore, the power supply to the bus is transferred from offsite power to DG power and the various loads connected to the emergency bus are load shed when the voltage on the bus drops below the Loss of Voltage Function Allowable Values. This ensures that adequate power will be available to the required equipment.

The Bus Undervoltage Allowable Values are low enough to prevent power supply transfer and load shedding unless offsite power has truly been lost, but high enough to ensure that power is available to the required equipment.

One channel of 4.16 kV Emergency Bus Undervoltage (Loss of Voltage) Function per associated emergency bus is only required to be OPERABLE in MODES 1, 2, 3, and when the associated DG is required to be OPERABLE by LCO 3.8.2, to ensure that no single instrument failure can preclude the DG function. (One channel provides input to the initiation logic for its respective division.) Refer to LCO 3.8.1, "AC Sources – Operating," and 3.8.2, "AC Sources – Shutdown," for Applicability Bases for the DGs.

2. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)

A reduced voltage condition on a 4.16 kV emergency bus indicates that, while offsite power may not be completely lost to the respective emergency bus, available power may be insufficient for starting large ECCS motors without risking damage to the motors that could disable the ECCS function. Therefore, power supply to the bus is transferred from offsite power to onsite DG power when the voltage on the bus drops below the Degraded Voltage Function Allowable Values (degraded voltage with a time delay). This ensures that adequate power will be available to the required equipment.

The Bus Undervoltage Allowable Values are low enough to prevent inadvertent power supply transfer, but high enough to ensure that sufficient power is available to the required

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage) (continued)

equipment. The Time Delay Allowable Values are long enough to provide time for the offsite power supply to recover to normal voltages, but short enough to ensure that sufficient power is available to the required equipment.

Four channels of 4.16 kv Emergency Bus Undervoltage (Degraded Voltage) Function per associated bus are only required to be OPERABLE in MODES 1, 2, and 3, and when the associated DG is required to be OPERABLE by LCO 3.8.2, to ensure that no single instrument failure can preclude the DG function. Refer to LCO 3.8.1 and LCO 3.8.2 for Applicability Bases for the DGs.

3. 4.16 kV Emergency Transformer Supply Undervoltage

An undervoltage condition on either the Startup Transformer secondary winding or the Standby Transformer secondary winding indicates that offsite power may be lost and is unable to supply sufficient power for proper operation of the applicable equipment. This Function is similar to Function 1, except whereas Function 1 monitors each emergency bus voltage, Function 3 monitors the voltage of the offsite sources directly via the secondary windings of the two transformers associated with the two offsite sources. Therefore, when an undervoltage condition exists on an individual transformer, the supply breakers from that transformer to the emergency buses are tripped open. If an undervoltage condition exists on both transformers, then the supply breakers from both transformers to the emergency buses are tripped open and a start signal is sent to the associated DG. This ensures that adequate power will be available to the required equipment.

The Emergency Transformer Supply Undervoltage Allowable Values are low enough to prevent inadvertent power supply transfer, but high enough to ensure that power is available to the required equipment.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. 4.16 kV Emergency Transformer Supply Undervoltage (continued)

Two channels of 4.16 kv Emergency Transformer Supply Undervoltage Function (one channel from the secondary winding of the Startup Transformer and one channel from the secondary winding of the Standby Transformer) per associated emergency bus are only required to be OPERABLE in MODES 1, 2, and 3, and when the associated DG is required to be OPERABLE by LCO 3.8.2, to ensure that no single failure can preclude the DG function. Refer to LCO 3.8.1 and LCO 3.8.2 for Applicability Bases for the DGs.

ACTIONS

A Note has been provided to modify the ACTIONS related to LOP 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 LOP 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 LOP instrumentation channel.

A.1

With one or more channels of Function 1 or 3 inoperable, the Function is not capable of performing the intended function. Therefore, only 1 hour is allowed to restore the inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action A.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure (within the LOP instrumentation), and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the channel in trip would result in a DG

(continued)

BASES

ACTIONS

A.1 (continued)

initiation). Condition C must be entered and its Required Action taken. The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

B.1 and B.2

With certain combinations of multiple Function 2 channels inoperable, the ability of the 4.16 kv Emergency Bus Undervoltage (Degraded Voltage) Function to separate the emergency buses from the offsite sources and to start the DG(s) may be lost. Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels of Function 2 result in automatic initiation capability being lost in one or both divisions. If initiation capability is lost, the associated DG(s) must be immediately declared inoperable. This requires entry into applicable Conditions and Required Actions of LCO 3.8.1 or LCO 3.8.2, which provide appropriate actions for the inoperable DG(s).

With one Function 2 channel inoperable, or with certain combinations of multiple Function 2 channels inoperable, the ability of the 4.16 kv Emergency Bus Undervoltage (Degraded Voltage) Function to initiate the required actions in either or both divisions is maintained by the remaining OPERABLE channels. In this situation, the requirement to place a channel in trip within 1 hour as in Required Action A.1, or to immediately declare the associated DG inoperable, is overly conservative. The 24 hour Completion Time of Required Action B.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 24 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

(continued)

BASES

ACTIONS (continued)

C.1

If the Required Action and associated Completion Time is not met, the associated Function is not capable of performing the intended function. Therefore, the associated DG(s) is declared inoperable immediately. This requires entry into applicable Conditions and Required Actions of LCO 3.8.1 and LCO 3.8.2, which provide appropriate actions for the inoperable DG(s).

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each LOP instrumentation Function are located in the SRs column of Table 3.3.8.1-1.

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 2 hours provided the associated Function maintains DG initiation capability. Upon completion of the Surveillance, or expiration of the 2 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken.

SR 3.3.8.1.1 and SR 3.3.8.1.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. The CHANNEL FUNCTIONAL TEST verifies acceptable response by verifying the change of state of at least one contact on the relay which inputs into the trip logic. The required contacts not tested during the CHANNEL FUNCTIONAL TEST are tested under the LOGIC SYSTEM FUNCTIONAL TEST. This is acceptable because operating experience shows that the contacts not tested during the CHANNEL FUNCTIONAL TEST normally pass the LOGIC SYSTEM FUNCTIONAL TEST.

The Frequencies of 31 days and 12 months are based on 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 or 12 month interval (as appropriate) is a rare event.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.8.1.3 and SR 3.3.8.1.4

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies 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. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequencies are based upon the assumption of either a 12 month or 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.8.1.5

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required actuation logic for a specific channel. The system functional testing performed in LCO 3.8.1 and LCO 3.8.2 overlaps this Surveillance to provide complete testing of the assumed safety functions.

The 24 month Frequency 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 these components usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. UFSAR, Section 5.2.
2. UFSAR, Section 6.3.
3. UFSAR, Chapter 15.

B 3.3 INSTRUMENTATION

B 3.3.8.2 Reactor Protection System (RPS) Electric Power Monitoring

BASES

BACKGROUND

RPS Electric Power Monitoring System is provided to isolate the RPS bus from the Motor Generator (MG) set or an alternate power supply in the event of overvoltage, undervoltage, or underfrequency. This system protects the loads connected to the RPS bus against unacceptable voltage and frequency conditions (Ref. 1) and forms an important part of the primary success path of the essential safety circuits. Some of the essential equipment powered from the RPS buses includes the RPS logic, scram solenoids, and Groups 1 through 5 isolation logic.

Each RPS Electrical Protection Assembly (EPA) will detect any abnormal high or low voltage or low frequency condition in the outputs of the two MG sets or the alternate power supply and will de-energize its respective RPS bus, thereby causing all safety functions normally powered by this bus to de-energize.

In the event of failure of an RPS Electric Power Monitoring System (e.g., both in-series EPAs), the RPS loads may experience significant effects from the unregulated power supply. Deviation from the nominal conditions can potentially cause damage to the scram solenoids and other Class 1E devices.

In the event of a low voltage condition for an extended period of time, the scram solenoids can chatter and potentially lose their pneumatic control capability, resulting in a loss of primary scram action.

In the event of an overvoltage condition, the RPS logic relays and scram solenoids may experience a voltage higher than their design voltage. If the overvoltage condition persists for an extended time period, it may cause equipment degradation and the loss of plant safety function.

Two redundant Class 1E circuit breakers are connected in series between each RPS bus and its MG set, and between the RPS buses and the alternate power supply. Each of these circuit breakers has an associated independent set of Class 1E overvoltage, undervoltage, and underfrequency

(continued)

BASES

BACKGROUND (continued)

sensing logic. Together, a circuit breaker and its sensing logic constitute an EPA. If the output of the MG set or the alternate power supply exceeds predetermined limits of overvoltage, undervoltage, or underfrequency, after a short time delay, (if applicable), a trip coil driven by this logic circuitry opens the circuit breaker, which removes the associated power supply from service.

APPLICABLE SAFETY ANALYSES

The RPS electric power monitoring is necessary to meet the assumptions of the safety analyses by ensuring that the equipment powered from the RPS buses can perform its intended function. RPS electric power monitoring provides protection to the RPS and other systems that receive power from the RPS buses, by acting to disconnect the RPS from the power supply under specified conditions that could damage the RPS bus powered equipment.

RPS electric power monitoring satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The OPERABILITY of each RPS EPA is dependent on the OPERABILITY of the overvoltage, undervoltage, and underfrequency logic, as well as the OPERABILITY of the associated circuit breaker. Two EPAs are required to be OPERABLE for each inservice power supply. This provides redundant protection against any abnormal voltage or frequency conditions to ensure that no single RPS EPA failure can preclude the function of RPS bus powered components. Each inservice EPA's trip logic setpoints are required to be within the specified Allowable Value. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

Allowable Values are specified for each RPS EPA trip logic (refer to SR 3.3.8.2.2). Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the

(continued)

BASES

LCO (continued)

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., overvoltage), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip coil) changes state. The Allowable Values (AV) are based on the nominal power supply voltage and frequency requirements for RPS components because there are no Analytic Limits in the safety analysis from which to derive the AVs. The nominal trip setpoints are determined from the AVs, accounting for instrument errors and provide adequate protection. 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 Allowable Values for the instrument settings are based on the RPS providing ≥ 57 Hz, 120 V ± 10%, to the RPS bus. The settings are calculated based on the loads on the buses and RPS MG set or alternate power supply being 120 VAC and 60 Hz.

APPLICABILITY

The operation of the RPS EPAs is essential to disconnect the RPS bus powered components from the MG set or alternate power supply during abnormal voltage or frequency conditions. Since the degradation of a nonclass 1E source supplying power to the RPS bus can occur as a result of any random single failure, the OPERABILITY of the RPS EPAs is required when the RPS bus powered components are required to be OPERABLE. This results in the RPS Electric Power Monitoring System OPERABILITY being required in MODES 1 and 2; and in MODES 3, 4 and 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies.

(continued)

BASES (continued)

ACTIONS

A.1

If one RPS EPA for an inservice power supply (MG set or alternate) is inoperable, or one RPS EPA on each inservice power supply is inoperable, the OPERABLE EPA will still provide protection to the RPS bus powered components under degraded voltage or frequency conditions. However, the reliability and redundancy of the RPS Electric Power Monitoring System is reduced, and only a limited time (72 hours) is allowed to restore the inoperable EPA to OPERABLE status. If the inoperable EPA cannot be restored to OPERABLE status, the associated power supply(s) must be removed from service (Required Action A.1). This places the RPS bus in a safe condition. An alternate power supply with OPERABLE EPAs may then be used to power the RPS bus.

The 72 hour Completion Time takes into account the remaining OPERABLE EPA and the low probability of an event requiring RPS electric power monitoring protection occurring during this period. It allows time for plant operations personnel to take corrective actions or to place the plant in the required condition in an orderly manner and without challenging plant systems.

Alternately, if it is not desired to remove the power supply from service (e.g., as in the case where removing the power supply(s) from service would result in a scram or isolation), Condition C or D, as applicable, must be entered and its Required Actions taken.

B.1

If both EPAs for an inservice power supply (MG set or alternate) are inoperable or both EPAs in each inservice power supply are inoperable, the system protective function is lost. In this condition, 1 hour is allowed to restore one assembly to OPERABLE status for each inservice power supply. If one inoperable EPA for each inservice power supply cannot be restored to OPERABLE status, the associated power supply(s) must be removed from service within 1 hour (Required Action B.1). An alternate power supply with OPERABLE EPAs may then be used to power one RPS bus. The 1 hour Completion Time is sufficient for the plant operations personnel to take corrective actions and is

(continued)

BASES

ACTIONS

B.1 (continued)

acceptable because it minimizes risk while allowing time for restoration of the EPA or removal of the associated power supply from service.

Alternately, if it is not desired to remove the power supply(s) from service (e.g., as in the case where removing the power supply(s) from service would result in a scram or isolation), Condition C or D, as applicable, must be entered and its Required Actions taken.

C.1

If any Required Action and associated Completion Time of Condition A or B are not met in MODE 1 or 2, a plant shutdown must be performed. This places the plant in a condition where minimal equipment, powered through the inoperable RPS EPA(s), is required and ensures that the safety function of the RPS (e.g., scram of control rods) is not required. The plant shutdown is accomplished by placing the plant in MODE 3 within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

If any Required Action and associated Completion Time of Condition A or B are not met in MODES 3, 4 or 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, the operator must immediately initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Required Action D.1 results in the least reactive condition for the reactor core and ensures that the safety function of the RPS (e.g., scram of control rods) is not required.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.8.2.1

A CHANNEL FUNCTIONAL TEST is performed on each overvoltage, undervoltage, and underfrequency channel to ensure that the channel will perform the intended function. The CHANNEL FUNCTIONAL TEST verifies acceptable response by verifying the change of state of at least one contact on the relay which inputs into the trip logic. The required contacts not tested during the CHANNEL FUNCTIONAL TEST are tested under the LOGIC SYSTEM FUNCTIONAL TEST. This is acceptable because operating experience shows that the contacts not tested during the CHANNEL FUNCTIONAL TEST normally pass the LOGIC SYSTEM FUNCTIONAL TEST.

As noted in the Surveillance, the CHANNEL FUNCTIONAL TEST is only required to be performed while the plant is in a condition in which the loss of the RPS bus will not jeopardize steady state power operation (the design of the system is such that the power source must be removed from service to conduct the Surveillance). The 24 hours is intended to indicate an outage of sufficient duration to allow for scheduling and proper performance of the Surveillance.

The 184 day Frequency and the Note in the Surveillance are based on guidance provided in Generic Letter 91-09 (Ref. 2).

SR 3.3.8.2.2

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test 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.

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

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.8.2.3

Performance of a system functional test demonstrates that, with a required system actuation (simulated or actual) signal, the system will automatically trip open the associated EPA. Only one signal per EPA is required to be tested. This Surveillance overlaps with the CHANNEL CALIBRATION to provide complete testing of the safety function. The system functional test of the Class 1E circuit breakers is included as part of this test to provide complete testing of the safety function. If the breakers are incapable of operating, the associated EPA would be inoperable.

The 24 month Frequency 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.

REFERENCES

1. UFSAR, Section 7.2.1.1.2.

2. NRC Generic Letter 91-09, "Modification of Surveillance Interval for the Electrical Protective Assemblies in Power Supplies for the Reactor Protection System."