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Procedure Manual
WSRC-TS-10003-Diver
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100-244

**SAVANNAH RIVER SITE
PRODUCTION REACTOR
TECHNICAL SPECIFICATIONS (U)
K PRODUCTION REACTOR**

Westinghouse Savannah River Company
Savannah River Site
Aiken, SC 29808

PREPARED FOR THE U.S. DEPARTMENT OF ENERGY UNDER CONTRACT NO. DE-AC09-89SR18035

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Savannah River Site Production Reactor Technical Specifications

K Production Reactor

Westinghouse Savannah River Company
Savannah River Laboratory
Aiken, SC 29808

SAVANNAH RIVER SITE
PREPARED FOR THE U.S. DEPARTMENT OF ENERGY UNDER CONTRACT NO. DE-AC05-82SR12035

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MASTER

PREFACE

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Technical Specifications are explicit restrictions on the operation of the Savannah River Site K Production Reactor. They are designed to preserve the validity of the plant safety analysis by ensuring that the plant is operated within the required conditions bounded by the analysis, and with the operable equipment that is assumed to mitigate the consequences of an accident. Technical Specifications preserve the primary success path relied upon to detect and respond to accidents.

DOE Order 5480.6 states that each DOE-owned reactor shall have Technical Specifications in accordance with 10 CFR 50.36.

Section 182a of the Atomic Energy Act of 1954, as amended (the Act), 47 U.S.C. 2011, at 2232, provides the legislative framework within which technical specifications are required. Section 182a of the Act requires in pertinent part:

"In connection with application for licenses to operate production or utilization facilities, the applicant shall state such technical specifications, including information on the amount, kind, and source of special nuclear material required, the place of use, the specific characteristics of the facility, and such other information as the Commission may, by rule or regulation, deem necessary in order to enable it to find that the utilization or production of special nuclear material will ...provide adequate protection to the health and safety of the public. Such technical specifications shall be a part of any license issued."

The regulatory framework implementing Section 182a of the Act is the NRC's regulation Title 10 Code of Federal Regulations Part 50, Section 50.36, "Technical Specifications". This regulation provides in part that each operating license:

"will include technical specifications...(to) be derived from the analysis and evaluation included in the safety analysis report, and amendments thereto...and may also include such additional technical specifications as the Commission finds appropriate."

The Commission has issued an interim "Proposed Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," 52 FR 3788, February 6, 1987. This Interim Policy Statement sets out specific criteria for the content of technical specifications. The interim policy statement specifically recognizes that:

"The purpose of Technical Specifications is to impose conditions or limitations upon reactor operation necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety by establishing those conditions of operation which cannot be changed without prior Commission approval and by identifying those features which are of controlling importance to safety."

These Technical Specifications establish these conditions and limitations for the Savannah River Site K Production Reactor.

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3.9.8-1	0	12/18/90	5-36	2	Deleted
3.9.8-2	3	08/22/91	5-37	2	Deleted
3.9.8-3	3	08/22/91	B 2.1-1	3	08/22/91
3.9.9-1	0	12/18/90	B 2.2-1	3	08/22/91
3.10.1-1	12	06/24/92	B 2.3-1	3	08/22/91
3.10.1-2	12	06/24/92	B 2.3-2	3	08/22/91
4.0-1	3	08/22/91	B 3.0-1	0	12/18/90
4.0-2	3	08/22/91	B 3.0-2	0	12/18/90
4.0-3	12	06/24/92	B 3.0-3	0	12/18/90
4.0-4	3	08/22/91	B 3.0-4	0	12/18/90
5-1	2	05/03/91	B 3.0-5	0	12/18/90
5-2	2	05/03/91	B 3.0-6	3	08/22/91
5-3	2	05/03/91	B 3.0-7	0	12/18/90
5-4	15	01/14/93	B 3.0-8	0	12/18/90
5-5	2	05/03/91	B 3.0-9	6	12/06/91
5-6	2	05/03/91	B 3.0-10	0	12/18/90
5-7	3	08/22/91	B 3.1.1-1	0	12/18/90
5-8	3	08/22/91	B 3.1.1-2	0	12/18/90
5-9	2	05/03/91	B 3.1.1-3	0	12/18/90
5-10	2	05/03/91	B 3.1.2-1	12	06/24/92
5-11	2	05/03/91	B 3.1.2-2	3	08/22/91
5-12	2	05/03/91	B 3.1.2-3	3	08/22/91
5-13	2	05/03/91	B 3.1.2-4	0	12/18/90
5-14	3	08/22/91	B 3.1.2-5	0	12/18/90
5-15	3	08/22/91	B 3.1.2-6	12	06/24/92
5-16	2	05/03/91	B 3.1.2-7	0	12/18/90
5-17	2	05/03/91	B 3.1.2-8	0	12/18/90
5-18	2	05/03/91	B 3.1.3-1	15	01/14/93
5-19	2	05/03/91	B 3.1.3-2	12	06/24/92
5-20	2	05/03/91	B 3.1.3-3	12	06/24/92
5-21	2	05/03/91	B 3.1.3-4	12	06/24/92
5-22	2	05/03/91	B 3.1.3-5	3	Deleted
5-23	2	05/03/91	B 3.1.4-1	0	12/18/90
5-24	2	05/03/91	B 3.1.4-2	3	08/22/91
5-25	15	01/14/93	B 3.1.4-3	0	12/18/90
5-26	2	05/03/91	B 3.1.4-4	3	08/22/91
5-27	2	05/03/91	B 3.1.4-5	3	08/22/91
5-28	2	05/03/91	B 3.1.5-1	3	08/22/91
5-29	2	Deleted	B 3.1.5-2	3	Deleted
5-30	2	Deleted	B 3.1.6-1	3	08/22/91
5-31	2	Deleted	B 3.1.6-2	3	08/22/91
5-32	2	Deleted	B 3.1.6-3	0	12/18/90
5-33	2	Deleted	B 3.1.7-1	3	08/22/91

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B 3.2.2-2	4	10/24/91	B 3.4.3.1-2	11	06/05/92
B 3.2.2-3	14	07/10/92	B 3.4.3.1-3	0	12/18/90
B 3.2.2-4	4	10/24/91	B 3.4.3.3-1	3	08/22/91
B 3.2.2-5	4	10/24/91	B 3.4.3.3-2	3	08/22/91
B 3.2.2-6	4	10/24/91	B 3.4.4-1	3	08/22/91
B 3.2.2-7	10	05/15/92	B 3.4.4-2	0	12/18/90
B 3.2.2-8	14	07/10/92	B 3.4.5.1-1	3	08/22/91
B 3.2.2-9	14	07/10/92	B 3.4.5.1-2	0	12/18/90
B 3.2.2-10	4	Deleted	B 3.4.5.2-1	3	08/22/91
B 3.3.1-1	3	08/22/91	B 3.4.6-1	0	12/18/90
B 3.3.1-2	3	08/22/91	B 3.4.7-1	0	12/18/90
B 3.3.1-3	3	08/22/91	B 3.4.8.1-1	0	12/18/90
B 3.3.1-4	3	08/22/91	B 3.4.8.1-2	3	08/22/91
B 3.3.1-5	3	08/22/91	B 3.4.8.2-1	3	08/22/91
B 3.3.2-1	0	12/18/90	B 3.4.8.3-1	15	01/14/93
B 3.3.2-2	0	12/18/90	B 3.4.8.3-2	15	01/14/93
B 3.3.3-1	3	08/22/91	B 3.4.8.3-3	15	01/14/93
B 3.3.4-1	0	12/18/90	B 3.4.8.3-4	15	01/14/93
B 3.3.4-2	3	08/22/91	B 3.4.8.3-5	15	01/14/93
B 3.3.5-1	3	08/22/91	B 3.4.8.3-6	15	01/14/93
B 3.3.5-2	3	08/22/91	B 3.4.8.3-7	15	01/14/93
B 3.3.5-3	3	08/22/91	B 3.4.8.4-1	0	12/18/90
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B 3.3.6-2	9	04/16/92	B 3.5.1.1-5	3	08/22/91
B 3.3.6-3	9	04/16/92	B 3.5.1.1-6	0	12/18/90
B 3.3.6-4	9	04/16/92	B 3.5.1.1-7	3	08/22/91
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B 3.5.2.1-3	3	08/22/91	B 3.9.1-1	0	12/18/90
B 3.5.2.1-4	3	08/22/91	B 3.9.1-2	0	12/18/90
B 3.5.2.1-5	3	08/22/91	B 3.9.3-1	3	08/22/91
B 3.5.2.1-6	3	08/22/91	B 3.9.3-1	0	12/18/90
B 3.5.2.1-7	3	08/22/91	B 3.9.3-2	0	12/18/90
B 3.5.2.1-8	3	08/22/91	B 3.9.4.1-1	0	12/19/90
B 3.5.2.1-9	3	08/22/91	B 3.9.4.2-1	3	08/22/91
B 3.5.2.1-10	3	08/22/91	B 3.9.5-1	0	12/18/90
B 3.5.2.1-11	3	08/22/91	B 3.9.5-2	0	12/18/90
B 3.5.2.1-12	3	08/22/91	B 3.9.5-3	0	12/18/90
B 3.5.2.1-13	3	08/22/91	B 3.9.6-1	3	08/22/91
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B 3.5.2.1-15	3	08/22/91	B 3.9.7.2	8	Deleted
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B 3.5.2.2-3	3	08/22/91			
B 3.5.2.2-4	3	08/22/91			
B 3.5.2.2-5	3	08/22/91			
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B 3.5.2.2-7	3	08/22/91			
B 3.4.2.2-8	3	08/22/91			
B 3.6.1-1	0	12/18/90			
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INFORMATION ONLY

Definitions
1.1

1.1 DEFINITIONS

NOTE

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases

<u>Term</u>	<u>Definition</u>
ACTIONS	ACTIONS shall be that part of a Technical Specification which prescribes Required Actions under designated Conditions which shall be completed within specified Completion Times.
ALLOWABLE VALUE	An ALLOWABLE VALUE is the minimum or maximum as found value that a scram or other actuation circuit (excluding sensor uncertainty) can be at before the circuit must be declared inoperable.
AXIAL POWER RATIO	The AXIAL POWER RATIO is the RTR for axial shapes.
BURNOUT RISK (BOR)	The risk of fission product release from film boiling burnout is defined as BURNOUT RISK (BOR) and is equal to the product of the consequences of burnout and the probability that it occurs.
BYPASS/BYPASSED	To electrically or mechanically align a circuit or component, using an approved procedure, to intentionally defeat the OPERABILITY of the circuit or component.
CHANNEL CALIBRATION	CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds within the required range and accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire CHANNEL/CIRCUIT including the sensors and alarm, interlock and/or trip functions and may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.
CHANNEL CHECK	A CHANNEL CHECK is a qualitative verification of acceptable performance by observation of CHANNEL/CIRCUIT behavior. This verification, where possible, includes comparison of the CHANNEL/CIRCUIT with other independent CHANNEL/CIRCUIT, known values or other circuits/systems monitoring the same variable.

(continued)

1.1 DEFINITIONS (continued)

CHANNEL TEST	A CHANNEL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock and/or trip functions (during REACTOR STARTUP and REACTOR OPERATION the CHANNEL TEST will terminate at the bypass switch). The CHANNEL TEST shall include adjustments, as necessary, of the alarm, interlock and/or trip setpoints such that the setpoints are within the required range and accuracy.
CHANNEL/CIRCUIT	A CHANNEL/CIRCUIT is the combination of sensor, line (cable or pneumatic), amplifier, signal conditioner and output devices which are required to complete the Technical Specification action of monitoring, protection and/or control of a parameter and/or piece of equipment.
CHARGE AND DISCHARGE OPERATIONS (C&D)	CHARGE AND DISCHARGE OPERATIONS (C&D) are those activities associated with alterations to the configuration of the reactor core when the actuator is raised that cause changes in reactivity as identified in the COR.
CONTROLLED REACTOR SHUTDOWN	A CONTROLLED REACTOR SHUTDOWN is a shutdown accomplished in a controlled manner by control and safety rod insertion.
CONTROLLING ROD	A CONTROLLING ROD is a rod/rods being used for reactor control by the M unit (Console Controls) or V unit (Trim Panel Controls).
CONTROL ROD SEQUENCE	The CONTROL ROD SEQUENCE for a full control rod cluster shall be as defined by the following sequence of withdrawal:

<u>Sequence No.</u>	<u>Rod Description</u>
1	cadmium
2	cadmium
3	strong (2 g/ft. ⁶ Li)
4	weak (0.55 g/ft. ⁶ Li)
5	weak (0.55 g/ft. ⁶ Li)

(continued)

1.1 DEFINITIONS (continued)

CORE OPERATION REPORT (COR)	The COR is the reactor and subcycle specific Core Physics and Thermal-Hydraulic Reactor Technology Memoranda that contains configuration, type, and contents of assemblies in the reactor; temperature coefficients, predicted reactivities, predicted margins of control, temperature limits for all assemblies, information for online calculations, and all flow design information in accordance with Specification 5.5.15. The memorandum shall be clearly identified to show applicability to reactor, fuel cycle, and target subcycle.
CORE SHUTDOWN REPORT (CSR)	The CSR is a reactor charge specific document that provides shutdown cooling time limits for HGA's based on forced cooling in the reactor, D-machine cooling, assembly in-air, and assembly horizontal in the disassembly basin. A special CSR will be issued for any failed assembly that requires discharge from the reactor. The cycle specific shutdown cooling limits shall be determined for each cycle in accordance with Specification 5.5.16. Plant operation within these limits is addressed in individual specifications.
CORNER CLUSTERS	CORNER CLUSTERS are clusters 41, 45, 49, 53, 57 and 61.
DOUBLE PARTIAL ROD COMPLEMENT	A DOUBLE PARTIAL ROD COMPLEMENT is a control rod configuration in which both partial rods in a cluster are inserted to the PARTIAL ROD DEMAND POSITION.
ENABLE	A switch function on logic circuits. When the switch is placed in the ENABLE position a valid input is simulated which reduces, by one, the number of valid inputs to the logic circuit to cause an output.
FISSILE MATERIAL	FISSILE MATERIAL is any isotope that can be fissioned by absorption of a neutron.
FULL POWER	FULL POWER is that power level corresponding to operation at the minimum thermal-hydraulic limit as specified in the COR for the most limiting parameter in the reactor core. FULL POWER is not itself a limit, but rather a result of operation at temperature limits.
FULL ROD DEMAND POSITION	The FULL ROD DEMAND POSITION is the veeder unit (vu) readout of the gang N unit plus the full rod trim.

(continued)

1.1 DEFINITIONS (continued)

FULLY INSERTED	A safety rod or control rod is FULLY INSERTED when it is off its bottom limit by ≤ 4 inches.
HEAT GENERATING ASSEMBLY (HGA)	Any fuel or target assembly that has undergone neutron irradiation during REACTOR OPERATION and requires water for cooling as defined in the CSR.
IMMEDIATELY	IMMEDIATELY is used as a Completion Time when a Condition can not be permitted to continue and the Required Action must be initiated without delay and continued until the Required Action is completed.
INITIAL RADIAL SHAPE FACTOR	The INITIAL RADIAL SHAPE FACTOR is the first RADIAL SHAPE FACTOR (RSF) measured during each reactor startup/power ascension when power is $< \text{ISAL}$.
INTEGRAL CONTROL ROD	INTEGRAL CONTROL ROD is a full length control rod which is FULLY INSERTED out of the normal control rod sequence. (This does not include unlatched faulted or MISALIGNED full length control rods.)
LEAKAGE-IDENTIFIED	IDENTIFIED LEAKAGE is: <ul style="list-style-type: none">a. Leakage into a closed system, such as pump seals or valve packing leaks that are captured and conducted to a collecting tank or container; orb. Leakage from sources that are both specifically located and known either not to interfere with the operation of leakage detection system or known not to be PRESSURE BOUNDARY LEAKAGE, e.g., leakage into the Purification Area; orc. Process water system leakage through the heat exchangers to the cooling water system.
LEAKAGE-PRESSURE BOUNDARY	PRESSURE BOUNDARY LEAKAGE is leakage (except heat exchanger tube leakage and head flange seepage) through a non-isolable fault in a Process Water System component body, pipe wall, or reactor tank wall.
LEAKAGE-UNIDENTIFIED	UNIDENTIFIED LEAKAGE is all leakage which is not IDENTIFIED LEAKAGE.

(continued)

1.1 DEFINITIONS (continued)

MARGIN OF CONTROL	MARGIN OF CONTROL (MOC) is the k_{eff} difference between unity and the charge reactivity for the condition where all safety rods are removed, all control rods are FULLY INSERTED, the reactor is xenon free and reactor temperature is at the lowest cooling water temperature assumed in the Safety Analysis.
MISALIGNED	A MISALIGNED full or partial length control rod is a rod which is out of demand position by greater than 4 veeder units and can not be restored to within 4 veeder units within 8 hours.
OPERABLE-OPERABILITY	A system, subsystem, component or device shall be OPERABLE when it is capable of performing its specified function(s) or is placed and maintained in the position required for its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, component or device to perform its function(s) are also capable of performing their related support function(s).
OPERATING LIMITS	OPERATING LIMITS are defined as the TRANSIENT LIMITS minus an allowance for normal process fluctuations.
PARTIAL ROD DEMAND POSITION	The PARTIAL ROD DEMAND POSITION is the veeder unit readout of the gang 0 unit plus the partial rod trim.
POSITIVE REACTIVITY CHANGE	A POSITIVE REACTIVITY CHANGE is a change in reactivity that is made when the reactor contains any FISSILE MATERIAL and; 1) any component(s) containing FISSILE MATERIAL are added to the reactor during CHARGE AND DISCHARGE OPERATIONS, 2) any target(s), safety rod(s) or control rod(s) are withdrawn from the reactor during CHARGE AND DISCHARGE OPERATIONS, 3) the safety rods or control rods are withdrawn for REACTOR STARTUP or testing, 4) the moderator level is increased between 1 and 14 ft indicated level in the reactor, or 5) the concentration of gadolinium is reduced in the Process Water System following a valid SSS actuation.
RADIAL SHAPE FACTOR	The RADIAL SHAPE FACTOR is the average assembly power in gang 3 and the buckle zone, divided by the average assembly power in gangs 1 and 2.

(continued)

1.1 DEFINITIONS (continued)

REACTOR OPERATION	REACTOR OPERATION is the reactor state when the reactor contains FISSILE MATERIAL and all safety rods are fully withdrawn.
REACTOR SHUTDOWN	REACTOR SHUTDOWN is the reactor state when the reactor contains FISSILE MATERIAL and either 1) 63 or more safety rods are FULLY INSERTED for the first 4 hours following REACTOR OPERATION, and no more than three rods (total combined safety rods and control rods) not FULLY INSERTED after 4 hours or 2) the Supplementary Safety System has been actuated.
REACTOR STARTUP	REACTOR STARTUP is the reactor state when the reactor contains FISSILE MATERIAL and more than three safety rods are off their bottom limit and all control rods are FULLY INSERTED.
REACTOR SUBCYCLE	A REACTOR SUBCYCLE is a period of REACTOR OPERATION that is defined in the COR.
REACTOR SYSTEM	REACTOR SYSTEM is the 60° section of the reactor associated with each process water circulating loop and the circulating loop.
ROD MOVE	The insertion or withdrawal of a full length or partial individual control rod, independant of gang movement, by more than 1 foot.
ROD SWAP	A ROD SWAP is a change in a control rod cluster's partial rod complement after any gang's FULL ROD DEMAND POSITION is < 5000 veeder units.
ROOFTOP RATIO	ROOFTOP RATIO (RTR) is the ratio of a sensor 2 signal of an axial power monitor rod to a sensor 6 signal. It is approximately the ratio of the power 1/4 of the distance from the top to bottom of the reactor to the power 3/4 of the distance from top to bottom.
SCRAM CIRCUIT RESPONSE TIME	SCRAM CIRCUIT RESPONSE TIME is the time interval from initiation of a scram signal until the simultaneous de-energizing of all safety rod clutch relays.
SAFETY ROD INSERTION TIME	SAFETY ROD INSERTION TIME is the time interval from the simultaneous de-energizing of all safety rod clutch relays until the application of variable snubbing current to each safety rod clutch.

(continued)

ONLY

1.1 DEFINITIONS (continued)

STRONG PARTIAL ROD COMPLEMENT	The STRONG PARTIAL ROD COMPLEMENT is a control rod configuration in which the strong partial rod is at its demand position and the weak partial rod is fully withdrawn.
TECHNICAL (SAFETY) LIMITS	The TECHNICAL (SAFETY) LIMITS are the process variable limits that would not be exceeded during actual transient conditions provided the process variable initial conditions are not exceeded (see TRANSIENT LIMITS).
TILT RATIO	<p>TILT RATIO is the ratio of the average assembly ΔT of all assemblies in gang 3 and gang 2, except gang 3 corner clusters (41, 45, 49, 53, 57 and 61), of a REACTOR SYSTEM with the diametrically opposite REACTOR SYSTEM as follows:</p> <p>REACTOR SYSTEM 1/REACTOR SYSTEM 4 REACTOR SYSTEM 5/REACTOR SYSTEM 2 REACTOR SYSTEM 3/REACTOR SYSTEM 6</p>
TRANSIENT LIMITS	The TRANSIENT LIMITS are the most restrictive of the Transient Protection Limits, Confinement Protection Limits, and the Emergency Cooling System Limits.
VOID CLUSTER	The VOID CLUSTER is a cluster that has all rods fully withdrawn.
VOID PARTIAL ROD COMPLEMENT	A VOID PARTIAL ROD COMPLEMENT is a control rod configuration in which both partial rods in a cluster are fully withdrawn.
WEAK PARTIAL ROD COMPLEMENT	The WEAK PARTIAL ROD COMPLEMENT is a control rod configuration in which the weak partial rod is at its demand position and the strong partial rod is fully withdrawn.

EXPLANATION ONLY

1.2 LOGICAL CONNECTORS

PURPOSE

Logical connectors are used in Technical Specifications to discriminate between, and yet connect, discrete lists of items, (namely, Conditions, Required Actions, Completion Times, Surveillances and Frequencies). The only logical connectors which appear in Technical Specifications are AND and OR. The physical arrangement of these connectors constitute logical conventions with specific meaning.

The purpose of this section is to explain the intended meaning of logical connectors and provide specific examples.

LOGICAL CONNECTORS

The use of the logical connector AND in the Technical Specifications means that each item of a list linked by AND must be satisfied. The use of the logical connector OR means that either one or the other of the alternate choices listed must be satisfied.

Left justified logical connectors must be evaluated first. Nested or indented logical connectors must be evaluated in order, left to right, after left justified logical connectors are evaluated.

In addition to the left to right position of the logical connectors, the alphanumeric designators used to identify Required Actions provide guidance in taking Required Actions.

The letter in each designator identifies Required Actions associated with a given Condition. The first number after the letter enumerates the Required Actions for the Condition. Condition A has Required Actions A.1, A.2, etc. When a Required Action consists of sub-actions, linked by nested logical connectors, an additional number is added to designate the group of sub-actions which constitute the Required Action. Such sub-actions are designated A.3.1, A.3.2, A.3.3 for example. All of these sub-actions constitute Required Action A.3 even though no A.3 exists by itself.

(continued)

1.2 LOGICAL CONNECTORS (continued)

EXAMPLE 1.2.1

This example, as illustrated below, demonstrates that for Condition A both Required Actions must be completed. In this case the logical connector AND is left justified and shows that both Required Actions A.1 and A.2 must be satisfied.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A.	A.1 Restore...	
	<u>AND</u>	
	A.2 Be in...	

(continued)

1.2 LOGICAL CONNECTORS (continued)

Example 1.2.2 This example, as illustrated below, is a more complex use of logical connectors. Required Actions A.1, A.2, and A.3.1 are alternate choices. Either may be chosen. If A.3.1 is chosen, an additional requirement, indicated by the nested (indented) logical connector AND, is imposed. This additional requirement is met by choosing A.3.2.1 or A.3.2.2. The nested position of the logical connector OR indicates that A.3.2.1 and A.3.2.2 are alternate and equal choices, one (but not both) of which must be performed.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A.	A.1 Restore...	
	<u>OR</u>	
	A.2 Align...	
	<u>OR</u>	
	A.3.1 Verify...	
	<u>AND</u>	
	A.3.2.1 Reduce...	
	<u>OR</u>	
	A.3.2.2 Reduce	

1.3 COMPLETION TIMES

PURPOSE

When a Limiting Condition for Operation (LCO) or Safety Limit is not met, Technical Specifications identify Required Actions which must be completed within specified times. An understanding of the correct meaning of Completion Times is necessary for compliance with the requirements of the Technical Specifications.

The purpose of this section is to discuss the use of Completion Times, and explain, by example, their proper meaning.

COMPLETION TIME

The Completion Time is the amount of time allowed to complete a Required Action and is referenced to the time it is discovered that an ACTIONS Condition has been entered, unless otherwise specified. Required Actions must be completed prior to the expiration of the specified Completion Time.

An ACTIONS Condition remains in effect and the Required Actions must be met until the Condition is no longer applicable or the Completion Time expires. When in more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the allowed Completion Time associated with that Condition. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of entry into the Condition. In some cases, separate Completion Times are tracked for the same Condition when entered more than once, starting from the time individual components are declared inoperable.

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

(continued)

1.3 COMPLETION TIMES

COMPLETION
TIME
(continued)

EXAMPLE 1.3.1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Actions and associated Completion Times not met.	B.1 Be in REACTOR SHUTDOWN.	90 minutes
	<u>AND</u> B.2 Restore cooling water sources.	4 hours

When a Required Action is composed of two or more requirements, they are linked by the logical connectors AND or OR as in Example 1.3.1, and each requirement has its own separate Completion Time. Each Completion Time is referenced to the time that the ACTIONS Condition was entered.

The requirement of Condition B in the example is to be in REACTOR SHUTDOWN in 90 minutes AND restore cooling water sources in 4 hours. A total of 90 minutes is allowed to reach REACTOR SHUTDOWN and a total of 4 hours (not 5.5 hours) is allowed to restore cooling water sources from the time that the ACTIONS Condition was entered. If REACTOR SHUTDOWN is reached in 60 minutes, the time allowed to restore cooling water sources is the next 3 hours because the total time to restore cooling water sources is not reduced from the allowable limit of 4 hours.

(continued)

1.3 COMPLETION TIMES

COMPLETION
TIME
(continued)

EXAMPLE 1.3.2

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 REACTOR SHUTDOWN.	90 minutes

In Example 1.3.2, when a pump is declared inoperable, Condition A is entered. If the Completion Time of 7 days expires before the pump is restored to OPERABLE status, Condition B is entered and the Completion Time clock for Required Action B.1 starts. If the inoperable pump is restored to OPERABLE status after the shutdown Condition B is entered, the Required Action of Condition B may be terminated.

When two pumps become inoperable, LCO 3.0.3 is entered, since the ACTIONS Conditions of this example do not include a Condition for two inoperable pumps.

If one of the two inoperable pumps, in example 1.3.2, is restored to OPERABLE status and the Completion Time for Condition A has not expired, the LCO 3.0.3 shutdown may be terminated and operation continued in Condition A. Thus, if all but one pump is returned to OPERABLE status, the original Completion Time for the remaining inoperable pump is not reset, but continues from the time that Condition A was initially entered.

If, while in LCO 3.0.3, one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in Condition B, providing the Completion Time for Condition B has not expired. The remaining Completion Time in Condition B is tracked from the time the Condition A Completion Time expired.

(continued)

1.3 COMPLETION TIMES

COMPLETION
TIME
(continued)

EXAMPLE 1.3.3

ACTIONS		
CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Two pumps inoperable.	B.1 Restore one pump to OPERABLE status.	72 hours
C. Required Actions and associated Completion Times not met.	C.1 Be in REACTOR SHUTDOWN.	90 minutes

When the ACTIONS Condition changes, a new Completion Time is established for the new ACTIONS Condition, but the Completion Time for the original Condition still exists. In example 1.3.3, when the Condition changes from a single inoperable pump to two inoperable pumps (Condition A and Condition B are applicable at the same time), the Completion Time for one inoperable pump continues and a separate Completion Time is established for the two pump inoperable Condition.

If the Required Action for the second Condition (B.1) is completed, and the Completion Time for the original Condition's Required Action (A.1) has not expired, the unit is still in the original Condition, with a Completion Time measured from the original time of entry into Condition A. This does not reset the time that Condition A was entered, and Required Action A.1 must still be met within its original Completion Time.

(continued)

1.3 COMPLETION TIMES

**COMPLETION
TIME**
(continued)

To illustrate example 1.3.3:

When pump #1 is declared inoperable, Condition A is entered. The Condition A clock starts and, the Completion Time is 7 days.

If pump #1 is not restored to OPERABLE status within 7 days, Condition C is entered.

If pump #2 is declared inoperable 1 day after entering Condition A, Condition B is also entered. All the Required Actions and the most restrictive Completion Time of both Conditions must be met.

The Condition B Completion Time expires in 3 days (4 days after entry into Condition A).

The Condition A Completion Time clock is still tracked, however, the more restrictive Completion Time clock of Condition B will cause Condition C to be entered prior to the expiration of the Condition A Completion Time.

If one pump is restored 3 days after entering Condition A, Condition B is exited. The Condition A Completion Time is not reset and only 4 days remain of the original 7 day Completion Time to restore the last pump.

(continued)

INFORMATION ONLY

1.3 COMPLETION TIMES

COMPLETION TIME (continued)

In example 1.3.3, if the Completion Time for the original Condition A (one pump inoperable) expires while still in the new Condition B (two pumps inoperable), then the Required Action associated with the original Condition is incomplete.

To illustrate:

When pump #1 is declared inoperable Condition A is entered. The Condition A clock starts and, the Completion Time is 7 days.

If pump #2 is declared inoperable 5 days after entering Condition A, Condition B is also entered. All the Required Actions and the most restrictive Completion Time of both Conditions must be met. The Condition B Completion Time expires in 3 days (8 days after entering Condition A).

The Condition A Completion Time is still tracked and, is more restrictive than the Completion Time of Condition B causing Condition C to be entered prior to the expiration of the Condition B Completion Time. Therefore, only 2 days, not 3 days, remain to restore both inoperable pumps.

If 1 pump is restored 7 days after entering Condition A, and the second pump remains inoperable, the Condition A Completion Time has expired and, there is no time remaining to restore the second pump. Therefore, the Required Action and associated Completion Time of Condition A is not met and, Condition C is entered.

(continued)

INFORMATION ONLY

Completion Times
1.3

1.3 COMPLETION TIMES

COMPLETION
TIME
(continued)

EXAMPLE 1.3.4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable due to excessive leakage.	A.1 Restore valves to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in REACTOR SHUTDOWN.	90 minutes

In Example 1.3.4 Condition A specifies "One or more valves inoperable...". Condition A can include more than one inoperable valve at a time, however, 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 the Condition and is not tracked on a component basis. Subsequent valves declared inoperable, while Condition A is still in effect for the first valve, do not cause separate Completion Times to be tracked. If the Completion Time of 4 hours (as measured from initial entry into Condition A) expires while one or more valves are still inoperable, Condition B is entered.

(continued)

1.3 COMPLETION TIMES

COMPLETION
TIME
(continued)

EXAMPLE 1.3.5

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable due to excessive leakage.	A.1 Restore valves to OPERABLE status.	-----NOTE----- Completion Time is on a per valve basis. ----- 4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in REACTOR SHUTDOWN.	90 minutes

In Example 1.3.5, a note is included in the Completion Time column which modifies how the Completion Time is tracked. In this example, Condition A may be entered for more than one valve and Completion Times tracked on a component 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.

If the Completion Time associated with valve(s) in Condition A expires, Condition B is entered. If the valve(s) which caused entry into Condition B are restored to OPERABLE status, Condition B is exited.

(continued)

INFORMATION ONLY

1.3 COMPLETION TIMES

**COMPLETION
TIME
(continued)**

EXAMPLE 1.3.6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Less than two EP-fans OPERABLE and in service.	A.1 Complete satisfactory performance of SR 3.01.C.1.3.	15 minutes <u>AND</u> Once per 4 hours thereafter
	<u>AND</u> A.2 Verify that at least one EP-fan is OPERABLE and in service.	15 minutes
	<u>AND</u> A.3 Restore at least two EP-fans to an OPERABLE and in service status.	72 hours
B. Required Actions and associated Completion Times not met.	B.1 Be in REACTOR SHUTDOWN.	90 minutes

Example 1.3.6 Condition A contains multiple Required Actions linked with the logical connector AND, with each Required Action having a separate Completion Time. The purpose of this example is to explain restoration from Condition B back into Condition A when the Completion Time has not expired.

(continued)

1.3 COMPLETION TIMES

**COMPLETION
TIME**

(continued)

If after Condition A is entered, Required Action A.1 is not met within either the initial 15 minutes, or any subsequent 4 hour Completion Time, Condition B is entered. The Completion Time clock for Condition A does not stop after Condition B is entered, but continues to be tracked from the time Condition A was initially entered. If after Condition B is entered, Required Action A.1 is met, operation may continue in Condition A, provided the Completion Time for Required Actions A.2 and A.3 have not expired or Required Action A.2 is met and the Completion Time for Required Action A.3 has not expired.

(continued)

1.3 COMPLETION TIMES

COMPLETION
TIME
(continued)

EXAMPLE 1.3.7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One rod position indicator per gang inoperable for one or more gangs	A.1 Verify position of rods with inoperable indicators by alternate means.	Once per 8 hours
	OR A.2 Reduce power until all assembly temperatures are < the Power Shape Applicability Limits defined in the COR.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in REACTOR SHUTDOWN.	90 minutes

Example 1.3.7 contains a Condition with two Required Actions linked with the logical connector OR. Completion of either Required Action allows operation to continue. The purpose of Example 1.3.7 is to explain restoration from Condition B or LCO 3.0.3 back into Condition A where a choice may be made between the applicable Required Actions that allow operation to continue.

(continued)

1.3 COMPLETION TIMES

COMPLETION TIME (continued)

Entry into Condition A of Example 1.3.7 offers a choice between Required Action A.1 or A.2. If Required Action A.1 is followed, and the Required Action is not met within the Completion Time initially or on any subsequent performance, Condition B is entered.

If after entry into Condition B, the Required Action A.1 is met, operation may then continue in Condition A. Alternatively, if Required Action A.2 is met while in Condition B, operation may continue in Condition A. Restoration to an ACTIONS Condition which permits continued operation is permissible if the Required Action(s) are met.

EXAMPLE 1.3.8

An exception to the Completion Time being referenced to the time an ACTION Condition is first entered is noted by the use of "thereafter" following the Completion Time. In this instance, the Completion Time might read "24 hours AND once per 12 hours thereafter." The 24 hour Completion time begins at the time the Condition was first entered; the "and once Per 12 hours thereafter" begins after the Required Action was initially satisfied. This requires the initiation of a periodic re-performance of the Required Action after the successful completion of the initial performance.

EXAMPLE 1.3.9

In certain cases a precise time span is not used for the Completion Time. The Completion Time will be expressed by the term IMMEDIATELY which is defined in section 1.1. This term is used when assigning a specific time span would be inappropriate.

1.4 FREQUENCY

PURPOSE

Each Surveillance Requirement has a specified Frequency in which the Surveillance must be met to support meeting the associated LCO. An understanding of the correct application of this Frequency is necessary for compliance with the Surveillance Requirements.

This section discusses the proper use and application of the Frequency.

FREQUENCY

Examples are provided in Table 1.4-1 and discussed below.

Example SR 1.4.1

Example SR 1.4.1 contains the type of simple Frequency most often encountered throughout the Technical Specifications. It specifies a time interval during which the associated Surveillance must be performed at least one time. Completion of this Surveillance initiates the subsequent interval. Although the Frequency is stated as 2 hours, an extension of the time interval is allowed by SR 3.0.2. If the Surveillance is not performed within the specified Frequency (per SR 3.0.2) SR 3.0.3 becomes applicable. The Surveillance must be current (last performance still within the provisions of SR 3.0.2) prior to entry into the specified condition stated in the Applicability of the associated LCO. Failure to do so would result in a violation of SR 3.0.4 unless otherwise stated in the individual Surveillance.

Example SR 1.4.2

Example SR 1.4.2 is a Surveillance with compound Frequency requirements which include a conditional event Frequency (Prior to startup) followed by a simple Frequency (30 days). The logical connector "AND" requires both Frequencies to be met. Sufficient guidance is provided with the conditional event Frequency to determine the time interval within which the Surveillance must be performed. Since the conditional event Frequency in this example is performed only once ("prior to" the event) the Frequency extension allowance of SR 3.0.2 does not apply. The simple Frequency of 30 days is then applied as described in Example SR 1.4.1.

(continued)

INFORMATION ONLY

1.4 FREQUENCY (continued)

Example SR 1.4.3 Example SR 1.4.3 is a Surveillance with two conditions linked by the logical connector OR. Each condition has its own Frequency. The Frequency used in performing the Surveillance depends on whether the specified equipment is OPERABLE or inoperable. The appropriate Frequency interval starts at the point in time at which the condition specified in the Surveillance changes. Once the specified equipment is declared inoperable, the Surveillance must be performed within the appropriate Frequency (30 minutes plus any extension allowance of SR 3.0.2). However, in no case shall the performance of the Surveillance exceed the specified Frequency for the OPERABLE equipment (2 hours plus the SR 3.0.2 allowance).

(continued)

INFORMATION ONLY

Frequency
1.4

1.4 FREQUENCY (continued)

Table 1.4-1

FREQUENCY EXAMPLES

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 1.4.1	Verify that reactor tank is full and overflowing.	2 hours
SR 1.4.2	Verify OPERABILITY of DENY, PERMIT, REQUEST, and TAKE CONTROL functions.	Prior to startup <u>AND</u> 30 days
SR 1.4.3	Verify valves [REDACTED] closed position lights are on.	30 minutes with the Essential Equipment Monitor and/or control computer inoperable <u>OR</u> 2 hours with the Essential Equipment Monitor and control computer OPERABLE

2.0 SAFETY LIMITS (Thermal-Hydraulic Limits)

2.1 SAFETY LIMITS

2.1.1 Assembly Effluent Temperatures - TECHNICAL (SAFETY) LIMITS

In REACTOR OPERATION, the average effluent temperature of any assembly shall be $\leq$ the TECHNICAL (SAFETY) LIMITS defined in the COR.

2.1.2 Normal Reactor BURNOUT RISK - TECHNICAL (SAFETY) LIMITS

In REACTOR OPERATION when reactor power is $\geq$ the limit defined in the COR for BOR applicability, the Normal Reactor BURNOUT RISK (Normal BOR) shall be $\leq$ the TECHNICAL (SAFETY) LIMITS defined in the COR.

2.1.3 Reactor Effluent Temperature - TECHNICAL (SAFETY) LIMITS

In REACTOR OPERATION, the reactor effluent temperatures shall be $\leq$ the TECHNICAL (SAFETY) LIMITS defined in the COR.

2.2 SAFETY LIMIT VIOLATION

2.2.1 If SL 2.1.1, 2.1.2, or 2.1.3 is violated, be in REACTOR SHUTDOWN IMMEDIATELY.

2.2.2 Follow the requirements of Administrative Controls Section 5.5.3.2.

CROSS REFERENCES

TITLE	NUMBER
Assembly Effluent Temperatures - TRANSIENT LIMITS	3.1.5
Transient Reactor BURNOUT RISK - TRANSIENT LIMITS	3.1.6
Reactor Effluent Temperatures - TRANSIENT LIMITS	3.1.7

3.0 APPLICABILITY

LIMITING CONDITIONS FOR OPERATION (LCO)

LCO 3.0.1 Limiting Conditions for Operation shall be met during the Conditions specified in the Applicability, except as provided in LCO 3.0.2.

LCO 3.0.2 Upon discovery of a failure to meet a Limiting Condition for Operation, the associated ACTIONS shall be met. If the Limiting Condition for Operation is restored prior to expiration of the specified Completion Time(s), completion of the Required Action is not required, unless otherwise stated. Conditions in an LCO's ACTIONS may be concurrently applicable unless otherwise stated.

The Completion Times of Required Actions are also applicable when a system or component is removed from service intentionally. Acceptable reasons for intentionally entering an ACTIONS Condition include, but are not limited to, performance of Surveillance Requirements, preventive maintenance, corrective maintenance, or investigation of operational problems. Except for Technical Specifications 3.4.1, 3.3.7.1 and 3.8.3.2, those ACTION Conditions with associated Completion Times of IMMEDIATELY can not be intentionally entered.

LCO 3.0.3 When a Limiting Condition for Operation is not met, and the associated ACTIONS are not met, or an associated ACTION is not provided, action shall be initiated to be in REACTOR SHUTDOWN in 90 minutes.

Exceptions to this requirement are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the Limiting Condition for Operation or ACTIONS, completion of the actions required by LCO 3.0.3 are not required.

LCO 3.0.3 is applicable in REACTOR OPERATION and REACTOR STARTUP.

(continued)

3.0 APPLICABILITY

LIMITING CONDITIONS FOR OPERATION (continued)

LCO 3.0.4

When a Limiting Condition for Operation is not met, entry into the reactor Conditions specified in the Applicability shall not be made unless the associated ACTIONS permit continued operation of the reactor for an unlimited period of time. This shall not prevent changes in reactor Conditions which are required to comply with Required Actions.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. These exceptions allow REACTOR STARTUP or REACTOR OPERATION or any other Condition associated with these Reactor Conditions specified in the Applicability of the LCO to be entered while meeting the associated ACTIONS. These exceptions are ACTIONS that do not permit continued operation in the specified Condition in the Applicability for an unlimited period of time.

LCO 3.0.5

Equipment removed from service, or declared inoperable, to comply with ACTIONS, may be returned to service under administrative control to perform testing required to demonstrate OPERABILITY, or the OPERABILITY of other equipment.

LCO 3.0.6

When a support system is inoperable and an LCO for that support system is specified in the Technical Specifications, the supported system is not required to be declared inoperable solely due to support system inoperability. Only the support system LCO's ACTIONS are required to be entered. This is a clarification of the definition of OPERABILITY.

When a support system is inoperable and there is not an LCO for that support system specified in the Technical Specifications, the impact of the inoperability or degradation of the support system function on the OPERABILITY of the supported system shall be evaluated. Upon determination that the supported system is inoperable, the ACTIONS of its LCO shall apply.

3.0 APPLICABILITY

SURVEILLANCE REQUIREMENTS (SR)

SR 3.0.1 Surveillance Requirements shall be met during the Reactor Conditions or other Conditions specified in the Applicability for individual Limiting Conditions for Operation, unless otherwise stated in the Surveillance Requirement. Failure to meet a Surveillance within the specified Frequency shall constitute failure to meet the Limiting Condition for Operation. Surveillance Requirements do not have to be performed on inoperable equipment.

SR 3.0.2 The specified Frequency for each Surveillance Requirement is met if the Surveillance is performed within the maximum allowable extension specified below.

<u>Nominal Surveillance Frequency^a</u>	<u>Maximum Surveillance Interval</u>
15 years (180 months)	198 months
10 years (120 months)	132 months
5 years (60 months)	66 months
4 years (48 months)	54 months
3 years (36 months)	42 months
2 years (24 months)	30 months
18 months	24 months
1 year (12 months)	18 months
9 months	12 months
6 months (180 days)	240 days
4 months (120 days)	150 days
3 months (90 days)	120 days
2 months (60 days)	75 days
1 month (30 days)	38 days
3 weeks (21 days)	28 days
2 weeks (14 days)	18 days
1 week (7 days)	9 days
2/week	5 days
3/week	4 days
2 days	3 days
1/day	2 days
More frequent than 1/day	2 times nominal interval ^b
50 KMWD's	60 KMWD's

- The nomenclature listed in the surveillance frequency may vary from that listed in this column (e.g. Once per 30 days rather than 1 month).
- For once-per-shift frequency: This means that if a scheduled surveillance cannot be completed during a shift it must be completed on the next shift.

3.0 APPLICABILITY

SURVEILLANCE REQUIREMENTS (SR)

SR 3.0.3 When a Limiting Condition for Operation is not met due to failure to perform a Surveillance within the specified Frequency, the requirement to declare the equipment inoperable may be delayed for up to 24 hours or up to the limit of the specified surveillance interval, whichever is less, from the time it is identified that the Surveillance has not been performed to permit the completion of the Surveillance. If the Surveillance is not performed within the delay period, the Completion Time of the Required Actions begins immediately upon expiration of the delay period. When the Surveillance is performed within the delay period and the Surveillance Requirements are not met, the Completion Time of the Required Actions begins immediately upon the failure of the Surveillance.

The intent of SR 3.0.3 is to allow performance of a missed surveillance for up to the delay period giving consideration to planning and the time required to perform the surveillance, not to allow the deference of all missed surveillances for up to the delay period.

SR 3.0.4 Entry into the reactor Conditions specified in the Applicability shall not be made unless the Surveillance Requirements for the applicable LCO have been met. This provision shall not prevent changes in reactor Conditions to comply with Required Actions.

Exceptions to these requirements are stated in the individual specifications.

INFORMATION ONLY

Safety Rod Withdrawal Limit
3.1.1

3.1 REACTOR

3.1.1 Safety Rod Withdrawal Limit

LCO 3.1.1 All Safety Rods shall be fully withdrawn.

APPLICABILITY REACTOR OPERATION.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any Safety Rod is <u>not</u> fully withdrawn.	-----NOTE----- Required Action A.1 must be completed any time Condition A is entered. ----- A.1 Be in REACTOR SHUTDOWN.	90 minutes

INTENTIONALLY ONLY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	<p>-----NOTE----- SR 3.0.4 is not applicable. -----</p> <p>Verify that all safety rods are fully withdrawn.</p>	<p>Within 1 hour prior to initial positioning of control rods after all safety rods are fully withdrawn</p> <p><u>AND</u></p> <p>12 hours</p>

Cross-References

TITLE	NUMBER
Safety Rod OPERABILITY	3.1.2

INT. ONLY

Safety Rod OPERABILITY
3.1.2.

3.1 REACTOR

3.1.2 Safety Rod OPERABILITY

LCO 3.1.2 At least 64 of 66 Safety Rods shall be OPERABLE.

APPLICABILITY: REACTOR STARTUP and REACTOR OPERATION.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Only 63 of 66 safety rods are OPERABLE.	A.1 Restore at least 1 inoperable safety rod to OPERABLE status.	12 hours
B. Safety Rod Negative Pulse is less negative than -123 volts.	B.1 Lower Reactor Power to the limit defined in the COR.	1 hour
	<u>AND</u> B.2 Restore Safety Rod Negative Pulse to a value more negative than -123 volts.	12 hours
C. Required Action and associated Completion Time of Condition A or B not met. <u>OR</u> Less than 63 safety rods are OPERABLE.	C.1 Be in REACTOR SHUTDOWN.	90 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
<p>-----NOTE----- SR 3.0.4 is not applicable. -----</p>		
SR 3.1.2.1	Verify that the safety rod negative pulse voltage is more negative than -123 volts.	12 hours
<p>-----NOTE----- This surveillance can only be performed on the B motor package test stand. -----</p>		
SR 3.1.2.2	Verify that snubbing for each safety rod does not initiate until the safety rod has traveled at least 75 inches from the fully withdrawn position.	<p>60 months</p> <p><u>AND</u></p> <p>After repair, replacement, modification or adjustment that may affect the initiation of snubbing</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>-----NOTES-----</p> <ol style="list-style-type: none"> 1. SR 3.0.4 is not applicable. 2. LCO 3.10.1 is applicable during the performance of safety rod testing per SR 3.1.2.3. 3. If during the performance of the following surveillance any safety rod insertion time generated by the safety rod insertion timer is missing, then the associated rod or rods must be assumed to fail the insertion time requirement. The same criteria is applicable whether the surveillance was scheduled or occurred as a result of a scram. <p>-----</p>	
<p>SR 3.1.2.3 Verify that the SAFETY ROD INSERTION TIME is ≤ 1.050 seconds for at least 64 of 66 safety rods when tripped simultaneously.</p>	<p>Subsequent to latching the safety rods and within 7 days prior to initial positioning of control rods for REACTOR OPERATION</p> <p><u>AND</u></p> <p>Within 7 days prior to control rod withdrawal for subsequent REACTOR OPERATION if more than 2 safety rods fail the insertion time requirement</p> <p>(continued)</p>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.1.2.3 (continued)</p>	<p>during a scram or any safety rod maintenance in the actuator tower or P- Panel has been performed</p>
<p style="text-align: center;">-----NOTES-----</p> <p>1. SR 3.0.4 is not applicable.</p> <p>2. LCO 3.10.1 is applicable during the performance of safety rod testing per SR 3.1.2.4.</p> <p style="text-align: center;">-----</p> <p>SR 3.1.2.4 Verify that the total time required for at least the 64 safety rods that satisfy the requirements of SR 3.1.2.3 to reach the down limit of travel is less than 30 seconds from scram relay de-energization.</p>	<p>Subsequent to latching the safety rods and within 7 days prior to initial positioning of control rods for REACTOR OPERATION</p>

(continued)

ONLY

Safety Rod OPERABILITY
3.1.2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.1.2.5 Visually inspect the safety rod latch mechanisms at the +66' level and above from outside the pattern, for evidence of slack cables or any condition which could interfere with safety rod OPERABILITY.</p>	<p>Within 24 hours prior to control rod withdrawal for REACTOR OPERATION</p> <p><u>AND</u></p> <p>Within 12 hours following the performance of any safety rod maintenance in the actuator</p>
<p>SR 3.1.2.6 Verify at the + 66' level that the rod level indicators do not indicate deformation on any of the mechanical snubbers.</p>	<p>After performance of SR 3.1.2.3 and prior to withdrawal of safety rods for REACTOR STARTUP</p>

CROSS-REFERENCES

TITLE	NUMBER
Safety Rod Testing	3.10.1

Reactivity Requirements - Shutdown
3.1.3

3.1 REACTOR

3.1.3 Reactivity Requirements - Shutdown

LCO 3.1.3 No more than three rods (total combined safety rods and control rods) shall be not FULLY INSERTED.

APPLICABILITY: REACTOR SHUTDOWN, with shutdown time greater than 4 hours.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. More than three rods (total combined safety rods and control rods) not FULLY INSERTED.	A.1 Actuate the SSS.	IMMEDIATELY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.3.1 Verify that no more than three rods (total combined safety rods and control rods) are not FULLY INSERTED.	Within 4 hours after entering REACTOR SHUTDOWN <u>AND</u> 24 hours with the actuator latched and power available to the control and/or safety rod drive motors.

CROSS-REFERENCES - None.

INFORMATION ONLY

3.1 REACTOR

3.1.4 Reactivity Anomaly

LCO 3.1.4

Predicted and actual core reactivity characteristics shall be compared during core lifetime to ensure that the as-built reactivity characteristics of the core are within the following limits:

- a. the actual core reactivity change with burnup shall be within $\pm 1.0\%$ Δk of the predicted core reactivity change with burnup, and
- b. the actual reactivity required to compensate for temperature and neutron poisons shall be within $\pm 1.0\%$ Δk of the predicted reactivity required to compensate for temperature and neutron poisons during initial power ascension of each REACTOR SUBCYCLE,
- c. the initial xenon free critical position shall be within $\pm 1.57\%$ Δk of the predicted critical position,
- d. the MARGIN OF CONTROL shall be $> 0\%$ Δk .

APPLICABILITY: REACTOR OPERATION

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Initial xenon free critical position is not within the above limit.	A.1 Be in REACTOR SHUTDOWN	90 minutes

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Actual reactivity required to compensate for temperature and neutron poisons is not within the limit.	B.1 Evaluate the impact of the reactivity difference on the safety analysis.	7 days
C. Actual core reactivity change with burnup is not within the limit.	C.1 Evaluate the impact of the reactivity difference on the safety analysis.	7 days
D. MARGIN OF CONTROL $\leq 0\% \Delta k$.	D.1 Be in REACTOR SHUTDOWN. <u>AND</u> D.2 Initiate action to fully insert <u>all</u> safety and control rods.	90 minutes IMMEDIATELY following entry into REACTOR SHUTDOWN
E. Required Action(s) and associated Completion Time(s) of Conditions B or C not met.	E.1 Be in REACTOR SHUTDOWN. <u>AND</u> E.2 Initiate action to fully insert <u>all</u> safety and control rods.	90 minutes IMMEDIATELY following entry into REACTOR SHUTDOWN

(continued)

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	<p>-----NOTE----- SR 3.0.4 is not applicable. -----</p> <p>Verify that the actual core reactivity change with burnup is within the limit.</p>	50 KMWD's
SR 3.1.4.2	<p>-----NOTE----- SR 3.0.4 is not applicable. -----</p> <p>Verify that the predicted vs. actual core reactivity required to compensate for temperature and neutron poisons is within the limit.</p>	First power ascension of each REACTOR SUBCYCLE
SR 3.1.4.3	<p>-----NOTE----- SR 3.0.4 is not applicable. -----</p> <p>Verify that the initial xenon free critical position is within the limit.</p>	First criticality of each REACTOR SUBCYCLE

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.1.4.4</p> <p>-----NOTE----- SR 3.0.4 is not applicable. -----</p> <p>Verify that the MARGIN OF CONTROL is > 0% Δk.</p>	<p>First criticality of each REACTOR SUBCYCLE</p> <p><u>AND</u></p> <p>Every 50 KMWD's</p>

CROSS-REFERENCES

TITLE	NUMBER
Administrative Controls	5.6

Assembly Effluent Temperatures - TRANSIENT LIMITS
3.1.5

INFORMATION ONLY

3.1 REACTOR

3.1.5 Assembly Effluent Temperatures - TRANSIENT LIMITS

LCO 3.1.5 The assembly average effluent temperature of any assembly shall be $\leq$ the assembly average effluent TRANSIENT LIMITS defined in the COR.

APPLICABILITY: REACTOR OPERATION.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any assembly average effluent temperature is $>$ the TRANSIENT LIMITS.	A.1 Restore assembly average effluent temperature(s) to $\leq$ the TRANSIENT LIMITS.	30 minutes
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in REACTOR SHUTDOWN.	90 minutes

Assembly Effluent Temperatures - TRANSIENT LIMITS 3.1.5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify that the assembly average effluent temperatures are $\leq$ the assembly average effluent TRANSIENT LIMITS.	12 hours <u>AND</u> Within 1 hour after exceeding the OPERATING LIMIT for assembly average effluent temperatures by an amount greater than or equal to the margin defined in the COR
SR 3.1.5.2	Verify that the assembly average effluent temperature is $\leq$ the assembly average effluent TECHNICAL (SAFETY) LIMIT defined in the COR.	IMMEDIATELY after exceeding the assembly average effluent temperature TRANSIENT LIMIT

CROSS-REFERENCES

TITLE	NUMBER
Assembly Effluent Temperatures - TECHNICAL (SAFETY) LIMITS	2.1.1

INFORMATION ONLY

Transient Reactor BURNOUT RISK - TRANSIENT LIMITS 3.1.6

3.1 REACTOR

3.1.6 Transient Reactor BURNOUT RISK - TRANSIENT LIMITS

LCO 3.1.6 The Transient Reactor BURNOUT RISK (Transient BOR) shall be $\leq$ the TRANSIENT LIMIT defined in the COR.

APPLICABILITY: REACTOR OPERATION when reactor power is $>$ the limit defined in the COR for BOR applicability.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Transient BOR $>$ the TRANSIENT LIMIT.	A.1 Restore Transient BOR to $\leq$ the TRANSIENT LIMIT.	30 minutes
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in REACTOR SHUTDOWN.	90 minutes

INFORMATION ONLY

Transient Reactor BURNOUT RISK - TRANSIENT LIMITS
3.1.6

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	Verify that Transient BOR is $\leq$ the TRANSIENT LIMIT.	12 hours <u>AND</u> Within 4 hours after exceeding the OPERATING LIMIT for Transient BOR
SR 3.1.6.2	Verify that Normal BOR is $\leq$ the TECHNICAL (SAFETY) LIMIT.	IMMEDIATELY after exceeding the BOR OPERATING LIMIT

CROSS-REFERENCES

TITLE	NUMBER
Normal Reactor BURNOUT RISK - TECHNICAL (SAFETY) LIMITS	2.1.2

3.1 REACTOR

3.1.7 Reactor Effluent Temperatures - TRANSIENT LIMITS

LCO 3.1.7 The reactor effluent temperatures shall be $\leq$ the TRANSIENT LIMITS defined in the COR.

APPLICABILITY: REACTOR OPERATION.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor Effluent Temperatures > the TRANSIENT LIMITS.	A.1 Restore reactor effluent temperatures to $\leq$ the TRANSIENT LIMITS.	30 minutes
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in REACTOR SHUTDOWN.	90 minutes

Reactor Effluent Temperatures - TRANSIENT LIMITS
3.1.7-

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify that the reactor effluent temperature is $\leq$ the reactor effluent temperature TRANSIENT LIMIT defined in the COR.	12 hours <u>AND</u> Within 1 hour after exceeding the OPERATING LIMIT for reactor effluent temperature by an amount greater than or equal to the margin defined in the COR
SR 3.1.7.2	Verify that the reactor effluent temperature is $\leq$ the reactor effluent TECHNICAL (SAFETY) LIMIT defined in the COR.	IMMEDIATELY after exceeding the reactor effluent temperature TRANSIENT LIMIT

CROSS-REFERENCES

TITLE	NUMBER
Reactor Effluent Temperatures - TECHNICAL (SAFETY) LIMITS	2.1.3

INFORMATION ONLY

Full Length Control Rod Positioning (Radial Flux Control) 3.2.1

3.2 POWER DISTRIBUTION CONTROL

3.2.1 Full Length Control Rod Positioning (Radial Flux Control)

LCO 3.2.1

A. The RADIAL SHAPE FACTOR (RSF) shall be $\geq$ the INITIAL RSF.

OR

B. The CONTROLLING ROD(s) in all three gangs shall have the same CONTROL ROD SEQUENCE number.

OR

C. The CONTROLLING ROD(s) in any gang shall be within 100 veeder units (vu) of any other gang CONTROLLING ROD when transitioning to the next rod.

AND

D. The RSF shall be ≥ 0.76 and ≤ 0.90 when power is $\geq$ the Instrument Shape Applicability Limit (ISAL) defined in the CORE OPERATIONS REPORT (COR).

AND

E. No full rods shall be MISALIGNED when power is ≥ 75 MW.

AND

F. The FULL ROD DEMAND POSITION in every cluster shall be $\leq$ the limits defined in Table 3.2.1-1 when power is $\geq$ the Power Shape Applicability Limit (PSAL) defined in the COR.

AND

G. The FULL ROD DEMAND POSITION in every cluster in gang 1 and gang 2 shall be ≥ 100 vu.

AND

H. The FULL ROD DEMAND POSITION shall be the same for every cluster in gang 1.

AND

(continued)

Full Length Control Rod Positioning (Radial Flux Control)
3.2.1

LCO 3.2.1
(continued)

- I. The FULL ROD DEMAND POSITION shall be the same for every cluster in gang 2.

AND

- J. The FULL ROD DEMAND POSITION shall be the same for every cluster within each gang 3 group defined in Table 3.2.1-2 except for voided CORNER CLUSTERS.

AND

- K. In gang 3, the FULL ROD DEMAND POSITION of each cluster shall be within 350 vu of every other cluster except for voided CORNER CLUSTERS.

AND

- L. The average FULL ROD DEMAND POSITION of all non-voided clusters in gang 3 shall be ≥ 100 vu.

AND

- M. The TILT RATIOS shall be within the limits of Table 3.2.1-3.

AND

- N. No ROD SWAPS shall be initiated when the RSF is < 0.76 or > 0.90 , or the TILT RATIO is < 0.88 or > 1.12 .

AND

- O. Every cluster with a FULL ROD DEMAND POSITION > 350 vu shall have a partial rod inserted when power is $> \text{ISAL}$.

AND

- P. No more than six full length control rods shall be inserted as INTEGRAL CONTROL RODs with:
1. Only strong rods used as INTEGRAL CONTROL RODs,
 2. No more than two INTEGRAL CONTROL RODs inserted in any one REACTOR SYSTEM,
 3. No INTEGRAL CONTROL RODs inserted in gang 1, and
 4. All INTEGRAL CONTROL RODs inserted to their FULLY INSERTED position.

AND

(continued)

Full Length Control Rod Positioning (Radial Flux Control)
3.2.1

LCO 3.2.1
(continued)

Q. The control rods in all three gangs shall have the same CONTROL ROD SEQUENCE.

AND

R. Power shall be maintained $\geq$ ISAL after power ascension to ISAL.

APPLICABILITY: REACTOR OPERATION.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The INITIAL RSF not recorded with power < ISAL.	A.1 Restore current RSF ≥ 0.76 and ≤ 0.90 .	8 hours
B. RSF is < INITIAL RSF. <u>AND</u> The CONTROLLING ROD(s) in all three gangs do not have the same CONTROL ROD SEQUENCE number. <u>AND</u> The CONTROLLING ROD(s) in any gang is not within 100 vu of all other gangs CONTROLLING ROD when transitioning to the next rod.	B.1 Restore RSF to $\geq$ INITIAL RSF. <u>OR</u> B.2 Restore the CONTROLLING ROD(s) to be the same CONTROL ROD SEQUENCE number for all three gangs. <u>OR</u> B.3 Restore the CONTROLLING ROD(s) in each gang to be within 100 vu of the other gangs CONTROLLING ROD when transitioning to the next rod.	8 hours
C. RSF is < 0.76 or > 0.90 with power $\geq$ the ISAL.	C.1 Restore the RSF to ≥ 0.76 and ≤ 0.90 .	8 hours

(continued)

INFORMATION ONLY

Full Length Control Rod Positioning (Radial Flux Control)
3.2.1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One control rod, full or partial length, is MISALIGNED while power is ≥ 75 MW.	D.1 Restore the MISALIGNED control rod to ≤ 40 vu out of alignment.	8 hours
E. Two or three control rods, full or partial length, are MISALIGNED while power is ≥ 75 MW.	E.1 Restore the MISALIGNED control rods to ≤ 40 vu out of alignment. <u>AND</u> E.2 Perform an engineering evaluation to determine the impact on the safety analysis and if continued operation is permitted.	6 hours <u>AND</u> 48 hours
F. The FULL ROD DEMAND POSITION in any cluster is $>$ the limits defined in Table 3.2.1-1.	F.1 Withdraw full rod(s) or insert additional partial rods until the FULL ROD DEMAND POSITION is $\leq$ the limits defined in Table 3.2.1-1.	8 hours
G. The FULL ROD DEMAND POSITION in any cluster in gang 1 or gang 2 is < 100 vu.	G.1 Move the full rod(s) until the FULL ROD DEMAND POSITION in every cluster in gang 1 and gang 2 is ≥ 100 vu.	8 hours

(continued)

INFORMATION ONLY

Full Length Control Rod Positioning (Radial Flux Control) 3.2.1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>H. The FULL ROD DEMAND POSITION is not the same for every gang 1 cluster.</p> <p><u>OR</u></p> <p>The FULL ROD DEMAND POSITION is not the same for every gang 2 cluster.</p>	<p>H.1 Restore the FULL ROD DEMAND POSITION to be the same for every gang 1 cluster.</p> <p><u>AND</u></p> <p>H.2 Restore the FULL ROD DEMAND POSITION to be the same for every gang 2 cluster.</p>	8 hours
I. The FULL ROD DEMAND POSITION is not the same for every non-voided cluster within each gang 3 cluster group defined in Table 3.2.1-2.	I.1 Restore the FULL ROD DEMAND POSITION for every non-voided gang 3 cluster to be the same as every other member of its gang 3 cluster group defined in Table 3.2.1-2.	8 hours
J. The FULL ROD DEMAND POSITION of each non-voided gang 3 cluster is not within 350 vu of every other non-voided gang 3 cluster.	J.1 Restore the FULL ROD DEMAND POSITION in every non-voided gang 3 cluster to within 350 vu of every other non-voided gang 3 cluster.	8 hours
K. The average FULL ROD DEMAND POSITION of all non-voided clusters in gang 3 is < 100 vu.	K.1 Restore the average FULL ROD DEMAND POSITION of all gang 3 non-voided clusters to ≥ 100 vu.	8 hours
L. Any TILT RATIO is not within the limits of Table 3.2.1-3.	L.1 Restore TILT RATIO(S) to within the limits of Table 3.2.1-3.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
M. One or two control rods, full or partial length are unlatched.	M.1 Verify TILT RATIO(S), RSF, and AXIAL POWER RATIOS are within limits. <u>AND</u> M.2 Restore each unlatched control rod, one at a time, to the FULL or PARTIAL ROD DEMAND POSITION.	Prior to performing Action M.2 8 hours
N. FULL ROD DEMAND POSITION > 350 vu in a cluster with no partial rod inserted and power > ISAL.	N.1 Decrease FULL ROD DEMAND POSITION to $\leq$ 350 vu.	8 hours
O. Control rod Fault Alarm System inoperable.	O.1 Restore Fault Alarm System to OPERABLE status.	8 hours
P. Greater than six full length control rods inserted as INTEGRAL CONTROL RODS. <u>OR</u> Any rod other than a full length strong control rod inserted as an INTEGRAL CONTROL ROD. <u>OR</u> Greater than two INTEGRAL CONTROL RODS inserted in any one REACTOR SYSTEM. (continued)	P.1 Be in REACTOR SHUTDOWN.	90 minutes

Full Length Control Rod Positioning (Radial Flux Control)
3.2.1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p><u>QR</u></p> <p>Any full length control rod inserted as an INTEGRAL CONTROL ROD in gang 1.</p>		
<p><u>QR</u></p> <p>Any INTEGRAL CONTROL ROD not inserted to its FULLY INSERTED position.</p>		
<p><u>QR</u></p> <p>Three or more control rods, full or partial length, are unlatched.</p>		
<p><u>QR</u></p> <p>Four or more control rods, full or partial length, are MISALIGNED while power is ≥ 75 MW.</p>		
<p><u>QR</u></p> <p>Power < ISAL after power ascension to ISAL.</p>		
<p><u>QR</u></p> <p>Required Action(s) and associated Completion Time(s) for Conditions A through O not met.</p>		

Full Length Control Rod Positioning (Radial Flux Control)
3.2.1

INFORMATION ONLY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Visually verify at the + 66 foot level that every full length rod is latched.	Within 4 hours after latching the actuator <u>AND</u> Prior to REACTOR STARTUP
SR 3.2.1.2	Verify that the TILT RATIOS are within the limits of Table 3.2.1-3.	Within 1 hour following a ROD SWAP or ROD MOVE <u>AND</u> 12 hours
SR 3.2.1.3	Verify that the FULL ROD DEMAND POSITION of every non-voided gang 3 cluster is within 350 vu of every other non-voided gang 3 cluster.	12 hours
SR 3.2.1.4	Verify that the FULL ROD DEMAND POSITION in every gang 1 and 2 cluster is ≥ 100 vu and $\leq$ the limits defined in Table 3.2.1-1.	Within 1 hour following a ROD MOVE <u>AND</u> 12 hours

(continued)

Full Length Control Rod Positioning (Radial Flux Control)
3.2.1

SURVEILLANCE REQUIREMENTS (continued) INFORMATION ONLY

SURVEILLANCE		FREQUENCY
SR 3.2.1.5	Verify that the average FULL ROD DEMAND POSITION of all non-voided gang 3 clusters is ≥ 100 vu and $\leq$ the limits defined in Table 3.2.1-1.	Within 1 hour following a ROD MOVE <u>AND</u> 12 hours
SR 3.2.1.6	Verify that every cluster in gang 1 has the same FULL ROD DEMAND POSITION.	Within 1 hour following a ROD MOVE <u>AND</u> 12 hours
SR 3.2.1.7	Verify that every cluster in gang 2 has the same FULL ROD DEMAND POSITION.	Within 1 hour following a ROD MOVE <u>AND</u> 12 hours
SR 3.2.1.8	Except for void clusters, verify that every cluster in each gang 3 group identified in Table 3.2.1-2 has the same FULL ROD DEMAND POSITION.	Within 1 hour following a ROD MOVE <u>AND</u> 12 hours

(continued)

Full Length Control Rod Positioning (Radial Flux Control)
3.2.1

SURVEILLANCE REQUIREMENTS (continued) INFORMATION ONLY

SURVEILLANCE		FREQUENCY
SR 3.2.1.9	Verify that all control rods in gang 1 are of the same CONTROL ROD SEQUENCE.	Within 7 days prior to REACTOR STARTUP
SR 3.2.1.10	Verify that all control rods in gang 2 are of the same CONTROL ROD SEQUENCE.	Within 7 days prior to REACTOR STARTUP
SR 3.2.1.11	Verify that all control rods in each gang 3 group identified in Table 3.2.1-2 are of the same CONTROL ROD SEQUENCE.	Within 7 days prior to REACTOR STARTUP
SR 3.2.1.12	<p>Verify that no more than six full length control rods are inserted as INTEGRAL CONTROL RODs with:</p> <ul style="list-style-type: none"> a) Only strong rods used as INTEGRAL CONTROL RODs, b) No more than two INTEGRAL CONTROL RODs inserted in any one REACTOR SYSTEM, c) No INTEGRAL CONTROL RODs inserted in gang 1, and d) All INTEGRAL CONTROL RODs inserted to their FULLY INSERTED position. 	<p>Within 1 hour following a ROD MOVE</p> <p><u>AND</u></p> <p>1 month</p>
SR 3.2.1.13	Perform a CHANNEL TEST on the control rod Fault Alarm System.	1 month

(continued)

Full Length Control Rod Positioning (Radial Flux Control)
3.2.1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
		<p>-----NOTE----- Not required when controlling RSF per SR 3.2.1.15 -----</p>
SR 3.2.1.14	Verify that gang 1, gang 2, and gang 3 are on the same CONTROLLING ROD or that the CONTROLLING ROD(S) in any gang are within 100 vu of any other gang CONTROLLING ROD when transitioning to the next rod in the CONTROL ROD SEQUENCE.	12 hours during reactor power $\leq$ ISAL
		<p>-----NOTE----- Not required if maintaining the same CONTROLLING ROD in each gang per SR 3.2.1.14 -----</p>
SR 3.2.1.15	Verify that the RSF is $\geq$ the INITIAL RSF measured at reactor power $<$ ISAL during reactor startup, and ≥ 0.76 and ≤ 0.90 with reactor power $\geq$ ISAL.	12 hours
SR 3.2.1.16	Verify every cluster with a FULL ROD DEMAND POSITION > 350 vu has a partial rod inserted.	12 hours during reactor power $>$ ISAL
SR 3.2.1.17	Verify power $\geq$ ISAL after power ascension to ISAL.	12 hours

Full Length Control Rod Positioning (Radial Flux Control)
3.2.1

CROSS-REFERENCES

INFORMATION ONLY

TITLE	NUMBER
Partial Length Control Rod Positioning (Axial Flux Control)	3.2.2

INFORMATION

Table 3.2.1-1

Full Rod Insertion Limits for any Cluster

When any assembly ΔT is $\geq$ the Power
Shape Applicability Limits defined in the COR

Partial Rod(s) <u>in Cluster</u>	FULL ROD DEMAND POSITION Limits <u>Maximum</u>
VOID PARTIAL ROD COMPLEMENT	350 veeder units
WEAK PARTIAL ROD COMPLEMENT	2250 veeder units
DOUBLE PARTIAL ROD COMPLEMENT	5000 veeder units

INFORMATION ONLY

Table 3.2.1-2

Definition of Gang 3 Rod Cluster Groups

<u>Group Number</u>	<u>Clusters In Group</u>
1.	40, 41, 42, 43
2.	44, 45, 46, 47
3.	48, 49, 50, 51
4.	52, 53, 54, 55
5.	56, 57, 58, 59
6.	60, 61, 38, 39

INFORMATION ONLY

Table 3.2.1-3

TILT RATIO Limits vs. FULL POWER

FULL POWER (%)	TILT RATIO
Power < ISAL	No Limit During Power Ascension
ISAL $\leq$ Power < PSAL	0.88 - 1.12
Power $\geq$ PSAL	0.97 - 1.03

3.2 POWER DISTRIBUTION CONTROL

INFORMATIONAL AREA

3.2.2 Partial Length Control Rod Positioning (Axial Flux Control)

LCO 3.2.2

-----NOTE-----
Requirement A does not apply to cluster 1.

- A. All partial control rod complements shall have 120-degree symmetry as defined in Table 3.2.2-1.

AND

- B. The PARTIAL ROD DEMAND POSITION for each cluster shall be ≥ 700 and ≤ 950 veeder units (vu), except:
- a. during a ROD SWAP, or
 - b. when the cluster has a VOID PARTIAL ROD COMPLEMENT, or
 - c. when all full length control rods are FULLY INSERTED.

AND

-----NOTE-----
Requirement C does not apply to cluster where a ROD SWAP is in progress.

- C. The partial rod trim for each cluster that does not have a VOID PARTIAL ROD COMPLEMENT shall be the corrected trim value obtained from the data and algorithm provided in Table 3.2.2-2.

AND

- D. No gang shall include both a VOID and a DOUBLE PARTIAL ROD COMPLEMENT.

AND

- E. The gang averaged AXIAL POWER RATIOS shall be ≥ 0.95 and ≤ 1.05 when power is $\geq$ the Power Shape Applicability Limit (PSAL) defined in the CORE OPERATIONS REPORT (COR).

AND

- F. At least 7 of 9 Axial Power Monitors (APM) shall be OPERABLE including APM #1 or APM #2 when power is $\geq$ the Instrument Shape Applicability Limit (ISAL) defined in the COR.

INFORMATION ONLY

LCO 3.2.2
(continued)

AND

- G. All OPERABLE APM's shall have AXIAL POWER RATIOS within the limits of Table 3.2.2-3.

AND

- H. No ROD SWAPS shall be initiated when any AXIAL POWER RATIO is < 0.80 or > 1.20 .

AND

- I. No partial rod shall be MISALIGNED when power is ≥ 75 MW.

AND

- J. Power shall be maintained $\geq$ ISAL after power ascension to ISAL.

AND

- K. No cluster shall contain a STRONG PARTIAL ROD COMPLEMENT.

APPLICABILITY: REACTOR OPERATION.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Partial rod complements do not have 120-degree symmetry as defined in Table 3.2.2-1.	A.1 Move partial rod(s) until 120-degree symmetry is restored.	8 hours
B. The PARTIAL ROD DEMAND POSITION is < 700 vu or > 950 vu.	B.1 Restore the PARTIAL ROD DEMAND POSITION to ≥ 700 vu and ≤ 950 vu.	8 hours

(continued)

INFORMATION ONLY

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. The partial rod trim for a cluster that does not have VOID PARTIAL ROD COMPLEMENT is not the corrected trim value obtained from the data and algorithm in Table 3.2.2-2.	C.1 Restore partial rod trim to within limits.	8 hours
D. A gang includes both a VOID and a DOUBLE PARTIAL ROD COMPLEMENT.	D.1 Complete ROD SWAP(s) to eliminate the presence of both VOID and DOUBLE PARTIAL ROD COMPLEMENTS.	8 hours
E. A gang averaged AXIAL POWER RATIO is < 0.95 or > 1.05 while power $\geq$ PSAL.	E.1 Restore gang averaged AXIAL POWER RATIO to ≥ 0.95 and ≤ 1.05	8 hours
F. Less than 7 of 9 required APM's are OPERABLE.	F.1 Restore required APM's to OPERABLE status.	8 hours

(continued)

INFORMATION ONLY

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. Any OPERABLE APM AXIAL POWER RATIO exceeds the limits of Table 3.2.2-3.	G.1 Restore AXIAL POWER RATIOS on all OPERABLE APM's to within the limits of Table 3.2.2-3.	8 hours
H. One control rod, full or partial length, is MISALIGNED while power is ≥ 75 MW.	H.1 Restore the MISALIGNED control rod to ≤ 40 vu out of alignment.	8 hours
I. Two or three control rods, full or partial length, are MISALIGNED while power is ≥ 75 MW.	I.1 Restore the MISALIGNED control rods to ≤ 40 vu out of alignment. <u>AND</u> I.2 Perform an engineering evaluation to determine the impact on the safety analysis and if continued operation is permitted.	8 hours <u>AND</u> 48 hours
J. One or two control rods, full or partial length, are unlatched.	J.1 Verify TILT RATIO(s), RSF, and AXIAL POWER RATIOS are within limits. <u>AND</u> J.2 Restore each unlatched control rod, one at a time, to the FULL or PARTIAL ROD DEMAND POSITION.	Prior to performing Action J.2 8 hours

(continued)

INFORMATION ONLY

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
K. Any cluster contains a STRONG PARTIAL ROD COMPLEMENT.	K.1 Complete ROD MOVE(s) to eliminate the presence of the STRONG PARTIAL ROD COMPLEMENT.	8 hours
L. Three or more control rods, full or partial length, are unlatched. <u>OR</u> Four or more control rods, full or partial length, are MISALIGNED. <u>OR</u> APM #1 and APM #2 inoperable. <u>OR</u> Power < ISAL after power ascension to ISAL. <u>OR</u> Required Action(s) and associated Completion Time(s) for Conditions A through K not met.	L.1 Be in REACTOR SHUTDOWN.	90 minutes

Partial Length Control Rod Positioning (Axial Flux Control)
3.2.2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Visually verify at the + 66 foot level that every partial rod is latched.	Within 4 hours after latching the actuator <u>AND</u> Prior to REACTOR STARTUP
SR 3.2.2.2	Verify that the gang averaged AXIAL POWER RATIOS are ≥ 0.95 and ≤ 1.05 .	12 hours with power $\geq$ PSAL
SR 3.2.2.3	Verify that each OPERABLE APM has an AXIAL POWER RATIO within the limits of Table 3.2.2-3.	12 hours
SR 3.2.2.4	Verify that partial rod trim for each cluster that does not have a VOID PARTIAL ROD COMPLEMENT is set to the corrected trim value obtained from the data and algorithm provided in Table 3.2.2-2.	Within 1 hour following movement of full control rods ≥ 50 vu <u>AND</u> Within 1 hour following a ROD MOVE <u>AND</u> 12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.2.5	Verify that the PARTIAL ROD DEMAND POSITION is ≥ 700 vu and ≤ 950 vu.	Within 1 hour following a ROD MOVE <u>AND</u> 12 hours
SR 3.2.2.6	Verify that the partial rod complements for the clusters in the symmetric trios shown on Table 3.2.2-1 are the same.	Within 1 hour following a ROD MOVE <u>AND</u> 7 days
SR 3.2.2.7	Verify that no control rod gang contains both a cluster with a VOID PARTIAL ROD COMPLEMENT and a cluster with a DOUBLE PARTIAL ROD COMPLEMENT.	Within 1 hour following establishment of a VOID or DOUBLE PARTIAL ROD COMPLEMENT within a gang <u>AND</u> 7 days
SR 3.2.2.8	Perform a calibration check of the required APM's.	1 month
SR 3.2.2.9	Verify power $\geq$ ISAL after power ascension to ISAL.	12 hours

(continued)

Partial Length Control Rod Positioning (Axial Flux Control)

Partial Length Control Rod Positioning (Axial Flux Control)
3.2.2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.2.10	Verify that no cluster contains a STRONG PARTIAL ROD COMPLEMENT.	Within 1 hour following a ROD MOVE <u>AND</u> 1 month

CROSS-REFERENCES

TITLE	NUMBER
Full Length Control Rod Positioning (Radial Flux Control)	3.2.1

INFORMATION ONLY

Partial Length Control Rod Positioning (Axial Flux Control)
3.2.2

Table 3.2.2-1

List of Symmetric Trios of Control Rod Clusters

GANG 1:	2,	4,	6.
	3,	5,	7.
	8,	12,	16.
	9,	13,	17.
	10,	14,	18.
	11,	15,	19.
GANG 2:	20,	26,	32.
	21,	27,	33.
	22,	28,	34.
	23,	29,	35.
	24,	30,	36.
	25,	31,	37.
GANG 3:	38,	46,	54.
	39,	47,	55.
	40,	48,	56.
	41,	49,	57.
	42,	50,	58.
	43,	51,	59.
	44,	52,	60.
	45,	53,	61.

INFORMATION ONLY

Partial Length Control Rod Positioning (Axial Flux Control)
3.2.2

Table 3.2.2-2

FULL ROD DEMAND POSITION (vu)	Relative Partial Rod Trim (vu)		FULL ROD DEMAND POSITION (vu)	Relative Partial Rod Trim (vu)	
	Weak	Double		Weak	Double
10	0	0	390	146	51
20	0	0	400	155	51
30	0	0	410	165	52
40	0	0	420	175	52
50	1	0	430	185	52
60	1	0	440	196	53
70	2	0	450	208	53
80	2	0	460	220	53
90	3	0	470	232	53
100	4	1	480	244	53
110	5	2	490	256	53
120	6	4	500	260	53
130	7	6	510	252	53
140	8	8	520	236	53
150	10	10	530	220	53
160	12	12	540	206	53
170	15	14	550	192	53
180	18	16	560	182	53
190	22	18	570	172	53
200	26	20	580	164	52
210	30	23	590	157	52
220	34	26	600	150	52
230	39	29	610	144	52
240	44	32	620	138	52
250	49	34	630	132	51
260	54	36	640	126	51
270	60	38	650	120	50
280	66	40	660	114	50
290	72	42	670	108	49
300	78	44	680	102	48
310	84	45	690	96	48
320	90	46	700	90	47
330	97	47	710	84	47
340	104	48	720	79	46
350	112	49	730	74	45
360	120	50	740	69	44
370	128	50	750	63	43
380	135	51	760	58	42

Corrected Trim = (a - b) + 50 Where a = partial rod trim from table; and
b = the minimum trim for each gang

Unless a) the resultant corrected trim gives a PARTIAL ROD DEMAND POSITION
of < 700 vu, then use a corrected trim that gives a PARTIAL ROD
DEMAND POSITION equal to 700 vu; or
the resultant corrected trim gives a PARTIAL ROD DEMAND POSITION
of > 950 vu, then use a corrected trim that gives a PARTIAL ROD
DEMAND POSITION equal to 950 vu.

INFORMATION ONLY

Partial Length Control Rod Positioning (Axial Flux Control) 3.2.2

Table 3.2.2-2

FULL ROD DEMAND POSITION (vu)	Relative Partial Rod Trim (vu)		FULL ROD DEMAND POSITION (vu)	Relative Partial Rod Trim (vu)	
	Weak	Double		Weak	Double
770	54	40	1150	10	8
780	50	38	1160	12	10
790	46	35	1170	15	12
800	42	32	1180	18	14
810	37	29	1190	21	16
820	32	25	1200	24	18
830	28	21	1210	28	20
840	24	17	1220	32	23
850	20	14	1230	36	26
860	16	11	1240	41	29
870	13	8	1250	46	31
880	10	6	1260	51	33
890	8	4	1270	56	34
900	6	2	1280	62	35
910	4	1	1290	68	37
920	3	0	1300	75	38
930	2	0	1310	83	40
940	1	0	1320	91	41
950	0	0	1330	100	42
960	0	0	1340	109	43
970	0	0	1350	120	44
980	0	0	1360	132	45
990	0	0	1370	145	46
1000	0	0	1380	160	46
1010	0	0	1390	178	47
1020	0	0	1400	198	47
1030	0	0	1410	226	47
1040	0	0	1420	260	48
1050	0	0	1430	260	48
1060	0	0	1440	260	48
1070	0	0	1450	260	48
1080	1	0	1460	260	49
1090	1	0	1470	260	49
1100	2	1	1480	260	49
1110	3	2	1490	260	49
1120	4	3	1500	260	50
1130	6	4	1510	260	50
1140	8	6	1520	260	50

INFORMATION ONLY

Partial Length Control Rod Positioning (Axial Flux Control) 3.2.2

Table 3.2.2-2

FULL ROD DEMAND POSITION (vu)	Relative Partial Rod Trim (vu)		FULL ROD DEMAND POSITION (vu)	Relative Partial Rod Trim (vu)	
	Weak	Double		Weak	Double
1530	260	50	1910	2	1
1540	260	50	1920	1	0
1550	260	50	1930	0	0
1560	244	50	1940	0	0
1570	232	49	1950	0	0
1580	218	49	1960	0	0
1590	206	49	1970	0	0
1600	195	48	1980	0	0
1610	184	48	1990	0	0
1620	173	48	2000	0	0
1630	163	47	2010	0	0
1640	154	46	2020	0	0
1650	145	45	2030	0	0
1660	138	44	2040	0	0
1670	129	43	2050	0	0
1680	120	42	2060	1	0
1690	112	41	2070	2	0
1700	104	40	2080	3	0
1710	96	39	2090	4	0
1720	89	38	2100	5	1
1730	82	37	2110	7	2
1740	75	36	2120	9	3
1750	68	35	2130	11	4
1760	61	34	2140	14	6
1770	55	32	2150	17	8
1780	49	30	2160	20	11
1790	43	28	2170	24	14
1800	38	26	2180	28	16
1810	33	23	2190	32	19
1820	28	20	2200	36	22
1830	24	18	2210	41	25
1840	20	16	2220	47	28
1850	16	14	2230	54	30
1860	13	11	2240	61	33
1870	10	8	2250	69	36
1880	7	6	2260	77	39
1890	5	4	2270	87	42
1900	4	2	2280	99	45

Partial Length Control Rod Positioning (Axial Flux Control)

FULL ROD DEMAND			FULL ROD DEMAND		
POSITION (vu)	Relative Partial Rod Trim (vu)		POSITION (vu)	Relative Partial Rod Trim (vu)	
	Weak	Double		Weak	Double
2290	114	48	2670	260	65
2300	136	52	2680	260	62
2310	161	55	2690	260	59
2320	187	58	2700	260	56
2330	213	61	2710	250	54
2340	239	64	2720	217	52
2350	260	67	2730	187	50
2360	260	69	2740	163	48
2370	260	72	2750	142	46
2380	260	74	2760	124	43
2390	260	76	2770	108	40
2400	260	78	2780	94	37
2410	260	80	2790	81	34
2420	260	82	2800	70	31
2430	260	84	2810	60	28
2440	260	86	2820	51	25
2450	260	87	2830	43	22
2460	260	88	2840	37	19
2470	260	89	2850	30	16
2480	260	90	2860	24	13
2490	260	90	2870	19	10
2500	260	90	2880	15	7
2510	260	90	2890	11	4
2520	260	90	2900	8	2
2530	260	89	2910	5	1
2540	260	88	2920	4	0
2550	260	87	2930	3	0
2560	260	86	2940	3	0
2570	260	85	2950	3	0
2580	260	84	2960	3	0
2590	260	82	2970	3	0
2600	260	80	2980	3	0
2610	260	78	2990	3	0
2620	260	76	3000	3	0
2630	260	74			
2640	260	72			
2650	260	70			
2660	260	68			

INFORMATION ONLY

Partial Length Control Rod Positioning (Axial Flux Control)
3.2.2

Table 3.2.2-3

Individual APM Rod AXIAL POWER RATIO

FULL POWER (%)	AXIAL POWER RATIO
Power < ISAL	No Limit During Power Ascension
ISAL $\leq$ Power < PSAL	0.80 - 1.20
Power $\geq$ PSAL	0.90 - 1.10

INFORMATION ONLY

Scram Circuits
3.3.1

3.3 INSTRUMENTATION

3.3.1 Scram Circuits

LCO 3.3.1 The Scram Circuits shown in Table 3.3.1-1 shall be OPERABLE with ALLOWABLE VALUES and SCRAM CIRCUIT RESPONSE TIMES as defined in the Table.

-----NOTE-----
BYPASSING of Scram Circuits is permitted for up to 12 hours provided that the number of OPERABLE circuits is $\geq$ MIN. NO. CIRCUITS REQUIRED column of Table 3.3.1-1.

APPLICABILITY: REACTOR OPERATION and REACTOR STARTUP

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more scram circuit(s) are inoperable, BYPASS permitted.	A.1 BYPASS the affected circuit(s).	15 minutes
	<u>AND</u> A.2 Restore the affected circuits to OPERABLE status.	12 hours
B. One or more scram circuit(s) are inoperable, BYPASS <u>NOT</u> permitted. <u>OR</u> A scram circuit is bypassed for more than 12 hours <u>OR</u> Required Action(s) and associated Completion Times of Condition A not met.	B.1 Be in REACTOR SHUTDOWN.	90 minutes

INFORMATION ONLY

Scram Circuits
3.3.1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Verify the circuits listed in the CIRCUITS REQUIRED column of Table 3.3.1-1 are OPERABLE by performance of surveillance requirements of Table 3.3.1-2.	As specified in Table 3.3.1-2
SR 3.3.1.2	Verify that circuits listed in the MIN. NO. CIRCUITS REQUIRED column of Table 3.3.1-1 are OPERABLE by verification of the BYPASSED status of each circuit.	Within 4 hours prior to REACTOR STARTUP <u>AND</u> Every time a circuit is BYPASSED <u>AND</u> 12 hours

CROSS-REFERENCES

TITLE	NUMBER
Remote Emergency Monitoring and Control System	3.3.5
Safety Computer	3.3.6

Table 3.3.1-1
Reactor Scram Circuits

REQUIRED INSTRUMENTATION	SETPOINT ⁵	ALLOWABLE VALUE	RESPONSE TIME	CIRCUITS REQUIRED	MIN. NO. CIRCUITS REQUIRED
1. High Level Flux Monitor (HLFM)	≤106%	≤108.1%	<0.3 sec.	4	2
2. Period Monitor	≥10 sec.	≥8.48 sec	<1.0 sec.	2	2 ¹
3. Plenum Pressure Monitor	≥90% of normal pressure	≥88.3% of normal pres.	<0.3 sec.	2	1
4. Blanket Gas Pressure	≥4 psig	≥3.68 psig	<2.1 sec.	2	1 ²
5. Cooling Water Supply Header Flow (as measured by Delta P)	≥81% of normal Delta P	≥77.6% of normal Delta P	<1.5 sec.	2	Note 3
6. Individual Heat Exchanger Cooling Water Flow (as measured by Delta P)	≥81% of normal Delta P	≥75.3% of normal Delta P	<0.5 sec.	2	Note 3
7. 3 High Level Flux Monitors Bypassed	N/A	N/A	N/A	1	1

- Both Period monitors may be BYPASSED indefinitely ≥75 MWth
- Both circuits may be BYPASSED for ≤90 minutes during a CONTROLLED REACTOR SHUTDOWN
- Between circuits 5 and 6, one circuit on the Near Side and one on the Far Side must be OPERABLE
- Required to scram upon the bypassing of any combination of three HLFMs
- The Setpoint for a Scram Circuit is considered to be adjusted consistent with the Setpoint Values as stated in this Technical Specification when the "as left" setpoint is within the band allowed for calibration accuracy.

(continued)

Table 3.3.1-1 (continued)
Reactor Scram Circuits

REQUIRED INSTRUMENTATION	SETPOINT ⁵	ALLOWABLE VALUE	RESPONSE TIME	CIRCUITS REQUIRED	MIN. NO. CIRCUITS REQUIRED
8. Control Rod Coolant Supply	≥81% of normal pressure	≥60.5% of normal pres.	N/A	1	0
9. Moderator Level	≤12 inches below normal	≤12.5 inches below normal	<2.5 sec.	1	0
10. Process Water Pump AC Motor Power Failure	≥70 VAC	≥64.6 VAC	<2.0 sec.	6	6
11. Control System Power Monitor	Loss of Power	N/A	N/A	1	1
12. Seismometer (High) (Low)	≤2.6 mm ≤0.3 mm	N/A N/A	N/A	1	Note 6
13. Manual Scram	N/A	N/A	N/A	1	1

5. The Setpoint for a Scram Circuit is considered to be adjusted consistent with the Setpoint Values as stated in this Technical Specification when the "as left" setpoint is within the band allowed for calibration accuracy.

6. BYPASS permitted only with the Galvanometer OPERABLE.

2

Table 3.3.1-2

Scram Circuit Surveillance Testing Requirements

REQUIRED INSTRUMENTATION	CHANNEL CHECK	CHANNEL TEST (Note 3)	SCRAM CIRCUIT RESPONSE TIME TEST	CHANNEL CALIBRATION
1. High Level Flux Monitor (HLFM)	N/A	7 days	Within 7 days of each REACTOR SUBCYCLE startup	12 months (Note 3)
2. Period Monitor	N/A	Within 7 days prior to REACTOR STARTUP	Within 7 days of each REACTOR SUBCYCLE startup	12 months (Note 3)
3. Plenum Pressure Monitor	N/A	1 month <u>AND</u> prior to initial power increase above 90% of FULL POWER (Note 1)	12 months	12 months
4. Blanket Gas Pressure	N/A	1 month	12 months	12 months
5. Cooling Water Supply Header Flow (as measured by Delta P)	N/A	1 month <u>AND</u> prior to initial power increase above 90% of FULL POWER (Note 1)	12 months	12 months
6. Individual Heat Exchanger Cooling Water Flow (as measured by Delta P)	N/A	1 month <u>AND</u> prior to initial power increase above 90% of FULL POWER (Note 1)	12 months	12 months
7. 3 High Level Flux Monitors Bypassed	N/A	Within 7 days prior to REACTOR STARTUP	N/A	N/A

NOTES:

1. SR 3.0.4 is not applicable.
2. Setpoint adjustment is not required.
3. Excluding the sensor.

(continued)

Table 3.3.1-2 (continued)
- Scram Circuit Surveillance Testing Requirements

REQUIRED INSTRUMENTATION	CHANNEL CHECK	CHANNEL TEST (Note 3)	SCRAM CIRCUIT RESPONSE TIME TEST	CHANNEL CALIBRATION
8. Control Rod Coolant Supply	N/A	1 month AND prior to initial power increase above 90% of FULL POWER (Note 1)	N/A	12 months
9. Moderator Level	N/A	1 month	12 months	12 months
10. Process Water Pump AC Motor Power Failure	N/A	N/A	12 months	12 months
11. Control System Power Monitor	N/A	N/A	N/A	12 months
12. Seismometer (High and Low)	N/A	2 months (Note 2)	N/A	12 months (Note 3)
13. Manual Scram	N/A	12 months	N/A	N/A

NOTES:

1. SR 3.0.4 is not applicable.
2. Setpoint adjustment is not required.
3. Excluding the sensor.

Scram Circuit - Neutron Flux Protection - Shutdown
3.3.3

3.3 INSTRUMENTATION

INFORMATION ONLY

3.3.3 Scram Circuit - Neutron Flux Protection - Shutdown

LCO 3.3.3 The following neutron flux protection Scram Circuits shall be OPERABLE:

- a. Two High Level Flux Monitor (HLFM) Scram Circuits, and
- b. Two period monitor Scram Circuits.
- c. Manual Scram.

APPLICABILITY: During REACTOR SHUTDOWN when the actuator is latched and power is available to the control rod drive motors.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Less than two HLFM Scram Circuits OPERABLE. <u>OR</u> Less than two period monitor Scram Circuits OPERABLE. <u>OR</u> Manual Scram is inoperable.	A.1 Fully insert all safety and control rods. <u>AND</u> A.2 Deenergize the control rod drive motors or delatch all safety and control rods.	IMMEDIATELY IMMEDIATELY

Scram Circuit - Neutron Flux Protection - Shutdown
3.3.3

INFORMATION ONLY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	Verify that two HLFM Scram Circuits, two period monitor Scram Circuits and the Manual Scram are OPERABLE.	Within 4 hours prior to making the safety or control rods capable of being withdrawn

CROSS-REFERENCES - None.

3.3 INSTRUMENTATION

3.3.4.1 Automatic Incident Action (AIA)

LCO 3.3.4.1 The AIA shall be OPERABLE.

APPLICABILITY: REACTOR STARTUP, REACTOR OPERATION and REACTOR SHUTDOWN with HGAs, when 6 PWS pump AC motors are operating.

-----NOTE-----
The AIA may be inoperable for up to 1 hour at a time not to exceed a cumulative time of 24 hours per reactor-year which is the cumulative time that the reactor is critical.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One power supply inoperable.	A.1 Restore affected power supply to OPERABLE status.	72 hours
B. Any output voter inoperable in the AIA "A" or "B" action logic.	B.1 Restore affected output voter to OPERABLE status.	72 hours
C. One voting circuit per input voter chasis inoperable.	C.1 ENABLE affected voting circuit.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>D. Two voting circuits inoperable on any one input voter chasis for the AIA "A" action logic.</p> <p><u>OR</u></p> <p>Two voting circuits inoperable on any one input voter chasis for the AIA "B" action logic.</p>	<p>D.1 ENABLE one affected voting circuit.</p> <p><u>AND</u></p> <p>D.2 Restore at least two voting circuits to OPERABLE status.</p>	<p>6 hours</p> <p>72 hours</p>
<p>E. Both power supplies are inoperable.</p> <p><u>OR</u></p> <p>Three voting circuits inoperable on any one input voter chasis for AIA "A" or "B" action logic.</p> <p><u>OR</u></p> <p>Any input voter inoperable for AIA "A" or "B" action logic.</p> <p><u>OR</u></p> <p>Two voting circuits inoperable in more than one input voter chasis for AIA "A" or "B" action logic.</p> <p><u>OR</u></p> <p>Two output voters inoperable for AIA "A" action logic.</p>	<p>E.1 Be in REACTOR SHUTDOWN.</p>	<p>90 minutes</p>

continued)

INFORMATION ONLY

Automatic Incident Action
3.3.4.1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p><u>OR</u></p> <p>Two output voters inoperable for AIA "B" action logic.</p>		
F. Required Actions and associated Completion Times for Conditions A, B, C, and D not met.	F.1 Be in REACTOR SHUTDOWN.	90 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.4.1.1	Verify that power supplies 1 and 2 each feed 6V and 28V AIA distribution busses.	24 hours
SR 3.3.4.1.2	Verify that conductivity signal conditioner readings are ≥ 1350 and ≤ 1970 .	24 hours
SR 3.3.4.1.3	Verify that gauge pressure signal conditioner readings are ≥ 1100 and ≤ 1999 .	24 hours
SR 3.3.4.1.4	Verify that differential pressure signal conditioner readings are ≥ 1715 and ≤ 1999 .	24 hours
SR 3.3.4.1.5	Verify that 6V and 28V power is available to the input voters.	24 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.4.1.6	Verify that 6V and 28V power is available to the output voters.	24 hours
SR 3.3.4.1.7	Verify all logic combinations and relays for AIA-A Action and AIA-B Action.	12 months
SR 3.3.4.1.8	Verify that the AIA AC motor trip/bypass circuitry is OPERABLE.	12 months
SR 3.3.4.1.9	Verify that AIA-A Action occurs at ≥ 12.8 feet with an ALLOWABLE VALUE of ≥ 11.7 feet.	12 months
SR 3.3.4.1.10	Verify that AIA-B Action occurs at ≥ 10.8 feet with an ALLOWABLE VALUE of ≥ 9.7 feet.	12 months
SR 3.3.4.1.11	Perform a CHANNEL CALIBRATION on the AIA tank level channels.	12 months
SR 3.3.4.1.12	Verify that the AIA-A Action and AIA-B Action response times are ≤ 5 seconds from an input signal to the input voters to closure of the ECS field contacts.	12 months
SR 3.3.4.1.13	Verify that AIA-B Action is not disabled for < 10 minutes following AIA-A Action.	12 months

CROSS-REFERENCES - None.

3.3 INSTRUMENTATION

3.3.4.2 Incident Control

LCO 3.3.4.2 The Incident Control Switch and associated circuitry shall be OPERABLE.

APPLICABILITY: Whenever the reactor contains HGAs.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Incident Control Switch and associated circuitry inoperable.	A.1 Restore the Incident Control Switch and associated circuitry to OPERABLE status.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in REACTOR SHUTDOWN. <u>AND</u> B.2 Initiate action to restore the Incident Control Switch and associated circuitry to OPERABLE status.	90 minutes IMMEDIATELY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.4.2.1 Verify that Incident Control Switch and associated circuitry functions properly and performs all required actions.	12 months

(continued)

INFORMATION ONLY

Incident Control
3.3.4.2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.4.2.2	Verify that loss of circulation alarm system functions properly.	12 months
SR 3.3.4.2.3	Verify that rotovalve interlock panel functions properly with incident action.	12 months

CROSS-REFERENCES - None.

INFORMATION ONLY

Remote Emergency Monitoring and Control System 3.3.5

3.3 INSTRUMENTATION

3.3.5 Remote Emergency Monitoring and Control System (REMACS)

LCO 3.3.5 The REMACS shall be OPERABLE.

APPLICABILITY: REACTOR STARTUP, REACTOR OPERATION, and REACTOR SHUTDOWN whenever the reactor contains HGAs.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One slave computer inoperable.	A.1 Restore affected slave computer to OPERABLE status.	24 hours
B. Both slave computers inoperable	B.1 Restore one slave computer to OPERABLE status.	8 hours
	<u>AND</u> B.2 Restore both slave computers to OPERABLE status.	24 hours

(continued)

Remote Emergency Monitoring and Control System
3.3.5

ACTIONS (continued) INFORMATION ONLY

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action(s) and associated Completion Time(s) for Condition(s) A, or B not met.	<p>-----NOTE----- Performance of C.2 activities shall not preclude completion of actions to establish a safe conservative condition. -----</p>	
	C.1 Be in REACTOR SHUTDOWN.	90 minutes
	<u>AND</u>	
	C.2 Suspend C&D.	IMMEDIATELY
	<u>AND</u>	
	C.3 Continue Required Actions of Conditions A or B.	Until all Required Actions are completed.

Remote Emergency Monitoring and Control System
3.3.5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	Verify REMACS communication modems are OPERABLE.	12 hours
SR 3.3.5.2	Verify that Power Security Switch enables power for digital outputs.	12 hours
SR 3.3.5.3	<p>-----NOTE----- The functions included in SR 3.3.5.4 are not included in this surveillance. -----</p> <p>Verify that the slave computers provide the required process monitoring and control functions.</p>	24 hours
SR 3.3.5.4	Verify OPERABILITY of Deny, Permit, Request, and Take Control functions.	<p>Within 12 hours prior to REACTOR STARTUP</p> <p><u>AND</u></p> <p>30 days</p>

CROSS-REFERENCES - None.

INFORMATION ONLY

Safety Computer
3.3.6

3.3 INSTRUMENTATION

3.3.6 Safety Computer

LCO 3.3.6 Two safety computers shall be OPERABLE and in operation.

-----NOTES-----

During a CONTROLLED REACTOR SHUTDOWN, the flow scram protection may be disabled on both safety computers below 70% of FULL POWER for a maximum of 90 minutes.

More than 20 but less than 100 invalid temperatures are acceptable for < 5 seconds to allow for momentary noise.

APPLICABILITY: REACTOR OPERATION and REACTOR STARTUP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One safety computer is inoperable.	A.1 Restore the inoperable computer to OPERABLE status and place in operation.	24 hours
B. One on-line safety computer has > 20 invalid inputs for temperature or flow.	B.1 BYPASS the affected safety computer. <u>AND</u> B.2 Restore the inoperable safety computer to OPERABLE status.	15 minutes 24 hours

(continued)

INFORMATION ONLY

Safety Computer
3.3.6

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two monitor pin thermocouples inoperable in any assembly requiring forced cooling.	C.1 Restore at least three monitor pin thermocouples to OPERABLE status.	12 hours
D. Thirty-one or more monitor pin thermocouples inoperable in assemblies requiring forced cooling.	D.1 Restore at least 1698 monitor pin thermocouples to OPERABLE status.	12 hours
E. Three or more monitor pin thermocouples inoperable in any assembly requiring forced cooling.	E.1 Be in REACTOR SHUTDOWN.	90 minutes
F. Two safety computers inoperable.	F.1 Be in REACTOR SHUTDOWN.	IMMEDIATELY
G. Required Action(s) and associated Completion Time(s) of Conditions A, B, C or D not met.	G.1 Be in REACTOR SHUTDOWN.	90 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.6.1 Verify that each safety computer is OPERABLE and in operation.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.6.2	Verify that each assembly requiring forced cooling is monitored for temperature or flow and that only one of the variables is aborted at any given time.	12 hours
SR 3.3.6.3	Verify that the Safety Computer Assembly Flow Monitoring scram setpoints are within the limits defined in the COR.	1 month
SR 3.3.6.4	Verify that the Safety Computer Assembly Average Effluent Temperature Monitoring scram setpoints are within the limits defined in the COR.	1 month
<p align="center">-----NOTE----- The Setpoint for a Scram Circuit is considered to be adjusted consistent with the Setpoint Values as stated in this Technical Specification when the "as left" setpoint is within the band allowed for calibration accuracy. -----</p>		
SR 3.3.6.5	Verify by performance of a CHANNEL TEST that the setpoint for each of the load shedding underfrequency relays is ≥ 59 Hz with an ALLOWABLE VALUE ≥ 58.9 Hz.	3 months
SR 3.3.6.6	Verify that the ABS/SC logic functions properly.	12 months
SR 3.3.6.7	Verify that the REMACS 190 pump trip logic functions properly.	12 months

(continued)

INFORMATION ONLY

Safety Computer
3.3.6

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.6.8	Verify that either 21 bad temperature or 21 bad flow inputs will take the safety computer off-line automatically.	12 months
SR 3.3.6.9	Verify that the Safety Computer Assembly Flow Monitoring SCRAM CIRCUIT RESPONSE TIME is < 0.5 seconds.	12 months
SR 3.3.6.10	Verify that the Safety Computer Assembly Average Effluent Temperature Monitoring SCRAM CIRCUIT RESPONSE TIME is < 1.0 second.	12 months
SR 3.3.6.11	Verify that the Loss of Pumping Accident Pump Trip System will automatically trip the Cooling Water System 190 pump motors and the Process Water System Bingham pump AC motors within 11.85 seconds after the Safety Computers receive a heat exchanger delta-p scram signal.	12 months
SR 3.3.6.12	Verify that the Loss of Pumping Accident Pump Trip System will automatically trip the Cooling Water System 190 pump motors and the Process Water System Bingham pump AC motors within 43.85 seconds after REMACS receives -40' high sump level alarm signals.	12 months
SR 3.3.6.13	Verify that the Safety Computer SCRAM CIRCUIT RESPONSE TIME initiated by underfrequency is $\leq$ 1.0 second.	12 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.14	Verify that at least 3 monitor pin thermocouples per assembly and a total of 1698 monitor pin thermocouples in all assemblies requiring forced cooling are OPERABLE.	Prior to REACTOR STARTUP if the surveillance has not been done in the previous month <u>AND</u> 1 month
SR 3.3.6.15	Verify that the SCRAM CIRCUIT RESPONSE TIME when both safety computers are BYPASSED or inoperable is < 1.0 second.	12 months
SR 3.3.6.16	Verify that 12 plenum inlet RTDs per safety computer are OPERABLE.	Within 7 days prior to each REACTOR SUBCYCLE startup
SR 3.3.6.17	Verify that each assembly requiring forced cooling has an OPERABLE flow transducer.	Within 7 days prior to each REACTOR SUBCYCLE startup
SR 3.3.6.18	Verify that each safety computer correctly processes scram and bypass signals from each scram circuit.	Within 1 month prior to each REACTOR SUBCYCLE startup

(continued)

SURVEILLANCE REQUIREMENTS (continued)

<p>-----NOTE----- The Setpoint for a Scram Circuit is considered to be adjusted consistent with the Setpoint Values as stated in this Technical Specification when the "as left" setpoint is within the band allowed for calibration accuracy. -----</p>																		
SR 3.3.6.19	Verify by performance of a CHANNEL CALIBRATION that the setpoint for each of the load shedding underfrequency relays is ≥ 59 Hz with an ALLOWABLE VALUE ≥ 58.9 Hz.		12 months															
<p>-----NOTE----- The Setpoint for a Scram Circuit is considered to be adjusted consistent with the Setpoint Values as stated in this Technical Specification when the "as left" setpoint is within the band allowed for calibration accuracy. -----</p>																		
SR 3.3.6.20	Verify by performance of a CHANNEL CALIBRATION that the Setpoints for the following REMACS 190 pump trip sensors are as listed:		12 months															
	<table><tr><th>Sensor</th><th>Setpoint</th><th>ALLOWABLE VALUE</th></tr><tr><td>[REDACTED] sump level</td><td>≤ -40.0 ft</td><td>≤ -39.7 ft</td></tr><tr><td>[REDACTED] sump level</td><td>≤ -40.0 ft</td><td>≤ -39.7 ft</td></tr><tr><td>[REDACTED] sump level</td><td>≤ -39.83 ft</td><td>≤ -39.7 ft</td></tr><tr><td>-40 motor room sump level</td><td>≤ -39.5 ft</td><td>≤ -39.4 ft</td></tr></table>	Sensor	Setpoint	ALLOWABLE VALUE	[REDACTED] sump level	≤ -40.0 ft	≤ -39.7 ft	[REDACTED] sump level	≤ -40.0 ft	≤ -39.7 ft	[REDACTED] sump level	≤ -39.83 ft	≤ -39.7 ft	-40 motor room sump level	≤ -39.5 ft	≤ -39.4 ft		
Sensor	Setpoint	ALLOWABLE VALUE																
[REDACTED] sump level	≤ -40.0 ft	≤ -39.7 ft																
[REDACTED] sump level	≤ -40.0 ft	≤ -39.7 ft																
[REDACTED] sump level	≤ -39.83 ft	≤ -39.7 ft																
-40 motor room sump level	≤ -39.5 ft	≤ -39.4 ft																

CROSS-REFERENCES

TITLE	NUMBER
Scram Circuits	3.3.1
"DELETED VERSION"	

Neutron Monitoring Requirements - Shutdown (No C&D)
3.3.7.1

3.3 INSTRUMENTATION

3.3.7.1 Neutron Monitoring Requirements - Shutdown (No C&D)

LCO 3.3.7.1 One fission counter shall be OPERABLE.

-----NOTE-----
The reactor shall contain an extraneous neutron source of sufficient magnitude to produce a detectable signal for the flux monitoring to take place.

APPLICABILITY: REACTOR SHUTDOWN with no C&D in progress and the reactor contains moderator.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. No fission counter is OPERABLE.	A.1 Suspend positive reactivity additions.	IMMEDIATELY
	AND A.2 Initiate actions to restore a fission counter to OPERABLE status.	IMMEDIATELY

INFORMATION ONLY

Neutron Monitoring Requirements - Shutdown (No C&D)
3.3.7.1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.7.1.1	Perform a CHANNEL CHECK.	12 hours
SR 3.3.7.1.2	Perform a discriminator check.	Between 3 and 8 hours after entering REACTOR SHUTDOWN
SR 3.3.7.1.3	-----NOTE----- For the performance of this surveillance, a CHANNEL CALIBRATION does not include the sensor. ----- Perform a CHANNEL CALIBRATION on the required fission counter.	30 days

CROSS-REFERENCES

TITLE	NUMBER
Scram Circuit - Neutron Flux Protection - Shutdown Neutron Monitoring Requirements - C&D Operations	3.3.3 3.3.7.2

3.3 INSTRUMENTATION INFORMATION ONLY

3.3.7.2 Neutron Monitoring Requirements - C&D

LCO 3.3.7.2 Two fission counters shall be OPERABLE.

-----NOTE-----
The reactor shall contain an extraneous neutron source of sufficient magnitude to produce a detectable signal for the flux monitoring to take place.

APPLICABILITY: CHARGE AND DISCHARGE OPERATION.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Less than two fission counters are OPERABLE.	<p>-----NOTE----- Performance of A.1 activities shall not preclude completion of actions to establish a safe conservative condition. -----</p> <p>A.1 Suspend CHARGE AND DISCHARGE OPERATIONS.</p>	IMMEDIATELY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.7.2.1 Perform a CHANNEL CHECK.	12 hours

(continued)

INTENTIONALLY LEFT ONLY

Neutron Monitoring Requirements - C&D
3.3.7.2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.7.2.2	Perform a discriminator check.	Within 24 hours prior to C&D
SR 3.3.7.2.3	<p>-----NOTE----- For the performance of this surveillance, a CHANNEL CALIBRATION does not include the sensor. -----</p> <p>Perform a CHANNEL CALIBRATION on the required fission counters.</p>	30 days

CROSS-REFERENCES

TITLE	NUMBER
Scram Circuit - Neutron Flux Protection - Shutdown Neutron Monitoring Requirements - Shutdown (No C&D)	3.3.3 3.3.7.1

INFORMATION ONLY

Process Water System Leakage Detection Instrumentation 3.4.1

3.4 PROCESS WATER SYSTEM

3.4.1 Process Water System Leakage Detection Instrumentation

LCO 3.4.1 A. Two Tritium Monitoring Systems shall be OPERABLE and monitoring the +148' level ventilation exhaust air stream, one of which shall be a Berthold Tritium Monitor (BTM).

AND

B. A third Tritium Monitoring System shall be OPERABLE and monitoring any ventilation exhaust air stream in the 105 Building.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. During REACTOR OPERATION or REACTOR STARTUP and one or more of the Tritium Monitoring Systems that is monitoring the +148' level ventilation exhaust air stream is inoperable.	A.1 Monitor the +148' level ventilation exhaust air stream with two OPERABLE Tritium Monitoring Systems one of which shall be a BTM.	8 hours

(continued)

Process Water System Leakage Detection Instrumentation
INFORMATION ONLY 3.4.1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. During REACTOR OPERATION or REACTOR STARTUP and the third Tritium Monitoring System that is monitoring any of the ventilation exhaust air streams in the 105 Building is inoperable.	B.1 Verify that the +148' level ventilation exhaust stream is monitored with two OPERABLE Tritium Monitoring Systems one of which shall be a BTM.	8 hours
	<u>AND</u> B.2 Monitor any of the ventilation exhaust air streams in the 105 Building with an OPERABLE Tritium Monitoring System.	7 days
C. During REACTOR OPERATION or REACTOR STARTUP and all required Tritium Monitoring Systems are inoperable.	C.1 Monitor the +148' level ventilation exhaust air stream with two OPERABLE Tritium Monitoring Systems one of which shall be a BTM.	8 hours
	<u>AND</u> C.2 Monitor any of the ventilation exhaust air streams in the 105 Building with an OPERABLE Tritium Monitoring System.	7 days
D. During REACTOR OPERATION or REACTOR STARTUP and Required Actions and associated Completion Times of Conditions A, B, or C not met.	D.1 Be in REACTOR SHUTDOWN.	90 minutes

(continued)

**Process Water System Leakage Detection Instrumentation
3.4.1**

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Conditions A, B, or C exist at any time except REACTOR OPERATION and REACTOR STARTUP.	E.1 Initiate action to restore the affected inoperable Tritium Monitoring Instrumentation to OPERABLE status.	IMMEDIATELY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Perform a CHANNEL CHECK of the Tritium Monitoring Instrumentation.	48 hours
SR 3.4.1.2	Perform a CHANNEL TEST of the Berthold Tritium Monitoring Instrumentation.	120 days
SR 3.4.1.3	Perform a CHANNEL CALIBRATION of the Berthold Tritium Monitoring Instrumentation.	As required by the performance of SR 3.4.1.2
SR 3.4.1.4	Perform a calibration of the Stack Tritium Monitoring Instrumentation using the Berthold Tritium Monitoring Instrumentation.	120 days

Process Water System Leakage Detection Instrumentation
3.4.1

CROSS-REFERENCES

INFORMATION ONLY

TITLE	NUMBER
Process Water System Operational Leakage	3.4.2

Process Water System Operational Leakage
3.4.2

3.4 PROCESS WATER SYSTEM

INFORMATION ONLY

3.4.2 Process Water System Operational Leakage

LCO 3.4.2

A. No PRESSURE BOUNDARY LEAKAGE.

AND

B. A total cumulative UNIDENTIFIED LEAKAGE of ≤ 96 pounds in a 24 hour period.

AND

C. An instantaneous UNIDENTIFIED LEAKAGE rate of ≤ 220 pounds/day.

APPLICABILITY: REACTOR STARTUP and REACTOR OPERATION

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The cumulative UNIDENTIFIED LEAKAGE is > 96 pounds in a 24 hour period. <u>AND</u> The instantaneous UNIDENTIFIED LEAKAGE rate is ≤ 220 pounds/day.	A.1 Identify or isolate the leak such that the cumulative UNIDENTIFIED LEAKAGE is ≤ 96 pounds in a 24 hour period.	2 hours

(continued)

INFORMATION ONLY

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. Any PRESSURE BOUNDARY LEAKAGE.</p> <p><u>OR</u></p> <p>The instantaneous UNIDENTIFIED LEAKAGE rate is > 220 pounds/day.</p>	B.1 Be in REACTOR SHUTDOWN.	90 minutes
C. Required Action and associated Completion Time of Condition A not met.	C.1 Be in REACTOR SHUTDOWN.	90 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>-----NOTE----- Heat exchanger head flange seepage shall be quantified together with UNIDENTIFIED LEAKAGE. -----</p> <p>SR 3.4.2.1 Verify that the 24 hour cumulative UNIDENTIFIED LEAKAGE is $\leq$ 96 pounds.</p>	12 hours

(continued)

INFORMATION ONLY

Process Water System Operational Leakage
3.4.2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
<p>-----NOTE----- Heat exchanger head flange seepage shall be quantified together with UNIDENTIFIED LEAKAGE. -----</p>		
SR 3.4.2.2	Verify that instantaneous UNIDENTIFIED LEAKAGE rate is $\leq$ 220 pounds/day.	12 hours

CROSS-REFERENCES

TITLE	NUMBER
Process Water System Leakage Detection Instrumentation	3.4.1

INFORMATION ONLY

Process Water System - Operating
3.4.3.1

3.4 PROCESS WATER SYSTEM

3.4.3.1 Process Water System - Operating

- LCO 3.4.3.1 A. The Process Water System (PWS) rotovalves and PWS Bingham pump suction isolation valves shall be fully open and six Bingham PWS pump AC motors shall be OPERABLE and in operation.

-----NOTE-----
All six Bingham PWS pump AC motors shall be in operation within 4 hours after starting the first Bingham PWS pump AC motor when the reactor contains HGAs.

AND

- B. The PWS septifoil supply valves shall be positioned such that flow to the septifoil ring header is within the limits defined in the COR.

APPLICABILITY: REACTOR STARTUP and REACTOR OPERATION

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more AC motor control circuits are inoperable.	A.1 Restore AC motor control circuits to OPERABLE status.	2 hours
B. One or more PWS septifoil supply valve(s) not deenergized.	<p>-----NOTE----- Upon entry into Condition B, Required Action B.1 must be completed. -----</p> <p>B.1 Be in REACTOR SHUTDOWN.</p>	90 minutes

(continued)

"DELETED VERSION"

Process Water System - Operating
3.4.3.1

INFORMATION ONLY

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action(s) and associated Completion Time(s) of Condition A not met.	C.1 Be in REACTOR SHUTDOWN.	90 minutes
D. Any PWS Bingham pump AC motor is inoperable or not in operation.	D.1 Be in REACTOR SHUTDOWN.	IMMEDIATELY
E. Any PWS rotovalve or PWS pump suction isolation valve is not fully open.	E.1 Be in REACTOR SHUTDOWN.	IMMEDIATELY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.3.1.1 Verify each PWS pump operating and circulating process water.	12 hours
SR 3.4.3.1.2 Verify that the following valves are open. o Rotovalves 11A, 11B, 21A, 21B, 31A, 31B, 41A, 41B, 51A, 51B, 61A, and 61B. o PWS pump suction valves [REDACTED]	12 hours
SR 3.4.3.1.3 Verify that all Bingham Pump AC motor control circuits are energized.	12 hours

(continued)

INFORM

Process Water System - Operating
3.4.3.1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.4.3.1.4 Verify that andale strainer valves 2 and 3 are open.	Within 48 hours prior to locking the -20' and -40' Radiologically Controlled Areas.
SR 3.4.3.1.5 Verify that the PWS septifoil supply valves are positioned as established by SR 3.4.3.1.7 and deenergized.	Within 48 hours prior to locking the -20' and -40' Radiologically Controlled Areas. <u>AND</u> Prior to entry into REACTOR STARTUP after shutdown required by ACTION B.1
SR 3.4.3.1.6 Verify that the PWS septifoil supply valves are deenergized.	7 days
SR 3.4.3.1.7 Verify that the PWS septifoil supply valves are positioned such that flow to the septifoil ring header is within the limits defined in the COR.	12 months <u>AND</u> Following any changes to the septifoil ring header flow limits as defined in the COR.

INFORMATION ONLY

Process Water System - Operating
3.4.3.1

CROSS-REFERENCES

TITLE	NUMBER
Process Water System Bingham Pump DC Motors	3.4.7
Process Water System - Shutdown	3.4.3.3

3.4 PROCESS WATER SYSTEM

3.4.3.2 Process Water System - Zone Average Assembly Flow

LCO 3.4.3.2 The zone average assembly flows shall be within the limits defined in the COR.

APPLICABILITY: REACTOR STARTUP and REACTOR OPERATION

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any zone average assembly flow is not within the limit.	A.1 Reduce reactor power to Power Shape Applicability Limits defined in the COR.	IMMEDIATELY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.3.2.1 Verify that the zone average assembly flows, calculated from the reactor plenum pressure, are within the limits defined in the COR.	After establishing 6 A.C. flow prior to REACTOR STARTUP

(continued)

Process Water System - Zone Average Assembly Flow
3.4.3.2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.4.3.2.2 -----NOTE----- SR 3.0.4 is not applicable. -----</p> <p>Verify that the zone average assembly flows calculated from the reactor plenum pressure measured at the Power Shape Applicability Limits are within the limits defined in the COR.</p>	<p>Within 6 hours following initial operation at Power Shape Applicability Limits at the beginning of a REACTOR SUBCYCLE</p>
<p>SR 3.4.3.2.3 Verify that the zone average assembly flows calculated from the reactor plenum pressure measured at the Power Shape Applicability Limits are within the limits defined in the COR.</p>	<p>1 week</p>

CROSS-REFERENCES - None.

INT ONLY

3.4 PROCESS WATER SYSTEM

3.4.3.3 Process Water System (PWS) - Shutdown

LCO 3.4.3.3 Forced circulation shall be sufficient to maintain the assembly average effluent temperature $< 80^{\circ}\text{C}$ in any HGA with:

- A. Three or more PWS Bingham pumps driven by their DC motors

AND

- B. At least one associated heat exchanger with at least gravity cooling and forced process water flow.

-----NOTE-----
All six Bingham PWS pump AC motors shall be shutdown within 1 hour after shutting down the first Bingham PWS pump AC motor.

APPLICABILITY: During REACTOR SHUTDOWN with HGAs in the reactor until the time after shutdown is greater than or equal to the forced cooling time limit defined in the CORE SHUTDOWN REPORT.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more assemblies with an assembly average effluent temperature $\geq 80^{\circ}\text{C}$.	A.1 Initiate action to restore assembly average effluent temperature(s) to $< 80^{\circ}\text{C}$.	IMMEDIATELY
B. Less than three PWS Bingham pumps being driven by their DC motors.	B.1 Initiate action to place at least three PWS Bingham pump DC motors in operation.	IMMEDIATELY

(continued)

INFORMATION ONLY

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Less than one heat exchanger with at least gravity cooling and forced process water flow.	C.1 Initiate action to restore one heat exchanger with at least gravity cooling and forced process water flow.	IMMEDIATELY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.3.3.1 Verify that assembly average effluent temperatures are < 80°C.	12 hours with alarm RC-200 OPERABLE OR 1 hour if alarm RC-200 is inoperable.
SR 3.4.3.3.2 Verify three or more PWS Bingham pumps are being driven by their DC motors.	12 hours
SR 3.4.3.3.3 Verify at least one heat exchanger has at least gravity cooling and forced process water flow.	12 hours

CROSS-REFERENCES

TITLE	NUMBER
Process Water System - Bulk Moderator Temperature Limit.	3.4.4
CHARGE AND DISCHARGE OPERATIONS - Process Water to Cooling Water AT	3.9.5

Process Water System - No Forced Circulation Bulk Moderator Temperature Limit
3.4.3.4

3.4 PROCESS WATER SYSTEM INFORMATION ONLY

3.4.3.4 Process Water System - No Forced Circulation Bulk Moderator Temperature Limit

LCO 3.4.3.4 The bulk moderator temperature shall be $\leq 50^{\circ}\text{C}$ during no forced circulation of the moderator.

OR

The maximum assembly average effluent temperature in any HGA shall be $< 80^{\circ}\text{C}$ during forced circulation of the moderator.

APPLICABILITY: During REACTOR SHUTDOWN with HGAs in the reactor when the time after shutdown is greater than the forced cooling time limit defined in the CORE SHUTDOWN REPORT.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The bulk moderator temperature $> 50^{\circ}\text{C}$ during no forced circulation.	A.1 Initiate action to restore bulk moderator temperature to $\leq 50^{\circ}\text{C}$.	IMMEDIATELY
B. One or more assemblies with an assembly average effluent temperature $\geq 80^{\circ}\text{C}$ during forced circulation.	B.1 Initiate action to restore assembly average effluent temperature(s) to $< 80^{\circ}\text{C}$.	IMMEDIATELY

INFORMATION ONLY

Process Water System - No Forced Circulation Bulk Moderator Temperature Limit
3.4.3.4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.4.1	Verify that the bulk moderator temperature $\leq 50^{\circ}\text{C}$ during no forced circulation of the moderator.	12 hours with shutdown panel alarm RC-205 OPERABLE or 1 hour if the shutdown panel alarm RC-205 is inoperable.
SR 3.4.3.4.2	Verify that assembly average effluent temperatures are $< 80^{\circ}\text{C}$ during forced circulation of the moderator.	12 hours with shutdown panel alarm RC-205 OPERABLE or 1 hour if the shutdown panel alarm RC-205 is inoperable.

CROSS-REFERENCES

TITLE	NUMBER
Process Water System - Shutdown	3.4.3.3

Process Water System - Natural Circulation Reactor Level
3.4.3.5

3.4 PROCESS WATER SYSTEM

3.4.3.5 Process Water System - Natural Circulation Reactor Level

LCO 3.4.3.5 The reactor level shall be $\geq$ overflow.

APPLICABILITY: During REACTOR SHUTDOWN with HGAs in the reactor and the time after shutdown greater than the forced cooling time limit defined in the CORE SHUTDOWN REPORT.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor level < overflow.	A.1 Initiate action to restore level.	IMMEDIATELY
	<u>AND</u> A.2 Initiate action to establish forced process water circulation.	IMMEDIATELY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.3.5.1 Verify reactor level is $\geq$ overflow.	12 hours

INFORMATION ONLY

Process Water System - Natural Circulation Reactor Level
3.4.3.5

CROSS-REFERENCES

TITLE	NUMBER
Process Water System - Shutdown	3.4.3.3

INFORMATION ONLY

Process Water System - Bulk Moderator Temperature Limit
3.4.4

3.4 PROCESS WATER SYSTEM

3.4.4 Process Water System - Bulk Moderator Temperature Limits

LCO 3.4.4 Bulk moderator temperatures at core elevations +6, +3, 0, and -3 feet from the reactor midplane shall be less than the limits defined in the COR.

APPLICABILITY: REACTOR OPERATION

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any bulk moderator temperature exceeds its limit.	A.1 Restore bulk moderator temperature(s) within limits.	30 minutes
B. Required Action and associated Completion Time not met.	B.1 Be in REACTOR SHUTDOWN.	90 minutes

INTEGRATION ONLY

Process Water System - Bulk Moderator Temperature Limit
3.4.4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify that bulk moderator temperatures are within limits.	12 hours <u>AND</u> Within 1 hour after exceeding the OPERATING LIMIT for bulk moderator temperatures by an amount greater than or equal to the margin defined in the COR

CROSS-REFERENCES - None.

INFORMATION ONLY**Process Water System Activity
3.4.5.1****3.4 PROCESS WATER SYSTEM****3.4.5.1 Process Water System Activity**

LCO 3.4.5.1 The 2 hour process water beta-gamma activity shall be
 $\leq 2.0E7$ c/m/mL.

APPLICABILITY: REACTOR STARTUP AND REACTOR OPERATION

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The 2 hour process water beta-gamma activity $> 2.0E7$ c/m/mL.	A.1 Restore to within limit.	48 hours
B. Required Action for Condition A and associated Completion Time not met.	B.1 Be in REACTOR SHUTDOWN.	90 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.5.1.1 Verify that the 2 hour process water beta-gamma activity is $\leq 2.0E7$ c/m/mL.	7 days

CROSS-REFERENCES - None.

INFORMATION ONLY

Process Water System Activity Monitoring Systems 3.4.5.2

3.4 PROCESS WATER SYSTEM

3.4.5.2 Process Water System Activity Monitoring Systems

LCO 3.4.5.2 At least two of the Process Water System Activity Monitoring Systems listed below shall be OPERABLE and in operation.

- a. Fuel Failure Blanket Gas Analyzer
- b. Low Energy Gamma Monitor (LEGM)
- c. Blanket Gas Activity Monitor
- d. 10 Minute Delay Monitor

APPLICABILITY: REACTOR OPERATION when reactor power > 20% of FULL POWER.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Less than two process water system activity monitoring systems OPERABLE and in operation.	A.1.1 Obtain and analyze grab samples of the Process Water System.	12 hours
	<u>AND</u>	<u>AND</u>
	A.1.2 Restore two process water system activity monitoring systems to OPERABLE status.	every 12 hours thereafter
	<u>OR</u>	30 days
	A.2.1 Reduce reactor power to $\leq$ 20% of FULL POWER.	90 minutes

INFORMATION ONLY

Process Water System Activity Monitoring Systems
3.4.5.2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.2.1	Perform a CHANNEL CHECK.	24 hours
SR 3.4.5.2.2	Perform a CHANNEL TEST.	3 months
SR 3.4.5.2.3	Perform a CHANNEL CALIBRATION (excluding the sensor).	12 months

CROSS-REFERENCES - None.

INFORMATION ONLY

Plenum Chamber Blanket Gas
3.4.6

3.4 PROCESS WATER SYSTEM

3.4.6 Plenum Chamber Blanket Gas

LCO 3.4.6 The blanket gas pressure shall be ≤ 5.2 psig.

APPLICABILITY: REACTOR STARTUP and REACTOR OPERATION

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Blanket gas pressure not within limit.	A.1 Restore blanket gas pressure to within limit.	2 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in REACTOR SHUTDOWN.	90 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.6.1 Verify that the blanket gas pressure is ≤ 5.2 psig.	12 hours

CROSS-REFERENCES - None.

3.4 PROCESS WATER SYSTEM

3.4.7 Process Water System Bingham Pump DC Motors

LCO 3.4.7 Each of the six Process Water System Bingham Pump DC Motors shall be OPERABLE and in operation with:

- a. Power provided by the process water diesel generators adjusted to ≥ 65 kW.

AND

- b. One process water diesel generator on each side aligned for containment operation.

APPLICABILITY: REACTOR STARTUP and REACTOR OPERATION

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One online DC motor diesel generator power output < 65 kW.	A.1 Verify that the output power of five operating diesel generators is ≥ 65 kW.	15 minutes
	AND A.2 Restore the output power of 6 diesel generators to ≥ 65 kW.	8 hours
B. One DC motor is inoperable or not in operation for reasons other than those addressed in Condition A.	B.1 Restore the inoperable DC motor to OPERABLE status and place in operation.	8 hours.

(continued)

INFORMATION ONLY

Process Water System Bingham Pump DC Motors
3.4.7

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. More than one process water diesel generator on each side aligned for containment operation. <u>OR</u> Less than one process water diesel generator on each side aligned for containment operation.	C.1 Restore to only one process water diesel generator on each side aligned for containment operation.	1 hour
D. More than 1 DC motor is inoperable or not in operation. <u>OR</u> Required Actions and associated Completion Times for A, B, or C not met.	D.1 Be in REACTOR SHUTDOWN.	90 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.7.1 Verify that the fuel oil day tank contains ≥ 32 gallons for each Cat diesel generator that is driving a DC motor that is not on containment operation.	4 hours

(continued)

INFORMATION ONLYProcess Water System Bingham Pump DC Motors
3.4.7**SURVEILLANCE REQUIREMENTS (continued)**

SURVEILLANCE		FREQUENCY
SR 3.4.7.2	Verify for all 6 online DC generators that the output power is ≥ 65 kW with AC motor online.	12 hours
SR 3.4.7.3	Verify that only one process water diesel generator on each side is aligned for containment operation.	12 hours
SR 3.4.7.4	Verify that the extended operation fuel oil tanks for each 108 building contain a combined volume $\geq 15,000$ gallons.	12 hours
SR 3.4.7.5	Verify that the volume of the extended operation lube oil sump in each 108 building is ≥ 225 gallons ($\geq 32"$).	12 hours
SR 3.4.7.6	Verify that the volume of the extended operation fresh water tank in each 108 building is ≥ 175 gallons ($\geq 28"$).	12 hours

CROSS-REFERENCES - None.

INFORMATION ONLY

3.4 PROCESS WATER SYSTEM

3.4.8.1 Deuterium Concentration

LCO 3.4.8.1 The blanket gas deuterium concentration shall be $\leq 3.8\%$ by volume.

APPLICABILITY: REACTOR STARTUP and REACTOR OPERATION

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Deuterium concentration $> 3.8\%$ by volume.	A.1 Be in REACTOR SHUTDOWN.	IMMEDIATELY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.8.1.1 Verify that the deuterium concentration is $\leq 3.8\%$ by volume.	12 hours

CROSS-REFERENCES - None.

INFORMATION ONLY

Oxygen Concentration
3.4.8.2

3.4 PROCESS WATER SYSTEM

3.4.8.2 Oxygen Concentration

LCO 3.4.8.2 The blanket gas oxygen concentration shall be $\leq 21\%$ by volume.

APPLICABILITY: REACTOR STARTUP and REACTOR OPERATION

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Oxygen concentration $> 21\%$ by volume.	A.1 Be in REACTOR SHUTDOWN.	IMMEDIATELY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.8.2.1 Verify that the oxygen concentration $\leq 21\%$ by volume.	12 hours

CROSS-REFERENCES - None.

INFORMATION ONLY

Process Water Quality - pH and Conductivity
3.4.8.3

3.4 PROCESS WATER SYSTEM

3.4.8.3 Process Water Quality - pH and Conductivity

LCO 3.4.8.3 The pH of the process water shall be ≥ 4.1 and ≤ 7.2 and the conductivity of the process water shall be ≤ 5.0 micromho/cm at 25°C.

APPLICABILITY: REACTOR STARTUP and REACTOR OPERATION

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Process water pH not within limits.	A.1 Restore process water pH to within limits.	24 hours
B. Conductivity not within limit. <u>OR</u> Required Action and associated Completion Time of Condition A not met.	B.1 Be in REACTOR SHUTDOWN.	90 minutes

INFORMATION ONLY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.3.1	Verify conductivity ≤ 5.0 micromho/cm at 25°C.	12 hours
SR 3.4.8.3.2	Verify pH is ≥ 4.1 and ≤ 7.2 .	12 hours

CROSS-REFERENCES - None.

INFORMATION ONLY

3.4 PROCESS WATER SYSTEM

3.4.8.4 Process Water Quality - Impurity Limits

LCO 3.4.8.4 The concentrations of corrosive elements/compounds in the process water in the reactor tank shall be less than:

10 PPB mercury
50 PPB silver
50 PPB copper
50 PPB gold
100 PPB chloride

APPLICABILITY: When the Process Water System contains moderator.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Concentrations of mercury, silver, copper or gold in the process water not within limits.	A.1 Be in REACTOR SHUTDOWN.	90 minutes
	<u>AND</u>	
	A.2.1 Restore concentrations to within limits.	7 days
	<u>OR</u>	
	A.2.2 Perform an engineering evaluation to determine the impact of the out of limit condition.	7 days
B. Concentration of chloride in the process water not within the limit.	B.1 Be in REACTOR SHUTDOWN.	90 minutes
	<u>AND</u>	
	B.2 Restore the chloride concentration to within the limit.	7 days

INFORMATION ONLY

Process Water Quality - Impurity Limits 3.4.8.4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.4.1	Verify that the concentration of chloride in the process water in the reactor tank is < 100 PPB.	7 days
SR 3.4.8.4.2	Verify that the concentration of mercury is < 10 PPB; silver, copper, and gold is < 50 PPB in the process water in the reactor tank.	4 months

CROSS-REFERENCES - None.

"DELETED VERSION"

INFORMATION ONLY

Emergency Cooling System (ECS) - Operating
3.5.1.1

3.5 EMERGENCY COOLING SYSTEMS/EMERGENCY SHUTDOWN SYSTEMS

3.5.1.1 Emergency Cooling System (ECS) - Operating

LCO 3.5.1.1

A. The three ECS water sources including headers, valves and pump(s) as specified in Table 3.5.1.1-1 shall be OPERABLE.

AND

B. At least one ECS water source including the associated valves and headers as specified in Table 3.5.1.1-2 shall be aligned, OPERABLE and in operation.

AND

C. The borated water header shall contain a sodium polyborate solution of ≥ 0.0175 and ≤ 0.04 g/ml at $> 5^{\circ}\text{C}$ and $\leq 40^{\circ}\text{C}$.

AND

D. Four ECS addition systems including [REDACTED] series valves and [REDACTED] series valves shall be OPERABLE.

AND

E. The Rotovalue Interlock Circuit and associated rotovalves 21-A, 21-B, 51-A and 51-B shall be OPERABLE.

APPLICABILITY: REACTOR STARTUP, REACTOR OPERATION and REACTOR SHUTDOWN with HGAs, and shutdown time less than or equal to 3 hours.

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INFORMATION ONLY

Emergency Cooling System (ECS) - Operating
3.5.1.1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ECS-A pump source or ECS-B pump source is inoperable.	A.1 Verify that the other ECS pump, either A or B, the ECS Booster Pump, and all associated valves and headers specified in Table 3.5.1.1-1 are OPERABLE.	1 hour <u>AND</u> At least every 8 hours.
	<u>AND</u> A.2 Restore the inoperable ECS water source to OPERABLE status.	24 hours
B. ECS booster pump source is inoperable.	B.1 Verify that ECS water sources ECS-A and ECS-B are aligned as specified in Table 3.5.1.1-1 and that the associated pumping sources are OPERABLE.	1 hour <u>AND</u> At least every 8 hours.
	<u>AND</u> B.2 Restore the booster pump to OPERABLE status.	24 hours
C. Sodium polyborate solution concentration and/or temperature <u>not</u> within the required limits.	C.1 Restore the sodium polyborate solution concentration and/or temperature to within the required limits.	24 hours
D. Crosstie header pressure <u>not</u> within the required limits.	D.1 Restore crosstie header pressure to within the required limits.	30 minutes

(continued)

Emergency Cooling System (ECS) - Operating
3.5.1.1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One ECS addition system inoperable.	E.1 Verify that the remaining ECS addition systems are OPERABLE. <u>AND</u> E.2 Restore the inoperable ECS addition system to OPERABLE status.	1 hour <u>AND</u> At least every 8 hours. 72 hours
F. Rotovalve(s) 21-A, 21-B, 51-A or 51-B are inoperable. <u>OR</u> Rotovalve Interlock circuit for Rotovalves 21-A, 21-B, 51-A, or 51-B are inoperable.	F.1 Restore the inoperable rotovalve(s) to OPERABLE status.	72 hours
G. Required Actions of Conditions A through F not met within required Completion Times.	G.1 Be in REACTOR SHUTDOWN. <u>AND</u> G.2 Continue actions to satisfy the LCO requirements.	90 minutes First 3 hours following REACTOR SHUTDOWN

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Emergency Cooling System (ECS) - Operating
3.5.1.1

INFORMATION ONLY

Table 3.5.1.1-1

EMERGENCY COOLING SYSTEM WATER SOURCES

PUMP(S) REQUIRED	HEADER(S) REQUIRED	NORMAL VALVE ALIGNMENT	ISOLATION VALVE STATUS
ECS-A pump		-----NOTE----- may be OPEN, if OPERABLE ----- CLOSED OPEN CLOSED OPEN CLOSED CLOSED	CLOSED and OPERABLE CLOSED and OPERABLE
ECS-B pump		-----NOTE----- may be OPEN, if OPERABLE ----- CLOSED OPEN CLOSED OPEN CLOSED CLOSED	CLOSED and OPERABLE CLOSED and OPERABLE
ECS Booster Pump		-----NOTE----- may be OPEN, if OPERABLE ----- CLOSED CLOSED OPEN CLOSED CLOSED	CLOSED and OPERABLE CLOSED and OPERABLE

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Emergency Cooling System (ECS) - Operating
3.5.1.1

INF

Table 3.5.1.1-2

COOLING WATER SYSTEM HEADERS AS ECS WATER SOURCES

PUMP(S) REQUIRED	HEADER(S) REQUIRED	VALVE ALIGNMENT DURING OPERATION	ISOLATION VALVE STATUS
* 190 building pumps	pressurized to ≥ 45 and ≤ 90 psig.	OPEN OPEN CLOSED OPEN	CLOSED and OPERABLE
* 190 building pumps	pressurized to ≥ 45 and ≤ 90 psig.	OPEN OPEN	CLOSED and OPERABLE

- * At least three Building 190 CW pumps must be operating on a header to qualify that header as an ECS water source. Additional pumps are required depending on the number of Process Water heat exchangers that are in service on the header. See the following table for requirements:

Number of heat exchangers in service on a CW header	Number of Building 190 CW pumps required on the respective header
1 or 2 heat exchangers	3 CW pumps
3 heat exchangers	4 CW pumps
4, 5, or 6 heat exchangers	5 CW pumps

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Emergency Cooling System (ECS) - Operating
3.5.1.1

INFORMATION

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1.1	Verify that: A) each [REDACTED] series crosstie header isolation valve is closed and OPERABLE. B) each [REDACTED] series ECS addition valve is closed and OPERABLE. C) each [REDACTED] series ECS addition valve is closed and OPERABLE. D) all valves associated with OPERABLE ECS pumping sources are properly aligned as specified in Table 3.5.1.1-1 and Table 3.5.1.1-2.	12 hours
SR 3.5.1.1.2	Verify that the crosstie header pressure is ≥ 45 psig.	12 hours
SR 3.5.1.1.3	A) Verify that [REDACTED] and [REDACTED] header pressures are ≥ 45 psig and ≤ 90 psig. B) Verify that all (five) of the 190 building cooling water pumps associated with each CW header are operating during REACTOR STARTUP and REACTOR OPERATION.	12 hours
SR 3.5.1.1.4	Verify that the Booster Pump is in a standby condition by performance of a status check.	12 hours

(continued)

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Emergency Cooling System (ECS) - Operating
3.5.1.1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.1.5	Verify that the borated water header temperature is $> 5^{\circ}\text{C}$ and $\leq 40^{\circ}\text{C}$.	24 hours
SR 3.5.1.1.6	Verify that manual ECS block addition valves [REDACTED] are in the full open position.	Within 48 hours prior to locking the -20' and -40' radiologically controlled areas
SR 3.5.1.1.7	Manually start each ECS-A, ECS-B and Booster Pump and run for at least 10 minutes.	14 days
SR 3.5.1.1.8	Verify by lab analysis that the borated water header sodium polyborate solution concentration is ≥ 0.0175 and ≤ 0.04 g/ml.	Every 24 hours if conductivity is < 3651 or > 8210 μmhos . <u>AND</u> Within 24 hours of any addition to the borated water header. <u>AND</u> 1 month
SR 3.5.1.1.9	Verify that the borated water header filter screen bypass lines begin opening when the pressure drop across the screen is 7 to 7.5 psid based on measured torque settings.	1 month

(continued)

"DELETED VERSION"

Emergency Cooling System (ECS) - Operating
3.5.1.1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.5.1.1.10 Start and run ECS-A and ECS-B pumps from the emergency buses for at least 30 minutes and then restart and run the pumps from the containment substation for an additional 30 minutes.</p>	<p>1 year</p>
<p>SR 3.5.1.1.11 Perform an ECS functional full-flow test through the test flow header with [REDACTED] and [REDACTED] pressurized by all ten CW pumps, and verify upon actuation of Manual Incident Action that:</p> <ul style="list-style-type: none">A. The Booster Pump, ECS-A Pump and ECS-B Pump start.B. The [REDACTED] series crosstie header isolation valves open.C. The [REDACTED] series addition valves open.D. Addition valves [REDACTED] open.E. Addition valve [REDACTED] switches to automatic mode and opens upon receipt of a simulated reactor tank level of 9 feet and closes upon receipt of a simulated reactor tank level of 14 feet.F. Total flow through the test lines with valve [REDACTED] open is $\geq 13,779$ gpm and that the pressure drop across the borated water filter screen is ≤ 7 psid based on a measured torque setting.G. Total flow through the test lines with valve [REDACTED] closed is $\geq 11,945$ gpm in ≤ 25 seconds.	<p>24 months</p>

(continued)

"DELETED VERSION"

SR 3.5.1.1.12	SURVEILLANCE	FREQUENCY																																																																																				
	<p>Perform a flow test, through the test lines, of the [REDACTED] header as a single ECS source with pump and valve status as indicated below. The flow test includes five subtests (A-E) for the specified combinations of [REDACTED] valve status. Verify that the total ECS flow versus the Cross-Tie [REDACTED] header centerline pressure for each subtest lies above the curve generated by connecting the points shown below with straight lines.</p> <table border="1"> <thead> <tr> <th>XTIE Pressure (psig)</th> <th>Total ECS Flow (gpm)</th> </tr> </thead> <tbody> <tr> <td>45.83</td> <td>0</td> </tr> <tr> <td>43.13</td> <td>4165</td> </tr> <tr> <td>37.83</td> <td>7219</td> </tr> <tr> <td>33.46</td> <td>9338</td> </tr> <tr> <td>30.21</td> <td>11305</td> </tr> <tr> <td>0.0</td> <td>11305</td> </tr> </tbody> </table> <p>190 Blde Pump Status</p> <table border="1"> <tbody> <tr><td>[REDACTED]</td><td>ON</td></tr> <tr><td>[REDACTED]</td><td>ON</td></tr> <tr><td>[REDACTED]</td><td>ON</td></tr> <tr><td>[REDACTED]</td><td>ON</td></tr> <tr><td>[REDACTED]</td><td>ON</td></tr> </tbody> </table> <p>ECS Pump Status</p> <p>ECS-A OFF ECS-B OFF Booster OFF</p> <table border="1"> <thead> <tr> <th>Valve#</th> <th>Status</th> <th>Valve#</th> <th>Status</th> <th>Valve#</th> <th>Status</th> </tr> </thead> <tbody> <tr> <td>[REDACTED]</td> <td>OPEN</td> <td>[REDACTED]</td> <td>OPEN</td> <td>[REDACTED]</td> <td>OPEN</td> </tr> <tr> <td>[REDACTED]</td> <td>CLOSED</td> <td>[REDACTED]</td> <td>OPEN</td> <td>[REDACTED]</td> <td>OPEN</td> </tr> <tr> <td>[REDACTED]</td> <td>CLOSED</td> <td>[REDACTED]</td> <td>OPEN</td> <td>[REDACTED]</td> <td>OPEN</td> </tr> <tr> <td>[REDACTED]</td> <td>CLOSED</td> <td>[REDACTED]</td> <td>OPEN</td> <td>[REDACTED]</td> <td>CLOSED</td> </tr> </tbody> </table> <table border="1"> <thead> <tr> <th>Subtest</th> <th colspan="4">Valve Status</th> </tr> </thead> <tbody> <tr> <td>A</td> <td>CLOSED</td> <td>CLOSED</td> <td>CLOSED</td> <td>CLOSED</td> </tr> <tr> <td>B</td> <td>OPEN</td> <td>CLOSED</td> <td>CLOSED</td> <td>CLOSED</td> </tr> <tr> <td>C</td> <td>OPEN</td> <td>OPEN</td> <td>CLOSED</td> <td>CLOSED</td> </tr> <tr> <td>D</td> <td>OPEN</td> <td>OPEN</td> <td>OPEN</td> <td>CLOSED</td> </tr> <tr> <td>E</td> <td>OPEN</td> <td>OPEN</td> <td>OPEN</td> <td>OPEN</td> </tr> </tbody> </table>	XTIE Pressure (psig)	Total ECS Flow (gpm)	45.83	0	43.13	4165	37.83	7219	33.46	9338	30.21	11305	0.0	11305	[REDACTED]	ON	[REDACTED]	ON	[REDACTED]	ON	[REDACTED]	ON	[REDACTED]	ON	Valve#	Status	Valve#	Status	Valve#	Status	[REDACTED]	OPEN	[REDACTED]	OPEN	[REDACTED]	OPEN	[REDACTED]	CLOSED	[REDACTED]	OPEN	[REDACTED]	OPEN	[REDACTED]	CLOSED	[REDACTED]	OPEN	[REDACTED]	OPEN	[REDACTED]	CLOSED	[REDACTED]	OPEN	[REDACTED]	CLOSED	Subtest	Valve Status				A	CLOSED	CLOSED	CLOSED	CLOSED	B	OPEN	CLOSED	CLOSED	CLOSED	C	OPEN	OPEN	CLOSED	CLOSED	D	OPEN	OPEN	OPEN	CLOSED	E	OPEN	OPEN	OPEN	OPEN	24 months
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Emergency Cooling System (ECS) - Operating
3.5.1.1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY																																																																																																		
SR 3.5.1.1.13	<p>Perform a flow test, through the test lines, of the [REDACTED] header as a single ECS source with pump and valve status as indicated below. The flow test includes five subtests (A-E) for the specified combinations of [REDACTED] valve status. Verify that the total ECS flow versus the Cross-Tie [REDACTED] header centerline pressure for each subtest lies above the curve generated by connecting the points shown below with straight lines.</p> <table border="1"> <thead> <tr> <th><u>Cross-Tie Pressure</u> (psig)</th> <th><u>Total ECS Flow</u> (gpm)</th> </tr> </thead> <tbody> <tr><td>45.76</td><td>0</td></tr> <tr><td>41.41</td><td>5177</td></tr> <tr><td>36.82</td><td>8858</td></tr> <tr><td>32.11</td><td>11578</td></tr> <tr><td>26.49</td><td>14023</td></tr> <tr><td>0.0</td><td>14023</td></tr> </tbody> </table> <table border="1"> <thead> <tr> <th colspan="2"><u>120 Blade Pump Status</u></th> <th colspan="2"><u>ECS Pump Status</u></th> </tr> </thead> <tbody> <tr><td>[REDACTED]</td><td>OFF</td><td>[REDACTED]</td><td>ON</td></tr> <tr><td>[REDACTED]</td><td>OFF</td><td>[REDACTED]</td><td>ON</td></tr> <tr><td>[REDACTED]</td><td>OFF</td><td>[REDACTED]</td><td>ON</td></tr> <tr><td>[REDACTED]</td><td>OFF</td><td>[REDACTED]</td><td>ON</td></tr> <tr><td>[REDACTED]</td><td>OFF</td><td>[REDACTED]</td><td>ON</td></tr> </tbody> </table> <table border="1"> <thead> <tr> <th><u>Valves</u></th> <th><u>Status</u></th> <th><u>Valves</u></th> <th><u>Status</u></th> <th><u>Valves</u></th> <th><u>Status</u></th> </tr> </thead> <tbody> <tr><td>[REDACTED]</td><td>CLOSED</td><td>[REDACTED]</td><td>OPEN</td><td>[REDACTED]</td><td>OPEN</td></tr> <tr><td>[REDACTED]</td><td>OPEN</td><td>[REDACTED]</td><td>OPEN</td><td>[REDACTED]</td><td>OPEN</td></tr> <tr><td>[REDACTED]</td><td>CLOSED</td><td>[REDACTED]</td><td>OPEN</td><td>[REDACTED]</td><td>OPEN</td></tr> <tr><td>[REDACTED]</td><td>CLOSED</td><td>[REDACTED]</td><td>OPEN</td><td>[REDACTED]</td><td>CLOSED</td></tr> </tbody> </table> <table border="1"> <thead> <tr> <th><u>Subtest</u></th> <th colspan="4"><u>Valve Status</u></th> </tr> </thead> <tbody> <tr><td>A</td><td>CLOSED</td><td>CLOSED</td><td>CLOSED</td><td>CLOSED</td></tr> <tr><td>B</td><td>OPEN</td><td>CLOSED</td><td>CLOSED</td><td>CLOSED</td></tr> <tr><td>C</td><td>OPEN</td><td>OPEN</td><td>CLOSED</td><td>CLOSED</td></tr> <tr><td>D</td><td>OPEN</td><td>OPEN</td><td>OPEN</td><td>CLOSED</td></tr> <tr><td>E</td><td>OPEN</td><td>OPEN</td><td>OPEN</td><td>OPEN</td></tr> </tbody> </table>	<u>Cross-Tie Pressure</u> (psig)	<u>Total ECS Flow</u> (gpm)	45.76	0	41.41	5177	36.82	8858	32.11	11578	26.49	14023	0.0	14023	<u>120 Blade Pump Status</u>		<u>ECS Pump Status</u>		[REDACTED]	OFF	[REDACTED]	ON	[REDACTED]	OFF	[REDACTED]	ON	[REDACTED]	OFF	[REDACTED]	ON	[REDACTED]	OFF	[REDACTED]	ON	[REDACTED]	OFF	[REDACTED]	ON	<u>Valves</u>	<u>Status</u>	<u>Valves</u>	<u>Status</u>	<u>Valves</u>	<u>Status</u>	[REDACTED]	CLOSED	[REDACTED]	OPEN	[REDACTED]	OPEN	[REDACTED]	OPEN	[REDACTED]	OPEN	[REDACTED]	OPEN	[REDACTED]	CLOSED	[REDACTED]	OPEN	[REDACTED]	OPEN	[REDACTED]	CLOSED	[REDACTED]	OPEN	[REDACTED]	CLOSED	<u>Subtest</u>	<u>Valve Status</u>				A	CLOSED	CLOSED	CLOSED	CLOSED	B	OPEN	CLOSED	CLOSED	CLOSED	C	OPEN	OPEN	CLOSED	CLOSED	D	OPEN	OPEN	OPEN	CLOSED	E	OPEN	OPEN	OPEN	OPEN	24 months
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Emergency Cooling System (ECS) - Operating
3.5.1.1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY																																																																																																		
<div>SR 3.5.1.1.14</div> <div>Perform a flow test, through the test lines, of the Booster Pump as a single ECS source with pump and valve status as indicated below. The flow test includes five subtests (A-E) for the specified combinations of valve status. Verify that the total ECS flow versus the Cross-Tie header centerline pressure for each subtest lies above the curve generated by connecting the points shown below with straight lines.</div> <div><table><tr><th>XTIE Pressure (psig)</th><th>Total ECS Flow (gpm)</th></tr><tr><td>76.02</td><td>0</td></tr><tr><td>54.05</td><td>5916</td></tr><tr><td>40.60</td><td>9213</td></tr><tr><td>30.62</td><td>10778</td></tr><tr><td>22.99</td><td>11952</td></tr><tr><td>0.0</td><td>11952</td></tr></table><div><table><tr><th colspan="2">190 Bldg Pump Status</th><th colspan="2">ECS Pump Status</th></tr><tr><td>OFF</td><td>ON</td><td>ECS-A</td><td>OFF</td></tr><tr><td>OFF</td><td>ON</td><td>ECS-B</td><td>OFF</td></tr><tr><td>OFF</td><td>ON</td><td>Booster</td><td>ON</td></tr><tr><td>OFF</td><td>ON</td><td></td><td></td></tr><tr><td>OFF</td><td>ON</td><td></td><td></td></tr></table><table><tr><th>Valve</th><th>Status</th><th>Valve</th><th>Status</th><th>Valve</th><th>Status</th></tr><tr><td></td><td>CLOSED</td><td></td><td>OPEN</td><td></td><td>OPEN</td></tr><tr><td></td><td>CLOSED</td><td></td><td>OPEN</td><td></td><td>OPEN</td></tr><tr><td></td><td>OPEN</td><td></td><td>OPEN</td><td></td><td>OPEN</td></tr><tr><td></td><td>OPEN</td><td></td><td>OPEN</td><td></td><td>CLOSED</td></tr></table><table><tr><th>Subtest</th><th colspan="4">Valve Status</th></tr><tr><td>A</td><td>CLOSED</td><td>CLOSED</td><td>CLOSED</td><td>CLOSED</td></tr><tr><td>B</td><td>OPEN</td><td>CLOSED</td><td>CLOSED</td><td>CLOSED</td></tr><tr><td>C</td><td>OPEN</td><td>OPEN</td><td>CLOSED</td><td>CLOSED</td></tr><tr><td>D</td><td>OPEN</td><td>OPEN</td><td>OPEN</td><td>CLOSED</td></tr><tr><td>E</td><td>OPEN</td><td>OPEN</td><td>OPEN</td><td>OPEN</td></tr></table></div></div>	XTIE Pressure (psig)	Total ECS Flow (gpm)	76.02	0	54.05	5916	40.60	9213	30.62	10778	22.99	11952	0.0	11952	190 Bldg Pump Status		ECS Pump Status		OFF	ON	ECS-A	OFF	OFF	ON	ECS-B	OFF	OFF	ON	Booster	ON	OFF	ON			OFF	ON			Valve	Status	Valve	Status	Valve	Status		CLOSED		OPEN		OPEN		CLOSED		OPEN		OPEN		OPEN		OPEN		OPEN		OPEN		OPEN		CLOSED	Subtest	Valve Status				A	CLOSED	CLOSED	CLOSED	CLOSED	B	OPEN	CLOSED	CLOSED	CLOSED	C	OPEN	OPEN	CLOSED	CLOSED	D	OPEN	OPEN	OPEN	CLOSED	E	OPEN	OPEN	OPEN	OPEN	<div>24 months</div>
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SR 3.5.1.1.15	<p>Perform a flow test, through the test lines, of the ECS-A pump as a single ECS source with pump and valve status as indicated below. The flow test includes five subtests (A-E) for the specified combinations of valve status. Verify that the total ECS flow versus the Cross-Tie header centerline pressure for each subtest lies above the curve generated by connecting the points shown below with straight lines.</p> <table><thead><tr><th>XTIE Pressure (psig)</th><th>Total ECS Flow (gpm)</th></tr></thead><tbody><tr><td>68.92</td><td>0</td></tr><tr><td>56.04</td><td>5612</td></tr><tr><td>36.56</td><td>9234</td></tr><tr><td>23.55</td><td>10754</td></tr><tr><td>19.14</td><td>11295</td></tr><tr><td>0.0</td><td>11295</td></tr></tbody></table> <p>190 Blade Pump Status</p> <table><tr><td>ON</td><td>OFF</td></tr><tr><td>ON</td><td>OFF</td></tr><tr><td>ON</td><td>OFF</td></tr><tr><td>ON</td><td>OFF</td></tr><tr><td>ON</td><td>OFF</td></tr></table> <p>ECS Pump Status</p> <table><tr><td>ECS-A ON</td></tr><tr><td>ECS-B OFF</td></tr><tr><td>Booster OFF</td></tr></table> <p>Valves Status</p> <table><tr><td>CLOSED</td><td>OPEN</td><td>OPEN</td></tr><tr><td>CLOSED</td><td>OPEN</td><td>OPEN</td></tr><tr><td>OPEN</td><td>OPEN</td><td>OPEN</td></tr><tr><td>OPEN</td><td>OPEN</td><td>CLOSED</td></tr></table> <p>Subtest Valve Status</p> <table><tr><td>A</td><td>CLOSED</td><td>CLOSED</td><td>CLOSED</td><td>CLOSED</td></tr><tr><td>B</td><td>OPEN</td><td>CLOSED</td><td>CLOSED</td><td>CLOSED</td></tr><tr><td>C</td><td>OPEN</td><td>OPEN</td><td>CLOSED</td><td>CLOSED</td></tr><tr><td>D</td><td>OPEN</td><td>OPEN</td><td>OPEN</td><td>CLOSED</td></tr><tr><td>E</td><td>OPEN</td><td>OPEN</td><td>OPEN</td><td>OPEN</td></tr></table>	XTIE Pressure (psig)	Total ECS Flow (gpm)	68.92	0	56.04	5612	36.56	9234	23.55	10754	19.14	11295	0.0	11295	ON	OFF	ON	OFF	ON	OFF	ON	OFF	ON	OFF	ECS-A ON	ECS-B OFF	Booster OFF	CLOSED	OPEN	OPEN	CLOSED	OPEN	OPEN	OPEN	OPEN	OPEN	OPEN	OPEN	CLOSED	A	CLOSED	CLOSED	CLOSED	CLOSED	B	OPEN	CLOSED	CLOSED	CLOSED	C	OPEN	OPEN	CLOSED	CLOSED	D	OPEN	OPEN	OPEN	CLOSED	E	OPEN	OPEN	OPEN	OPEN	24 months
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Emergency Cooling System (ECS) - Operating
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SURVEILLANCE REQUIREMENTS (continued)

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SR 3.5.1.1.16	<p>Perform a flow test, through the test lines, of the ECS-B pump as a single ECS source with pump and valve status as indicated below. The flow test includes five subtests (A-E) for the specified combinations of valve status. Verify that the total ECS flow versus the Cross-Tie header centerline pressure for each subtest lies above the curve generated by connecting the points shown below with straight lines.</p> <table><thead><tr><th>XTIE Pressure (psig)</th><th>Total ECS Flow (gpm)</th></tr></thead><tbody><tr><td>68.64</td><td>0</td></tr><tr><td>56.54</td><td>5620</td></tr><tr><td>36.88</td><td>9241</td></tr><tr><td>22.12</td><td>10722</td></tr><tr><td>18.36</td><td>11245</td></tr><tr><td>0.0</td><td>11245</td></tr></tbody></table> <table><thead><tr><th colspan="2">190 Blde Pump Status</th><th colspan="2">ECS Pump Status</th></tr></thead><tbody><tr><td>ON</td><td>OFF</td><td>ECS-A</td><td>OFF</td></tr><tr><td>ON</td><td>OFF</td><td>ECS-B</td><td>ON</td></tr><tr><td>ON</td><td>OFF</td><td>Booster</td><td>OFF</td></tr><tr><td>ON</td><td>OFF</td><td></td><td></td></tr><tr><td>ON</td><td>OFF</td><td></td><td></td></tr></tbody></table> <table><thead><tr><th>Valve</th><th>Status</th><th>Valve</th><th>Status</th><th>Valve</th><th>Status</th></tr></thead><tbody><tr><td></td><td>CLOSED</td><td></td><td>OPEN</td><td></td><td>OPEN</td></tr><tr><td></td><td>CLOSED</td><td></td><td>OPEN</td><td></td><td>OPEN</td></tr><tr><td></td><td>OPEN</td><td></td><td>OPEN</td><td></td><td>OPEN</td></tr><tr><td></td><td>OPEN</td><td></td><td>OPEN</td><td></td><td>CLOSED</td></tr></tbody></table> <table><thead><tr><th>Subtest</th><th colspan="4">Valve Status</th></tr></thead><tbody><tr><td>A</td><td>CLOSED</td><td>CLOSED</td><td>CLOSED</td><td>CLOSED</td></tr><tr><td>B</td><td>OPEN</td><td>CLOSED</td><td>CLOSED</td><td>CLOSED</td></tr><tr><td>C</td><td>OPEN</td><td>OPEN</td><td>CLOSED</td><td>CLOSED</td></tr><tr><td>D</td><td>OPEN</td><td>OPEN</td><td>OPEN</td><td>CLOSED</td></tr><tr><td>E</td><td>OPEN</td><td>OPEN</td><td>OPEN</td><td>OPEN</td></tr></tbody></table>	XTIE Pressure (psig)	Total ECS Flow (gpm)	68.64	0	56.54	5620	36.88	9241	22.12	10722	18.36	11245	0.0	11245	190 Blde Pump Status		ECS Pump Status		ON	OFF	ECS-A	OFF	ON	OFF	ECS-B	ON	ON	OFF	Booster	OFF	ON	OFF			ON	OFF			Valve	Status	Valve	Status	Valve	Status		CLOSED		OPEN		OPEN		CLOSED		OPEN		OPEN		OPEN		OPEN		OPEN		OPEN		OPEN		CLOSED	Subtest	Valve Status				A	CLOSED	CLOSED	CLOSED	CLOSED	B	OPEN	CLOSED	CLOSED	CLOSED	C	OPEN	OPEN	CLOSED	CLOSED	D	OPEN	OPEN	OPEN	CLOSED	E	OPEN	OPEN	OPEN	OPEN	24 months
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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY																				
SR 3.5.1.1.17	Run the Booster Pump for ≥ 1 hour at a flow of $\geq 11,952$ gpm. Shutdown the pump and restart the pump within 60 seconds. Continue to run the pump for at least an additional 30 minutes.	24 months																				
SR 3.5.1.1.18	Simulate a loss of AC and DC power to the Bingham pump motors and verify upon Manual Incident Action, the rotovalves 21-A, 21-B, 51-A and 51-B close automatically in ≥ 10 and ≤ 25 seconds.	24 months																				
SR 3.5.1.1.19	<p>Perform an ECS functional flow test through the test lines with ECS-A and ECS-B pumps, in standby, all other pumping sources off¹, and all ECS valves initially aligned as indicated below and verify that upon actuation of Manual Incident Action that total ECS flow reaches $\geq 10,171$ gpm in ≤ 25 seconds.</p> <table><tr><td><u>Valve #</u></td><td><u>Status</u></td><td><u>Valve #</u></td><td><u>Status</u></td></tr><tr><td rowspan="7"> </td><td>CLOSED¹</td><td rowspan="7"> </td><td>CLOSED²</td></tr><tr><td>CLOSED¹</td><td>CLOSED²</td></tr><tr><td>CLOSED²</td><td>CLOSED²</td></tr><tr><td>CLOSED²</td><td>CLOSED¹</td></tr><tr><td>CLOSED²</td><td>CLOSED¹</td></tr><tr><td>CLOSED²</td><td>CLOSED¹</td></tr><tr><td>CLOSED²</td><td>CLOSED²</td></tr></table> <p>1) Valves must remain closed and pumps must remain off throughout the test.</p> <p>2) Valves must be capable of opening upon receipt of an actuation signal from Manual Incident Action.</p>	<u>Valve #</u>	<u>Status</u>	<u>Valve #</u>	<u>Status</u>		CLOSED ¹		CLOSED ²	CLOSED ¹	CLOSED ²	CLOSED ²	CLOSED ²	CLOSED ²	CLOSED ¹	CLOSED ²	CLOSED ¹	CLOSED ²	CLOSED ¹	CLOSED ²	CLOSED ²	24 months
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SR 3.5.1.1.20	Verify valve closed and is open.	1 month																				

Emergency Cooling System (ECS) - Operating
3.5.1.1

INFORMATION ONLY
CROSS-REFERENCES

TITLE	NUMBER
Emergency Cooling System (ECS) Cooling Water System Flow	3.5.1.2 3.7.1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.5.2.1.24 A. Verify by injection test that each side of the SSS is capable of delivering a gadolinium solution via pump suction injection to the tie-ins of the Process Water System loops in ≤ 7.8 seconds following loss of Process Water AC pump power.</p> <p style="padding-left: 40px;"><u>AND</u></p> <p> B. Verify by injection test that each side of the SSS is capable of delivering a gadolinium solution via pump suction injection to the tie-ins of the Process Water System Loops in ≤ 7.2 seconds following actuation of the seismic triggers.</p> <p style="padding-left: 40px;"><u>AND</u></p> <p> C. Verify the SSS is capable of delivering a gadolinium solution via pump suction injection to the tie-ins of the Process Water System loops at greater than the limits of Figure 3.5.2.1-2.</p>	<p>-----NOTE----- Both sides of the SSS shall be tested in 96 months. -----</p> <p>96 months</p>

CROSS-REFERENCES - None.

Supplementary Safety System (SSS) - Operating
3.5.2.1

FIGURE 3.5.2.1-1
SPARGER INJECTION

TIME FROM INJECTION START(s)	INJECTION RATE (gal/s)	TIME FROM INJECTION START(s)	INJECTION RATE (gal/s)
0	1.486	18	0.722
1	1.412	19	0.704
2	1.342	20	0.688
3	1.277	21	0.674
4	1.216	22	0.660
5	1.160	23	0.648
6	1.107	24	0.636
7	1.059	25	0.624
8	1.014	26	0.613
9	0.972	27	0.602
10	0.934	28	0.591
11	0.896	29	0.580
12	0.866	30	0.569
13	0.836	31	0.557
14	0.809	32	0.544
15	0.784	33	0.530
16	0.761	34	0.516
17	0.741	35	0.500

FIGURE 3.5.2.1-2
PUMP SUCTION INJECTION

TIME FROM INJECTION START(s)	INJECTION RATE (gal/s)	TIME FROM INJECTION START(s)	INJECTION RATE (gal/s)	TIME FROM INJECTION START(s)	INJECTION RATE (gal/s)
0	0.734	22	0.378	44	0.304
1	0.705	23	0.371	45	0.302
2	0.679	24	0.365	46	0.301
3	0.653	25	0.359	47	0.299
4	0.629	26	0.354	48	0.298
5	0.607	27	0.349	49	0.296
6	0.585	28	0.344	50	0.295
7	0.565	29	0.340	51	0.293
8	0.546	30	0.336	52	0.292
9	0.528	31	0.333	53	0.290
10	0.511	32	0.330	54	0.289
11	0.495	33	0.327	55	0.288
12	0.481	34	0.324	56	0.286
13	0.467	35	0.321	57	0.285
14	0.454	36	0.319	58	0.284
15	0.442	37	0.317	59	0.282
16	0.431	38	0.314	60	0.281
17	0.420	39	0.312	61	0.280
18	0.410	40	0.311	62	0.279
19	0.401	41	0.309	63	0.278
20	0.393	42	0.307	64	0.278
21	0.385	43	0.305	65	0.277

Supplementary Safety System (SSS) - Shutdown 3.5.2.2

3.5 EMERGENCY COOLING SYSTEMS/EMERGENCY SHUTDOWN SYSTEMS

3.5.2.2 Supplementary Safety System (SSS) - Shutdown

LCO 3.5.2.2 Both sides of the Supplementary Safety System (SSS) shall be OPERABLE to permit injection of gadolinium via the sparger injection paths following manual actuation from the SSS Panel Pull Ring or the explosive valve firing switches.

APPLICABILITY: REACTOR SHUTDOWN when the actuator is latched and power is available to the control rod drive motors or POSITIVE REACTIVITY CHANGES are in progress.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One side of the SSS is inoperable.	A.1 Restore the inoperable SSS side to OPERABLE status.	72 hours
B. The SSS Panel Pull Ring System is inoperable. <u>AND</u> The explosive valve manual firing switches are inoperable. <u>OR</u> Both sides of the SSS are inoperable. <u>OR</u> Required Action and associated Completion Time for Condition A not met.	B.1 Suspend all operations involving POSITIVE REACTIVITY CHANGES. <u>AND</u> B.2 If rods are capable of being withdrawn or are withdrawn, insert any withdrawn rods and de-energize the rod drive power supplies.	IMMEDIATELY IMMEDIATELY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.2.1	Verify that the nitrogen pressure in each of the gadolinium storage tanks for sparger injection is ≥ 280 psig and ≤ 320 psig	12 hours
SR 3.5.2.2.2	Verify that the gadolinium volume in each of the storage tanks for sparger injection is ≥ 43.0 and ≤ 44.7 gallons.	12 hours
SR 3.5.2.2.3	Verify that at least 2 of 3 probe pot line level lights for sparger injection are illuminated on each side.	12 hours
SR 3.5.2.2.4	Verify that the sparger injection line block valves are open.	12 hours
SR 3.5.2.2.5	<p>Verify that both explosive valve circuit monitors indicate ≥ 8 mV and ≤ 15 mV.</p> <p><u>AND</u></p> <p>Verify that at least one explosive valve battery per side has a voltage ≥ 24 volts with the associated charger off.</p>	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.2.6	Verify that the sparger injection pneumatic valve nitrogen closure pressure is ≤ 27 psig.	<p>-----NOTE----- Only required when explosive valve actuation is inoperable -----</p> <p>24 hours</p>
SR 3.5.2.2.7	Verify that each explosive valve nickel cadmium (nicad) battery unit is ≤ 40 months old.	6 months
SR 3.5.2.2.8	Verify that all explosive valve primer assemblies are ≤ 52 months old.	6 months
SR 3.5.2.2.9	Verify that each (two per side) 24 volt explosive valve battery unit and associated firing circuitry can provide at least 5 amperes for at least 8 msec.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.2.10	Verify that the clearance between the primer assembly in each explosive valve and the trigger body is ≤ 0.004 inch.	During primer assembly installation.
SR 3.5.2.2.11	Verify that the gadolinium concentration charged to the SSS is ≥ 0.14 g/ml and ≤ 0.50 g/ml.	Prior to charging gadolinium to the SSS.
SR 3.5.2.2.12	Verify that the injection lines associated with each sparger injection storage tank contain ≤ 500 ml of air.	Following refilling of the sparger injection lines.
SR 3.5.2.2.13	Verify that all injection line fill/drain/vent connections for sparger injection are double isolated (blind flanges, quick disconnect fittings, caps, etc.).	Prior to declaring the SSS OPERABLE after charging gadolinium to the SSS.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.2.14	Verify that the sparger injection orifices are installed.	Prior to declaring the SSS OPERABLE after charging gadolinium to the SSS.
SR 3.5.2.2.15	Actuate the pull ring and verify that the sparger injection pneumatic valve for each side of the SSS opens completely within 5 seconds.	24 months
SR 3.5.2.2.16	<p>A. Verify by injection test that each side of the SSS is capable of delivering a gadolinium solution via sparger injection to the moderator space in ≤ 4.8 seconds following loss of Process Water AC pump power.</p> <p><u>AND</u></p> <p>B. Verify the SSS is capable of delivering a gadolinium solution via sparger injection into the moderator space at greater than the limits of Figure 3.5.2.2-1.</p>	<p>-----NOTE----- Both sides of the SSS shall be tested in 96 months. -----</p> <p>96 months</p>

CROSS-REFERENCES - None.

FIGURE 3.5.2.2-1
SPARGER INJECTION

TIME FROM INJECTION START(s)	INJECTION RATE (gal/s)	TIME FROM INJECTION START(s)	INJECTION RATE (gal/s)
0	1.486	18	0.722
1	1.412	19	0.704
2	1.342	20	0.688
3	1.277	21	0.674
4	1.216	22	0.660
5	1.160	23	0.648
6	1.107	24	0.636
7	1.059	25	0.624
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9	0.972	27	0.602
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12	0.866	30	0.569
13	0.836	31	0.557
14	0.809	32	0.544
15	0.784	33	0.530
16	0.761	34	0.516
17	0.741	35	0.500

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INFORMATION ONLY

Emergency Cooling System (ECS) - Shutdown
3.5.1.2

3.5 EMERGENCY COOLING SYSTEMS/EMERGENCY SHUTDOWN SYSTEMS

3.5.1.2 Emergency Cooling System (ECS) - Shutdown

LCO 3.5.1.2

- A. Two ECS water sources including headers, valves and pump(s) as specified in Tables 3.5.1.2-1 or 3.5.1.2-2 shall be OPERABLE.

AND

- B. The borated water header shall contain a sodium polyborate solution of ≥ 0.0175 and ≤ 0.04 g/ml at $> 5^{\circ}\text{C}$ and $\leq 40^{\circ}\text{C}$.

AND

- C. Two ECS addition systems, (at least one of which shall be addition system two or five), including [REDACTED] series valves and [REDACTED] series valves shall be OPERABLE

AND

- D. Either Rotovalve Interlock Circuit and associated rotovalves 21-A and 21-B or Rotovalve Interlock Circuit and associated rotovalves 51-A and 51-B shall be OPERABLE for at least one of the addition systems required to be OPERABLE in C above.

APPLICABILITY: REACTOR SHUTDOWN, with HGAs in the reactor and shutdown time greater than 3 hours:

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INFORMATION ONLY

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Less than two of the required ECS sources are OPERABLE.	A.1 Suspend CHARGE AND DISCHARGE OPERATIONS.	IMMEDIATELY
	<u>AND</u> A.2 Restore at least two ECS sources to an OPERABLE status.	72 hours
B. Crosstie header pressure not within required limits.	B.1 Restore crosstie header pressure to within limits.	30 minutes
C. Rotovalve(s) 21-A, 21-B, 51-A or 51-B are inoperable.	C.1 Restore the inoperable rotovalve(s) to OPERABLE status.	72 hours
	<u>OR</u> C.2 Close and deenergize the inoperable rotovalve(s).	72 hours
D. Less than 2 of the required ECS addition systems are OPERABLE.	D.1 Suspend CHARGE AND DISCHARGE OPERATIONS.	IMMEDIATELY
	<u>AND</u> D.2 Restore at least 2 ECS addition systems to an OPERABLE status.	72 hours
E. Sodium polyborate solution concentration and/or temperature <u>not</u> within the required limits.	E.1 Restore the sodium polyborate solution concentration and/or temperature to within the required limits.	96 hours

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INFORMATION ONLY

Emergency Cooling System (ECS) - Shutdown
3.5.1.2

Table 3.5.1.2-1

EMERGENCY COOLING SYSTEM WATER SOURCES

PUMP(S) REQUIRED	HEADER(S) REQUIRED	NORMAL VALVE ALIGNMENT	ISOLATION VALVE STATUS
ECS-A pump		-----NOTE----- may be OPEN, if OPERABLE ----- CLOSED OPEN CLOSED OPEN CLOSED CLOSED	CLOSED and OPERABLE CLOSED and OPERABLE
ECS-B pump		-----NOTE----- may be OPEN, if OPERABLE ----- CLOSED OPEN CLOSED OPEN CLOSED CLOSED	CLOSED and OPERABLE CLOSED and OPERABLE
ECS Booster Pump		-----NOTE----- may be OPEN, if OPERABLE ----- CLOSED CLOSED OPEN CLOSED CLOSED	CLOSED and OPERABLE CLOSED and OPERABLE

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Emergency Cooling System (ECS) - Shutdown
3.5.1.2.

Table 3.5.1.2-2

COOLING WATER SYSTEM HEADERS AS ECS WATER SOURCES

PUMP(S) REQUIRED	HEADER(S) REQUIRED	VALVE ALIGNMENT DURING OPERATION	ISOLATION VALVE STATUS
* 190 building pumps	pressurized to ≥ 45 and ≤ 90 psig.	OPEN OPEN CLOSED OPEN	CLOSED and OPERABLE
* 190 building pumps	pressurized to ≥ 45 and ≤ 90 psig.	OPEN OPEN	CLOSED and OPERABLE

- At least three Building 190 CW pumps must be operating on a header to qualify that header as an ECS water source. Additional pumps are required depending on the number of Process Water heat exchangers that are in service on the header. See the following table for requirements:

Number of heat exchangers in service on a CW header	Number of Building 190 CW pumps required on the respective header
1 or 2 heat exchangers	3 CW pumps
3 heat exchangers	4 CW pumps
4, 5, or 6 heat exchangers	5 CW pumps

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Emergency Cooling System (ECS) - Shutdown
3.5.1.2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.2.1	Verify that: A) each required [REDACTED] series crosstie header isolation valve is closed and OPERABLE. B) each required [REDACTED] series ECS addition valve is closed and OPERABLE. C) each required [REDACTED] series ECS addition valve is closed and OPERABLE. D) all valves associated with required OPERABLE ECS pumping sources are properly aligned as specified in Table 3.5.1.2-1.	12 hours
SR 3.5.1.2.2	Verify that the Booster Pump is in a standby condition by performance of a status check.	12 hours when the Booster Pump is used as an ECS source.
SR 3.5.1.2.3	Verify that the borated water header temperature is $> 5^{\circ}\text{C}$ and $\leq 40^{\circ}\text{C}$.	24 hours
SR 3.5.1.2.4	Verify for the required ECS addition systems that the manual block valves [REDACTED] are in the full open position.	12 hours

(continued)

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Emergency Cooling System (ECS) - Shutdown
3.5.1.2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.5.1.2.5 Verify that [REDACTED] or [REDACTED] header pressures are ≥ 45 psig and ≤ 90 psig.	12 hours when [REDACTED] or [REDACTED] are used as ECS sources
SR 3.5.1.2.6 Verify that the crosstie header pressure is ≥ 45 psig.	12 hours
SR 3.5.1.2.7 Manually start each ECS-A, ECS-B and Booster Pump separately and run for at least 10 minutes.	14 days
SR 3.5.1.2.8 Verify by lab analysis that the borated water header sodium polyborate solution concentration is ≥ 0.0175 and ≤ 0.04 g/ml.	Every 24 hours if conductivity is < 3651 or > 8210 μ mhos. <u>AND</u> Within 24 hours of any addition to the borated water header. <u>AND</u> 1 month

(continued)

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Emergency Cooling System (ECS) - Shutdown
3.5.1.2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.2.9	Verify that the borated water header filter screen bypass lines open when the pressure drop across the screen is 7 to 7.5 psid based on measured torque settings.	1 month
SR 3.5.1.2.10	Start and run ECS-A and ECS-B pumps from the emergency buses for at least 30 minutes and then restart and run the pumps from the containment substation for an additional 30 minutes.	1 year

(continued)

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Emergency Cooling System (ECS) - Shutdown
3.5.1.2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY																																				
SR 3.5.1.2.11	24 months																																				
<p>Perform a flow test, through the test lines, of the [REDACTED] header as a single ECS source with pump and valve status as indicated below. The flow test includes five subtests (A-E) for the specified combinations of [REDACTED] valve status. Verify that the total ECS flow versus the Cross-Tie [REDACTED] header centerline pressure for each subtest lies above the curve generated by connecting the points shown below with straight lines.</p>																																					
<table><thead><tr><th><u>XTIE Pressure</u> (psig)</th><th><u>Total ECS Flow</u> (gpm)</th></tr></thead><tbody><tr><td>45.83</td><td>0</td></tr><tr><td>43.13</td><td>4165</td></tr><tr><td>37.83</td><td>7219</td></tr><tr><td>33.46</td><td>9338</td></tr><tr><td>30.21</td><td>11305</td></tr><tr><td>0.0</td><td>11305</td></tr></tbody></table>		<u>XTIE Pressure</u> (psig)	<u>Total ECS Flow</u> (gpm)	45.83	0	43.13	4165	37.83	7219	33.46	9338	30.21	11305	0.0	11305																						
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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY																																																																																																		
SR 3.5.1.2.12	<p>Perform a flow test, through the test lines, of the [REDACTED] header as a single ECS source with pump and valve status as indicated below. The flow test includes five subtests (A-E) for the specified combinations of [REDACTED] valve status. Verify that the total ECS flow versus the Cross-Tie [REDACTED] header centerline pressure for each subtest lies above the curve generated by connecting the points shown below with straight lines.</p> <table><tr><th>XTIE Pressure (psig)</th><th>Total ECS Flow (gpm)</th></tr><tr><td>45.76</td><td>0</td></tr><tr><td>41.41</td><td>5177</td></tr><tr><td>36.82</td><td>8858</td></tr><tr><td>32.11</td><td>11578</td></tr><tr><td>26.49</td><td>14023</td></tr><tr><td>0.0</td><td>14023</td></tr></table> <table><tr><th colspan="2">190 Bldg Pump Status</th><th colspan="2">ECS Pump Status</th></tr><tr><td>[REDACTED]</td><td>OFF</td><td>[REDACTED]</td><td>ON</td></tr><tr><td>[REDACTED]</td><td>OFF</td><td>[REDACTED]</td><td>ON</td></tr><tr><td>[REDACTED]</td><td>OFF</td><td>[REDACTED]</td><td>ON</td></tr><tr><td>[REDACTED]</td><td>OFF</td><td>[REDACTED]</td><td>ON</td></tr><tr><td>[REDACTED]</td><td>OFF</td><td>[REDACTED]</td><td>ON</td></tr></table> <table><tr><th>Valves</th><th>Status</th><th>Valves</th><th>Status</th><th>Valves</th><th>Status</th></tr><tr><td>[REDACTED]</td><td>CLOSED</td><td>[REDACTED]</td><td>OPEN</td><td>[REDACTED]</td><td>OPEN</td></tr><tr><td>[REDACTED]</td><td>OPEN</td><td>[REDACTED]</td><td>OPEN</td><td>[REDACTED]</td><td>OPEN</td></tr><tr><td>[REDACTED]</td><td>CLOSED</td><td>[REDACTED]</td><td>OPEN</td><td>[REDACTED]</td><td>OPEN</td></tr><tr><td>[REDACTED]</td><td>CLOSED</td><td>[REDACTED]</td><td>OPEN</td><td>[REDACTED]</td><td>CLOSED</td></tr></table> <table><tr><th>Subtest</th><th colspan="4">Valve Status</th></tr><tr><td>A</td><td>CLOSED</td><td>CLOSED</td><td>CLOSED</td><td>CLOSED</td></tr><tr><td>B</td><td>OPEN</td><td>CLOSED</td><td>CLOSED</td><td>CLOSED</td></tr><tr><td>C</td><td>OPEN</td><td>OPEN</td><td>CLOSED</td><td>CLOSED</td></tr><tr><td>D</td><td>OPEN</td><td>OPEN</td><td>OPEN</td><td>CLOSED</td></tr><tr><td>E</td><td>OPEN</td><td>OPEN</td><td>OPEN</td><td>OPEN</td></tr></table>	XTIE Pressure (psig)	Total ECS Flow (gpm)	45.76	0	41.41	5177	36.82	8858	32.11	11578	26.49	14023	0.0	14023	190 Bldg Pump Status		ECS Pump Status		[REDACTED]	OFF	[REDACTED]	ON	[REDACTED]	OFF	[REDACTED]	ON	[REDACTED]	OFF	[REDACTED]	ON	[REDACTED]	OFF	[REDACTED]	ON	[REDACTED]	OFF	[REDACTED]	ON	Valves	Status	Valves	Status	Valves	Status	[REDACTED]	CLOSED	[REDACTED]	OPEN	[REDACTED]	OPEN	[REDACTED]	OPEN	[REDACTED]	OPEN	[REDACTED]	OPEN	[REDACTED]	CLOSED	[REDACTED]	OPEN	[REDACTED]	OPEN	[REDACTED]	CLOSED	[REDACTED]	OPEN	[REDACTED]	CLOSED	Subtest	Valve Status				A	CLOSED	CLOSED	CLOSED	CLOSED	B	OPEN	CLOSED	CLOSED	CLOSED	C	OPEN	OPEN	CLOSED	CLOSED	D	OPEN	OPEN	OPEN	CLOSED	E	OPEN	OPEN	OPEN	OPEN	24 months
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Emergency Cooling System (ECS) - Shutdown
3.5.1.2

SURVEILLANCE REQUIREMENTS (continued)

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Emergency Cooling System (ECS) - Shutdown
3.5.1

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.2.14	SURVEILLANCE	FREQUENCY																																																																				
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Emergency Cooling System (ECS) - Shutdown
3.5.1.2

SURVEILLANCE REQUIREMENTS (continued)

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SR 3.5.1.2.15	<p>Perform a flow test, through the test lines, of the ECS-B pump as a single ECS source with pump and valve status as indicated below. The flow test includes five subtests (A-E) for the specified combinations of valve status. Verify that the total ECS flow versus the Cross-Tie Header centerline pressure for each subtest lies above the curve generated by connecting the points shown below with straight lines.</p> <table><thead><tr><th>XTIE Pressure (psig)</th><th>Total ECS Flow (gpm)</th></tr></thead><tbody><tr><td>68.64</td><td>0</td></tr><tr><td>56.54</td><td>5620</td></tr><tr><td>36.88</td><td>9241</td></tr><tr><td>22.12</td><td>10722</td></tr><tr><td>18.36</td><td>11245</td></tr><tr><td>0.0</td><td>11245</td></tr></tbody></table> <p>190 Blade Pump Status</p> <table><tr><td>ON</td><td>OFF</td></tr><tr><td>ON</td><td>OFF</td></tr><tr><td>ON</td><td>OFF</td></tr><tr><td>ON</td><td>OFF</td></tr><tr><td>ON</td><td>OFF</td></tr></table> <p>ECS Pump Status</p> <table><tr><td>ECS-A OFF</td></tr><tr><td>ECS-B ON</td></tr><tr><td>Booster OFF</td></tr></table> <p>Valves Status</p> <table><tr><td>CLOSED</td><td>OPEN</td><td>OPEN</td></tr><tr><td>CLOSED</td><td>OPEN</td><td>OPEN</td></tr><tr><td>OPEN</td><td>OPEN</td><td>OPEN</td></tr><tr><td>OPEN</td><td>OPEN</td><td>CLOSED</td></tr></table> <p>Subtest Valve Status</p> <table><tr><td>A</td><td>CLOSED</td><td>CLOSED</td><td>CLOSED</td><td>CLOSED</td></tr><tr><td>B</td><td>OPEN</td><td>CLOSED</td><td>CLOSED</td><td>CLOSED</td></tr><tr><td>C</td><td>OPEN</td><td>OPEN</td><td>CLOSED</td><td>CLOSED</td></tr><tr><td>D</td><td>OPEN</td><td>OPEN</td><td>OPEN</td><td>CLOSED</td></tr><tr><td>E</td><td>OPEN</td><td>OPEN</td><td>OPEN</td><td>OPEN</td></tr></table>	XTIE Pressure (psig)	Total ECS Flow (gpm)	68.64	0	56.54	5620	36.88	9241	22.12	10722	18.36	11245	0.0	11245	ON	OFF	ON	OFF	ON	OFF	ON	OFF	ON	OFF	ECS-A OFF	ECS-B ON	Booster OFF	CLOSED	OPEN	OPEN	CLOSED	OPEN	OPEN	OPEN	OPEN	OPEN	OPEN	OPEN	CLOSED	A	CLOSED	CLOSED	CLOSED	CLOSED	B	OPEN	CLOSED	CLOSED	CLOSED	C	OPEN	OPEN	CLOSED	CLOSED	D	OPEN	OPEN	OPEN	CLOSED	E	OPEN	OPEN	OPEN	OPEN	24 months
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Emergency Cooling System (ECS) - Shutdown
3.5.1.2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.2.16	Run the Booster Pump for ≥ 1 hour at a flow of $\geq 11,952$ gpm. Shutdown the pump and restart the pump within 60 seconds. Continue to run the pump for at least an additional 30 minutes.	24 months
SR 3.5.1.2.17	Simulate a loss of AC and DC power to the Bingham pump motors and verify upon Manual Incident Action, the rotovalves 21-A, 21-B, 51-A and 51-B close automatically in ≥ 10 and ≤ 25 seconds.	24 months
SR 3.5.1.2.18	Verify valve [REDACTED] closed and [REDACTED] is open.	1 month

CROSS-REFERENCES - None.

"DELETED VERSION"

Supplementary Safety System (SSS) - Operating
3.5.2.1

3.5 EMERGENCY COOLING SYSTEMS/EMERGENCY SHUTDOWN SYSTEMS

3.5.2.1 Supplementary Safety System (SSS) - Operating

LCO 3.5.2.1 Both sides of the Supplementary Safety System (SSS) shall be OPERABLE.

APPLICABILITY: REACTOR, STARTUP and REACTOR OPERATION

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One side of the SSS is inoperable.	A.1 Restore the inoperable SSS side to OPERABLE status.	72 hours
B. The SSS Panel Pull Ring System is inoperable.	B.1 Restore the SSS Panel Pull Ring System to OPERABLE status.	72 hours
C. Both sides of the SSS are inoperable. <u>OR</u> Required Action(s) and associated Completion Time(s) for Conditions A or B not met.	C.1 Be in REACTOR SHUTDOWN.	90 minutes

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.1.6	Verify that the gadolinium solution level is within the standpipe sight glass for each side of pump suction injection or by the absence of the gadolinium solution tank level alarm.	12 hours
SR 3.5.2.1.7	Perform a status check on the pump suction injection control panel (-40').	12 hours
SR 3.5.2.1.8	Verify that the sparger injection and pump suction injection in-line block valves are open.	Within 48 hours prior to locking the -20' and -40' radiologically controlled areas.
SR 3.5.2.1.9	Verify battery voltage for each Seismic Trigger at the -40' power supply panel is ≥ 11.5 VDC.	12 hours
SR 3.5.2.1.10	Verify that the sparger injection and pump suction injection pneumatic valve nitrogen closure pressure is ≤ 27 psig.	24 hours
SR 3.5.2.1.11	Verify that the gadolinium concentration charged to the SSS is ≥ 0.14 g/ml and ≤ 0.50 g/ml.	Prior to charging gadolinium to the SSS.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.1.12	Verify that the injection lines associated with each sparger injection storage tank contain ≤ 500 ml of air.	Prior to REACTOR STARTUP if the SSS injection lines have been drained and refilled
SR 3.5.2.1.13	Verify that all injection line fill/drain/vent connections for sparger injection and pump suction injection are double isolated (blind flanges, quick disconnect fittings, caps, etc.).	Prior to REACTOR STARTUP after charging gadolinium to the SSS.
SR 3.5.2.1.14	Verify that the sparger injection and pump suction injection orifices are installed.	Prior to REACTOR STARTUP
SR 3.5.2.1.15	Verify that each Seismic Trigger -40' Power Supply Panel battery is ≤ 18 months old.	6 months
SR 3.5.2.1.16	Perform a CHANNEL CALIBRATION of the seismic triggers.	Every scheduled C&D outage.
SR 3.5.2.1.17	Verify that each explosive valve nickel cadmium (nicad) battery unit is ≤ 40 months old.	6 months
SR 3.5.2.1.18	Verify that all explosive valve primer assemblies are ≤ 52 months old.	6 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.1.19	Verify that each (two per side) 24 volt explosive valve battery unit and associated firing circuitry can provide at least 5 amperes for at least 8 msec.	18 months
SR 3.5.2.1.20	Actuate the pull ring and verify that the sparger injection pneumatic valve for each side of the SSS opens completely within 5 seconds.	24 months
SR 3.5.2.1.21	Pressure test the spargers and verify that the internal sparger pressure increase is < 1.0 psig when process water flow is increased from 6 DC to 6 AC flow.	48 months <u>AND</u> Prior to REACTOR STARTUP after installation of sparger(s)
SR 3.5.2.1.22	Verify that the clearance between the primer assembly in each explosive valve and the trigger body is ≤ 0.004 inch.	During primer assembly installation

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.5.2.1.23 A. Verify by injection test that each side of the SSS is capable of delivering a gadolinium solution via sparger injection to the moderator space in ≤ 4.8 seconds following loss of Process Water AC pump power.</p> <p><u>AND</u></p> <p>B. Verify by injection test that each side of the SSS is capable of delivering a gadolinium solution via sparger injection to the moderator space in ≤ 4.2 seconds following actuation of the seismic triggers.</p> <p><u>AND</u></p> <p>C. Verify the SSS is capable of delivering a gadolinium solution via sparger injection into the moderator space at greater than the limits of Figure 3.5.2.1-1.</p>	<p>-----NOTE----- Both sides of the SSS shall be tested in 96 months. -----</p> <p>96 months</p>

(continued)

"DELETED VERSION"

Airborne Activity Confinement System - HGAs
3.6.1

3.6 CONFINEMENT SYSTEMS

INFORMATION ONLY

3.6.1 Airborne Activity Confinement System (AACS) - HGAs

LCO 3.6.1

The AACS shall be OPERABLE and in operation with:

A. At least two [REDACTED] exhaust fans in service.

AND

B. At least three AACS filter compartments in service.

APPLICABILITY: REACTOR STARTUP, REACTOR OPERATION, and REACTOR SHUTDOWN with HGAs in the reactor tank or movement of any component containing irradiated FISSILE MATERIAL (excluding Californium source rods).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Incident ventilation control switch inoperable.	A.1 Restore the switch to OPERABLE status.	8 hours
B. [REDACTED] or [REDACTED] damper interlock inoperable.	B.1 Restore affected interlock to OPERABLE status.	8 hours

(continued)

"DELETED VERSION"

"DELETED VERSION"

Airborne Activity Confinement System - HGAs
3.6.1

INFORMATION ONLY

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Less than two [REDACTED] fans OPERABLE and in service.	C.1.1 Verify that at least one [REDACTED] fan is OPERABLE and in service.	15 minutes
	<u>AND</u>	
	C.1.2 Complete satisfactory performance of SR 3.6.1.2.	15 minutes
	<u>AND</u>	
	C.2 Restore at least two [REDACTED] fans to an OPERABLE and in service status.	3 hours
D. Less than three AACS filter compartments are OPERABLE and in service.	D.1.1 Verify that at least two AACS filter compartments are OPERABLE and in service.	15 minutes
	<u>AND</u>	
	D.1.2 Complete satisfactory performance of SR 3.6.1.2.	15 minutes
	<u>AND</u>	
	D.2 Restore at least three AACS filter compartments to an OPERABLE and in service status.	1 hour

(continued)

"DELETED VERSION"

INFORMATION ONLY

Shelter Activity Confinement System - HGAs
3.6.1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. The process room, or the pump rooms and heat exchanger bays, or the purification process areas are not at negative pressure with respect to atmosphere.	E.1 Restore the process room, pump rooms and heat exchanger bays, and the purification process areas to a negative pressure with respect to atmosphere.	4 hours
	<u>AND</u> E.2 Suspend CHARGE AND DISCHARGE OPERATIONS.	IMMEDIATELY
F. Required Actions of Conditions A through E not met within the required Completion Times during REACTOR STARTUP or REACTOR OPERATION.	F.1 Be in REACTOR SHUTDOWN.	90 minutes
G. Required Actions of Conditions A through E not met within the required Completion Times during REACTOR SHUTDOWN.	G.1 Suspend all operations involving POSITIVE REACTIVITY CHANGES.	IMMEDIATELY
	<u>OR</u> G.2 Stop CHARGE AND DISCHARGE OPERATIONS.	IMMEDIATELY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.1.1 Verify that at least three AACs filter compartments are latched, sealed and their associated H dampers are fully opened.	12 hours

"DELETED VERSION"
INTENTIONALLY ONLY

Airborne Activity Confinement System - HGAs
 3.6.1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.2	Verify that the process room, pump room, heat exchanger bays, and the purification process areas are at a negative pressure with respect to atmosphere.	12 hours
SR 3.6.1.3	Verify that [REDACTED] damper is open and the [REDACTED] damper is closed.	-----NOTE----- Only required during REACTOR STARTUP and REACTOR OPERATION ----- 12 hours
SR 3.6.1.4	Verify that at least two [REDACTED] fans are in service and their associated [REDACTED] dampers are open.	24 hours
SR 3.6.1.5	Verify that the carbon in each AACS filter compartment has been in service (flow through it) for less than or equal to 18 months.	1 month
SR 3.6.1.6	Verify that the resistance to flow through the online filter compartments is such that $F/p \geq 21,000$, where: F = The total air flow through all the online filter compartments in cfm, and p = The average pressure drop across all HEPA filters in service, in inches of water.	1 month
SR 3.6.1.7	Verify that the [REDACTED] damper interlocks function properly.	12 months

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"DELETED VERSION"

INTENTIONALLY DELETED

Airborne Activity Confinement System - HGAs
3.6.1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.8	Verify that the Fan damper interlock is OPERABLE.	12 months
SR 3.6.1.9	Verify that the Incident Ventilation Control Switch functions properly and performs all required actions.	12 months
SR 3.6.1.10	Verify that the filter compartment damper controls are OPERABLE.	12 months
SR 3.6.1.11	Verify that the process room door seals function properly.	12 months
SR 3.6.1.12	Verify the instantaneous bypass flow for the carbon filter in each AACS filter compartment is $< 0.05\%$.	18 months
SR 3.6.1.13	Verify the instantaneous bypass flow for the HEPA filter in each AACS filter compartment is $< 0.05\%$.	18 months
SR 3.6.1.14	Verify that the ignition temperature of the carbon used in the AACS filter compartments is $\geq 330^{\circ}\text{C}$.	Prior to installation.
SR 3.6.1.15	Verify elemental iodine retention is $\geq 99.5\%$ for the carbon filters.	Prior to installation.

"DELETED VERSION"

INFORMATION ONLY

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.16	Verify that the HEPA filters have the capability to retain $\geq 99.97\%$ of particulates ≥ 0.3 microns in diameter.	Prior to installation.

CROSS-REFERENCES

TITLE	NUMBER
Incident Control	3.3.4.2

Airborne Activity Confinement System - No HGAs

INFORMATION ONLY

3.6 CONFINEMENT SYSTEMS

3.6.2 Airborne Activity Confinement System (AACS) - No HGAs

LC0 3.6.2 The AACS shall be OPERABLE and in operation with:

A. At least one [REDACTED] exhaust fan in service.

AND

B. At least one AACCS filter compartment in service.

-----NOTE-----
Both shield doors may be opened for up to 1 hour for the purpose of moving the C & D machines into the process room.

APPLICABILITY: 1) REACTOR SHUTDOWN with no HGA's and POSITIVE REACTIVITY CHANGES in progress

2) Movement of Californium source rods in progress.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. Less than one [REDACTED] fan OPERABLE and in service.</p> <p><u>OR</u></p> <p>Less than one AACS filter compartment OPERABLE and in service.</p> <p><u>OR</u></p> <p>Incident ventilation control switch inoperable.</p> <p><u>OR</u></p> <p>[REDACTED] damper interlock inoperable.</p>	<p>A.1 Suspend POSITIVE REACTIVITY CHANGES.</p> <p><u>AND</u></p> <p>-----NOTE----- Performance of A.2 activities shall not preclude completion of actions to establish a safe conservative condition. -----</p> <p>A.2 Suspend movement of Californium source rods.</p> <p>"DELETED VERSION"</p>	<p>IMMEDIATELY</p> <p>IMMEDIATELY</p>

"DELETED VERSION"

Airborne Activity Confinement System - No HGAs
3.6.2

INFORMATION ONLY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.1	Verify that at least one AACS filter compartment is latched, sealed and its associated dampers are fully opened.	12 hours
SR 3.6.2.2	Verify that the process room is at a negative pressure with respect to atmosphere.	12 hours
SR 3.6.2.3	Verify that and dampers are OPERABLE.	12 hours
SR 3.6.2.4	Verify that at least one fan is in service, and its associated damper is fully opened.	24 hours
SR 3.6.2.5	Verify that the carbon in each online AACS filter compartment has been in service (flow through it) for ≤ 18 months.	1 month
SR 3.6.2.6	<p>Verify that the resistance to flow through the online filter compartments is such that</p> <p>$F/p \geq 21,000$, where:</p> <p>F = The total air flow through all the online filter compartments in cfm, and</p> <p>p = The average pressure drop across all HEPA filters in service, in inches of water.</p>	1 month
SR 3.6.2.7	Verify that the damper interlocks function properly.	12 months

"DELETED VERSION"

(continued)

"DELETED VERSION"

Airborne Activity Confinement System - No HGAs
3.6.2

INFORMATION ONLY

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.2.8	Verify that the Incident Ventilation Control Switch functions properly and performs all required actions.	12 months
SR 3.6.2.9	Verify that the Fan damper interlock is OPERABLE.	12 months
SR 3.6.2.10	Verify that the filter compartment damper controls are OPERABLE.	12 months
SR 3.6.2.11	Verify the instantaneous bypass flow for the carbon filter in each AACS filter compartment is $< 0.05\%$.	18 months
SR 3.6.2.12	Verify the instantaneous bypass flow for the HEPA filter in each AACS filter compartment is $< 0.05\%$.	18 months
SR 3.6.2.13	Verify that the ignition temperature of the carbon used in the AACS filter compartments is $\geq 330^{\circ}\text{C}$.	Prior to installation.
SR 3.6.2.14	Verify elemental iodine retention is $\geq 99.5\%$ for the carbon filters.	Prior to installation.

(continued)

"DELETED VERSION"

INFORMATION ONLY

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.2.15	Verify that the HEPA filters have the capability to retain $\geq 99.97\%$ of particulates ≥ 0.3 microns in diameter.	Prior to installation.

CROSS-REFERENCES - None.

"DELETED VERSION"

INFORMATION ONLY

Cooling Water System Flow
3.7.1

3.7 PLANT SYSTEMS

3.7.1 Cooling Water System Flow

LCO 3.7.1

A. Cooling Water System pumps [REDACTED] shall be
OPERABLE and in operation.

AND

B. The Cooling Water System valves defined in Table 3.7.1-1
shall be fully open.

APPLICABILITY: REACTOR STARTUP and REACTOR OPERATION.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any Cooling Water System pump inoperable or not in operation. <u>OR</u> Any Cooling Water System valve defined in Table 3.7.1-1 not fully open.	A.1 Be in REACTOR SHUTDOWN.	90 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.1.1 Verify that Cooling Water System pumps [REDACTED] are OPERABLE and in operation.	12 hours

"DELETED VERSION"

(continued)

INFORMATION ONLY

Cooling Water System Flow
3.7.1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.1.2	Verify that all Cooling Water System valves defined in Table 3.7.1-1 are fully open.	12 hours

CROSS-REFERENCES - None.

"DELETED VERSION"

INFORMATION ONLY

Cooling Water System Flow
3.7.1

TABLE 3.7.1-1

COOLING WATER SYSTEM VALVES

"DELETED VERSION"

INFORMATION ONLY

Ultimate Heat Sink
3.7.2

3.7 PLANT SYSTEMS

3.7.2 Ultimate Heat Sink

LCO 3.7.2

The 186 basin shall be OPERABLE with:

A. The cooling water temperature $\geq 5^{\circ}\text{C}$, and $\leq 30^{\circ}\text{C}$.

AND

B. Pump suction well level ≥ 13.8 ft.

AND

C. All three sections of the 186 basin and two 190 pump suction wells interconnected.

APPLICABILITY: REACTOR STARTUP, REACTOR OPERATION and REACTOR SHUTDOWN with HGAs in the reactor.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. 186 basin pump suction well level < 13.8 ft.	A.1 Be in REACTOR SHUTDOWN.	IMMEDIATELY
	<u>AND</u>	
	A.2 Initiate action to restore the 186 basin pump suction well level to ≥ 13.8 ft.	IMMEDIATELY
	<u>AND</u>	
	A.3 Initiate cooling water conservation measures.	IMMEDIATELY

(continued)

"DELETED VERSION"

INFORMATION ONLY

Ultimate Heat Sink
3.7.2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Cooling water temperature < 5°C	B.1 Align the pumps for recirculation to the 186 basin.	1 hour
	OR	
	B.2 Be in REACTOR SHUTDOWN AND B.3 Initiate action to restore the cooling water temperatures to $\geq 5^{\circ}\text{C}$.	90 minutes IMMEDIATELY
C. Cooling water temperature > 30°C	C.1 Be in REACTOR SHUTDOWN	90 minutes
D. All 186 basin sections or two 190 pumps suction wells not interconnected.	D.1 Initiate action to interconnect all 186 basin sections and two 190 pump suction wells.	IMMEDIATELY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	Verify the cooling water temperature is $\geq 5^{\circ}\text{C}$, and $\leq 30^{\circ}\text{C}$.	12 hours
SR 3.7.2.2	Verify 186 basin pump suction well level is ≥ 13.8 ft.	12 hours
SR 3.7.2.3	Verify that all three sections of the 186 basin and two 190 pump suction wells are interconnected.	7 days

"DELETED VERSION"

INFORMATION ONLY

Ultimate Heat Sink
3.7.2

CROSS-REFERENCES

TITLE	NUMBER
Cooling Water System Flow	3.7.1

"DELETED VERSION" INFORMATION ONLY

Control Rod Actuator Power Supply
3.8.1

3.8 ELECTRICAL SYSTEMS

3.8.1 Control Rod Actuator Power Supply

LC0 3.8.1

The following reactor shutdown systems power sources and related equipment shall be OPERABLE with:

A. The rod equipment room (RER) 125 volt floating emergency battery and one motor-generator set driving a motor-alternator set including the voltage and frequency control.

AND

B. Automatic Transfer Switches [REDACTED] and [REDACTED]

APPLICABILITY: REACTOR OPERATION, REACTOR STARTUP, and REACTOR SHUTDOWN when the actuator is latched and power is available to the control rod drive motor.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RER 125 volt floating emergency battery is inoperable.	A.1 Restore the inoperable RER 125 volt floating emergency battery to OPERABLE status.	8 hours
B. No RER motor-alternator set is being driven by a motor-generator set.	B.1 Restore at least one motor-alternator set driven by a motor-generator set to an OPERABLE in-service condition.	8 hours
C. [REDACTED] or [REDACTED] is inoperable.	C.1 Restore the inoperable ATS to OPERABLE status.	8 hours

(continued)

"DELETED VERSION"

"DELETED VERSION"

Control Rod Actuator Power Supply
3.8.1

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action(s) and associated Completion Time(s) for Conditions A, B or C not met. <u>OR</u> [REDACTED] and [REDACTED] are inoperable.	D.1 Be in REACTOR SHUTDOWN.	90 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify that the RER battery bus voltage is ≥ 129 and ≤ 131 vdc.	12 hours
SR 3.8.1.2	Verify that the RER 125 volt Floating Emergency Battery is not discharging.	12 hours
SR 3.8.1.3	Verify that the RER motor-generator/motor-alternator setup is correct.	12 hours
SR 3.8.1.4	Verify that the alternate power sources are available for [REDACTED] and [REDACTED].	12 hours
SR 3.8.1.5	Verify that automatic transfer switches [REDACTED] and [REDACTED] will transfer to their alternate power source upon loss of the primary power source.	6 months

(continued)

"DELETED VERSION"

INSTRUCTIONS ONLY

**Control Rod Actuator Power Supply
3.8.1**

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.6	Perform a battery discharge test on the RER battery.	18 months

CROSS-REFERENCES - None.

"DELETED VERSION"

Nuclear and Hydraulic Parameter Monitoring Power
3.8.2

3.8 ELECTRICAL SYSTEMS

3.8.2 Nuclear and Hydraulic Parameter Monitoring Power

LCO 3.8.2 A. A backup power supply independent of offsite (normal) power and capable of supplying required loads for 72 hours shall be available for monitoring the critical nuclear and hydraulic parameters listed below.

- a. Neutron Flux
- b. Reactor Period
- c. Bulk Moderator Temperature
- d. Assembly Effluent Temperature
- e. Cooling Water Temperature
- f. Plenum Inlet Temperature
- g. Reactor Tank Level

AND

B. Automatic Transfer Switches [REDACTED] and [REDACTED] shall be OPERABLE.

APPLICABILITY: Whenever the instrumentation required to monitor the above parameters is required to be OPERABLE.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required backup power source(s) are inoperable.	A.1 Restore backup power source(s) to an OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met during REACTOR OPERATION or REACTOR STARTUP. <u>OR</u> [REDACTED] and [REDACTED] are inoperable.	B.1 Be in REACTOR SHUTDOWN. "DELETED VERSION"	90 minutes

"DELETED VERSION"

Nuclear and Hydraulic Parameter Monitoring Power
3.8.2

INFORMATION ONLY

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. [REDACTED] is inoperable.	C.1 Restore the inoperable ATS to OPERABLE STATUS.	8 hours
D. Required Action and associated Completion Time of Condition A not met during REACTOR SHUTDOWN.	D.1 Suspend all operations involving POSITIVE REACTIVITY CHANGES. <u>OR</u> D.2 Suspend CHARGE AND DISCHARGE OPERATIONS.	IMMEDIATELY IMMEDIATELY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.2.1	Verify that the backup power supplies for monitoring of the nuclear and hydraulic parameters identified in the LCO are OPERABLE.	24 hours
SR 3.8.2.2	Verify that the alternate power sources are available for [REDACTED] and [REDACTED].	12 hours
SR 3.8.2.3	Verify that automatic transfer switches [REDACTED] and [REDACTED] will transfer to their alternate power source upon loss of the primary power source.	6 months

"DELETED VERSION"

Nuclear and Hydraulic Parameter Monitoring Power
3.8.2

CROSS-REFERENCES

TITLE	NUMBER
Assembly Effluent Temperatures - TECHNICAL (SAFETY) LIMITS	2.1
Assembly Effluent Temperatures - TRANSIENT LIMITS	3.1.5
Scram Circuit - Setpoints and Response Times	3.3.1
Neutron Flux Scram Circuit Protection	3.3.3
Neutron Monitoring Requirements - Shutdown (No C&D)	3.3.7.1
Neutron Monitoring Requirements - C&D Operations	3.3.7.2
Process Water System - Shutdown	3.4.3.3
Process Water System - Natural Circulation Bulk Moderator Temperature	3.4.3.4
Process Water System - Natural Circulation Reactor Level	3.4.3.5
Process Water System - Bulk Moderator Temperature Limits	3.4.4
Reactor Level	3.9.2
Charge - Discharge Operations - Process to Cooling Water ΔT	3.9.5

Emergency GM Diesel Generators - Operating
3.8.3.1

3.8 ELECTRICAL SYSTEMS

3.8.3.1 Emergency GM Diesel Generators - Operating

LCO 3.8.3.1 Two Emergency GM Diesel Generators and the associated 480 volt emergency power distribution system shall be OPERABLE.

APPLICABILITY: REACTOR STARTUP, REACTOR OPERATION and REACTOR SHUTDOWN, with shutdown time less than or equal to 3 hours.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One emergency GM diesel inoperable.	A.1 Restore emergency GM diesel to OPERABLE status.	72 hours
B. One 108 Building Emergency Switchgear inoperable.	B.1 Restore both 108 Emergency Switchgear to OPERABLE status.	72 hours
C. One Transformer Room Emergency Bus inoperable. <u>OR</u> Both Transformer Room Emergency Busses associated with the same emergency GM diesel inoperable.	C.1 Restore the inoperable Transformer Room Emergency Bus(es) to OPERABLE status.	72 hours

(continued)

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"DELETED VERSION" ONLY

Emergency GM Diesel Generators - Operating
3.8.3.1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY																												
SR 3.8.3.1.3	<p>Verify that control power is available for the following transformer room breakers and that they are in the position as listed:</p> <p>Transformer rooms [REDACTED]</p> <table><tr><td><u>Breaker #</u></td><td><u>Position</u></td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Open</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Open</td></tr></table> <p>Transformer rooms [REDACTED]</p> <table><tr><td><u>Breaker #</u></td><td><u>Position</u></td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Open</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Open</td></tr></table>	<u>Breaker #</u>	<u>Position</u>	[REDACTED]	Closed	[REDACTED]	Closed	[REDACTED]	Open	[REDACTED]	Closed	[REDACTED]	Closed	[REDACTED]	Open	<u>Breaker #</u>	<u>Position</u>	[REDACTED]	Closed	[REDACTED]	Closed	[REDACTED]	Open	[REDACTED]	Closed	[REDACTED]	Closed	[REDACTED]	Open	24 hours
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SR 3.8.3.1.4	<p>Verify that control power is available for the following GM diesel generator breakers and that they are in the position as listed:</p> <p>GM #1 Switchgear</p> <table><tr><td><u>Breaker #</u></td><td><u>Position</u></td></tr><tr><td>[REDACTED]</td><td>Open</td></tr><tr><td>[REDACTED]</td><td>Open</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr></table> <p>GM #2 Switchgear</p> <table><tr><td><u>Breaker #</u></td><td><u>Position</u></td></tr><tr><td>[REDACTED]</td><td>Open</td></tr><tr><td>[REDACTED]</td><td>Open</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr></table>	<u>Breaker #</u>	<u>Position</u>	[REDACTED]	Open	[REDACTED]	Open	[REDACTED]	Closed	[REDACTED]	Closed	[REDACTED]	Closed	<u>Breaker #</u>	<u>Position</u>	[REDACTED]	Open	[REDACTED]	Open	[REDACTED]	Closed	[REDACTED]	Closed	[REDACTED]	Closed	24 hours				
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Emergency GM Diesel Generators - Operating
3.8.3.1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.3.1.5	Verify that Station Batteries #1 and #2 are being charged by their Station Battery charger on float charge at ≥ 129 and ≤ 131 vdc or on equalize charge at ≥ 139 and ≤ 140 vdc.	24 hours
SR 3.8.3.1.6	Verify that the GM diesel generator starting batteries #1 and #2 are being charged by their MG set (MG #1) on float charge at ≥ 129 and ≤ 131 vdc or on equalize charge at ≥ 139 and ≤ 140 vdc.	24 hours
SR 3.8.3.1.7	Verify that the emergency GM diesel fuel oil day tank contains ≥ 460 gallons for each diesel generator when the unit is in the standby mode.	24 hours
SR 3.8.3.1.8	Verify that the lube oil tank level for each GM diesel generator contains ≥ 120 gallons (20 inches) with the diesel generator in standby mode, or contains ≥ 60 gallons (10 inches) with the diesel generator operating.	24 hours
SR 3.8.3.1.9	Verify that the jacket water expansion tank contains ≥ 20 gallons (20 inches) for each diesel generator.	24 hours
SR 3.8.3.1.10	Complete emergency GM diesel auto mode starting tests.	1 month
SR 3.8.3.1.11	Verify that [REDACTED] and the 125 VDC power ATS transfer to their alternate power source upon loss of the primary power source.	6 months

"DELETED VERSION"

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.3.1.12	Complete emergency GM diesel full load test.	every scheduled C&D outage

CROSS-REFERENCES

TITLE	NUMBER
Process Water System Bingham Pump DC Motors Emergency Cooling System (ECS) - Operating	3.4.7 3.5.1.1

3.8 ELECTRICAL SYSTEMS

3.8.3.2 Emergency GM Diesel Generators - Shutdown

LCO 3.8.3.2 One Emergency GM Diesel Generator and the associated 480 volt emergency power distribution system shall be OPERABLE.

APPLICABILITY: REACTOR SHUTDOWN with HGAs in the reactor and shutdown time greater than or equal to 3 hours.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. No emergency GM diesels are OPERABLE.</p> <p><u>OR</u></p> <p>Both 108 Building Emergency Switchgear are inoperable.</p> <p><u>OR</u></p> <p>All four Transformer Room Emergency Busses are inoperable.</p>	<p>A.1 Suspend CHARGE & DISCHARGE OPERATIONS.</p>	IMMEDIATELY
	<p><u>AND</u></p> <p>A.2 Initiate action to restore at least one emergency GM diesel to an OPERABLE status.</p>	IMMEDIATELY
	<p><u>AND</u></p> <p>A.3 Initiate action to restore the 108 Emergency Switchgear to OPERABLE status.</p>	IMMEDIATELY
	<p><u>AND</u></p> <p>A.4 Initiate Action to restore the Transformer Room Emergency Bus(es) associated with the OPERABLE emergency GM diesel to OPERABLE status.</p>	IMMEDIATELY

"DELETED VERSION"

INFORMATION ONLY

**Emergency GM Diesel Generators - Shutdown
3.8.3.2**

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY																												
SR 3.8.3.2.1	Verify that the emergency GM diesel auto start setup is correct.	12 hours																												
SR 3.8.3.2.2	Verify that the transformer room volt meter reading is ≥ 460 and ≤ 490 volts.	24 hours																												
SR 3.8.3.2.3	<p>Verify that control power is available for the following transformer room breakers associated with each OPERABLE emergency GM diesel(s) and that they are in the position as listed:</p> <p>Transformer rooms [REDACTED]</p> <table><tr><td><u>Breaker #</u></td><td><u>Position</u></td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Open</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Open</td></tr></table> <p>Transformer rooms [REDACTED]</p> <table><tr><td><u>Breaker #</u></td><td><u>Position</u></td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Open</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Open</td></tr></table>	<u>Breaker #</u>	<u>Position</u>	[REDACTED]	Closed	[REDACTED]	Closed	[REDACTED]	Open	[REDACTED]	Closed	[REDACTED]	Closed	[REDACTED]	Open	<u>Breaker #</u>	<u>Position</u>	[REDACTED]	Closed	[REDACTED]	Closed	[REDACTED]	Open	[REDACTED]	Closed	[REDACTED]	Closed	[REDACTED]	Open	24 hours
<u>Breaker #</u>	<u>Position</u>																													
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"DELETED VERSION"

"DELETED VERSION"

INFORMATION ONLY

Emergency GM Diesel Generators - Shutdown
3.8.3.2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY																
SR 3.8.3.2.4	<p>Verify that control power is available for the following GM diesel generator breakers associated with each OPERABLE emergency GM diesel(s) and that they are in the position as listed:</p> <p>GM #1 Switchgear</p> <table><tr><td><u>Breaker #</u></td><td><u>Position</u></td></tr><tr><td rowspan="5"> </td><td>Open</td></tr><tr><td>Open</td></tr><tr><td>Closed</td></tr><tr><td>Closed</td></tr><tr><td>Closed</td></tr></table> <p>GM #2 Switchgear</p> <table><tr><td><u>Breaker #</u></td><td><u>Position</u></td></tr><tr><td rowspan="5"> </td><td>Open</td></tr><tr><td>Open</td></tr><tr><td>Closed</td></tr><tr><td>Closed</td></tr><tr><td>Closed</td></tr></table>	<u>Breaker #</u>	<u>Position</u>		Open	Open	Closed	Closed	Closed	<u>Breaker #</u>	<u>Position</u>		Open	Open	Closed	Closed	Closed	24 hours
<u>Breaker #</u>	<u>Position</u>
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<u>Breaker #</u>	<u>Position</u>
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SR 3.8.3.2.5	<p>Verify that Station Battery associated with each OPERABLE emergency GM diesel(s) is being charged by its Station Battery charger on float charge at ≥ 129 and ≤ 131 vdc or on equalize charge at ≥ 139 and ≤ 140 vdc.</p>	24 hours																
SR 3.8.3.2.6	<p>Verify that the OPERABLE GM diesel generator starting battery is being charged by its MG set (MG #1) on float charge at ≥ 129 and ≤ 131 vdc or on equalize charge at ≥ 139 and ≤ 140 vdc.</p>	24 hours																
SR 3.8.3.2.7	<p>Verify that the emergency GM diesel fuel oil day tank contains ≥ 460 gallons when the unit is in the standby mode.</p>	24 hours																

"DELETED VERSION"

(continued)

"DELETED VERSION"

INTERNAL USE ONLY

Emergency GM Diesel Generators - Shutdown
3.8.3.2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.3.2.8	Verify that the emergency GM diesel lube oil tank contains ≥ 120 gallons (20 inches) with the diesel generator in the standby mode, or contains ≥ 60 gallons (10 inches) diesel generator operating.	24 hours
SR 3.8.3.2.9	Verify that the emergency GM diesel jacket water expansion tank contains ≥ 20 gallons (20 inches).	24 hours
SR 3.8.3.2.10	Complete emergency GM diesel auto mode starting test.	1 month
SR 3.8.3.2.11	Verify that [REDACTED] and the 125 VDC Power AIS transfer to their alternate power source upon loss of the primary power source.	6 months
SR 3.8.3.2.12	Complete emergency GM diesel full load test.	every scheduled C&D outage

CROSS-REFERENCES

TITLE	NUMBER
Process Water System Bingham Pump DC Motors Emergency Cooling System (ECS) - Shutdown	3.4.7 3.5.1.2

"DELETED VERSION"

3.8 ELECTRICAL SYSTEMS

3.8.3.3 Emergency GM Diesel Generators - Movements of HGAs

LCO 3.8.3.3 Two Emergency GM Diesel Generators and the associated 480 volt emergency power distribution system shall be OPERABLE.

APPLICABILITY: During the CHARGE AND DISCHARGE OPERATIONS of HGAs.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One emergency GM diesel inoperable.</p> <p><u>OR</u></p> <p>Both emergency GM diesels inoperable.</p> <p><u>OR</u></p> <p>One 108 Building Emergency Switchgear inoperable.</p> <p><u>OR</u></p> <p>One or more Transformer Room Emergency Bus(es) inoperable.</p> <p><u>OR</u></p> <p>(continued)</p>	<p>-----NOTE----- Performance of A.1 activities shall not preclude completion of actions to establish a safe conservative condition. -----</p> <p>A.1 Suspend CHARGE AND DISCHARGE OPERATIONS.</p>	IMMEDIATELY

(continued)

ONLY

Emergency GM Diesel Generators - Movements of HGAs
3.8.3.3

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Both Transformer Room Emergency Busses associated with the same emergency GM diesel inoperable. <u>OR</u> Both 108 Building Emergency Switchgear inoperable.		

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.3.1	Verify that the emergency GM diesel auto start setup is correct.	12 hours
SR 3.8.3.3.2	Verify that the transformer room volt meter reading is ≥ 460 and ≤ 490 volts.	24 hours

(continued)

"DELETED VERSION"

Emergency GM Diesel Generators - Movements of HGAs
3.8.3.3

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY																												
SR 3.8.3.3.3	<p>Verify that control power is available for the following transformer room breakers and that they are in the position as listed:</p> <p>Transformer rooms [REDACTED] & [REDACTED]</p> <table><tr><td><u>Breaker #</u></td><td><u>Position</u></td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Open</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Open</td></tr></table> <p>Transformer rooms [REDACTED] & [REDACTED]</p> <table><tr><td><u>Breaker #</u></td><td><u>Position</u></td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Open</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Open</td></tr></table>	<u>Breaker #</u>	<u>Position</u>	[REDACTED]	Closed	[REDACTED]	Closed	[REDACTED]	Open	[REDACTED]	Closed	[REDACTED]	Closed	[REDACTED]	Open	<u>Breaker #</u>	<u>Position</u>	[REDACTED]	Closed	[REDACTED]	Closed	[REDACTED]	Open	[REDACTED]	Closed	[REDACTED]	Closed	[REDACTED]	Open	24 hours
<u>Breaker #</u>	<u>Position</u>																													
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[REDACTED]	Open																													
SR 3.8.3.3.4	<p>Verify that control power is available for the following GM diesel generator breakers and that they are in the position as listed:</p> <p>GM #1 Switchgear</p> <table><tr><td><u>Breaker #</u></td><td><u>Position</u></td></tr><tr><td>[REDACTED]</td><td>Open</td></tr><tr><td>[REDACTED]</td><td>Open</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr></table> <p>GM #2 Switchgear</p> <table><tr><td><u>Breaker #</u></td><td><u>Position</u></td></tr><tr><td>[REDACTED]</td><td>Open</td></tr><tr><td>[REDACTED]</td><td>Open</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr><tr><td>[REDACTED]</td><td>Closed</td></tr></table>	<u>Breaker #</u>	<u>Position</u>	[REDACTED]	Open	[REDACTED]	Open	[REDACTED]	Closed	[REDACTED]	Closed	[REDACTED]	Closed	<u>Breaker #</u>	<u>Position</u>	[REDACTED]	Open	[REDACTED]	Open	[REDACTED]	Closed	[REDACTED]	Closed	[REDACTED]	Closed	24 hours				
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"DELETED VERSION"

"DELETED VERSION"

INFORMATION ONLY

Emergency GM Diesel Generators - Movements of HGAs
3.8.3.3

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.3.3.5	Verify that Station Batteries #1 and #2 are being charged by their Station Battery charger on float charge at ≥ 129 and ≤ 131 vdc or on equalize charge at ≥ 139 and ≤ 140 vdc.	24 hours
SR 3.8.3.3.6	Verify that the GM diesel generator starting batteries #1 and #2 are being charged by their MG set (MG #1) on float charge at ≥ 129 and ≤ 131 vdc or on equalize charge at ≥ 139 and ≤ 140 vdc.	24 hours
SR 3.8.3.3.7	Verify that the emergency GM diesel fuel oil day tank contains ≥ 460 gallons for each diesel generator when the unit is in the standby mode.	24 hours
SR 3.8.3.3.8	Verify that the lube oil tank level for each GM diesel generator contains ≥ 120 gallons (20 inches) with the diesel generator in standby mode, or contains ≥ 60 gallons (10 inches) with the diesel generator operating.	24 hours
SR 3.8.3.3.9	Verify that the jacket water expansion tank contains ≥ 20 gallons (20 inches) for each diesel generator.	24 hours
SR 3.8.3.3.10	Complete emergency GM diesel auto mode starting tests.	1 month
SR 3.8.3.3.11	Verify that [REDACTED] and the 125 VDC Power AIS transfer to their alternate power source upon loss of the primary power source.	6 months

"DELETED VERSION"

INFORMATION ONLY

Emergency GM Diesel Generators - Movements of HGAs 3.8.3.3

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.3.3.12	Complete emergency GM diesel full load test.	every scheduled C&D outage

CROSS REFERENCES

Emergency Cooling System (ECS) - Shutdown	3.5.1.2
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"DELETED VERSION"

INFORMATION ONLY

Airborne Activity Confinement System Power
3.8.4

3.8 ELECTRICAL SYSTEMS

3.8.4 Airborne Activity Confinement System Power

LCO 3.8.4 Two [REDACTED] fan emergency diesel generators shall be OPERABLE.

APPLICABILITY: REACTOR STARTUP, REACTOR OPERATION and REACTOR SHUTDOWN with HGAs in the process room or the reactor tank.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One [REDACTED] fan emergency diesel generator inoperable.	A.1 Restore two [REDACTED] fan emergency diesel generators to OPERABLE status.	72 hours
B. Diesel fuel oil tank contains < 7200 gallons.	B.1 Restore volume in diesel fuel oil tank to $\geq$ 7200 gallons.	7 days
C. Diesel lube oil tank contains < 52 gallons.	C.1 Restore volume in lube oil tank to $\geq$ 52 gallons.	7 days

(continued)

"DELETED VERSION"

INFORMATION ON LINE

3.8.4

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>D. Required Action A, B, or C and associated Completion Time not met.</p> <p><u>OR</u></p> <p>Two [REDACTED] fan emergency diesel generators inoperable.</p>	<p>D.1 Be in REACTOR SHUTDOWN.</p> <p><u>AND</u></p> <p>-----NOTE----- Performance of D.2 activities shall not preclude completion of actions to establish a safe conservative condition. -----</p> <p>D.2 Suspend all operations involving POSITIVE REACTIVITY CHANGES.</p>	<p>90 minutes</p> <p>IMMEDIATELY</p>

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify that the standby line-up for the [REDACTED] fan emergency diesel generators is correct.	24 hours
SR 3.8.4.2	Perform [REDACTED] EGs functional test.	7 days
SR 3.8.4.3	Verify that the diesel fuel oil tank contains ≥ 7200 gallons.	7 days
SR 3.8.4.4	Verify that the diesel lube oil tank contains ≥ 52 gallons.	7 days

**Savannah River Site
K Production Reactor**

Rev. 8,3

INTENTIONALLY ONLY

Airborne Activity Confinement System Power
3.8.4

CROSS-REFERENCES

TITLE	NUMBER
Airborne Activity Confinement System - HGAS	3.6.1

3.9 COMPONENTS HANDLING

3.9.1 Irradiated Assemblies

LCO 3.9.1 No HGA shall be discharged from the reactor until the time after shutdown is greater than or equal to the limit defined in the CORE SHUTDOWN REPORT.

APPLICABILITY: During the charge and discharge of HGAs

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. An HGA is discharged from the reactor prior to the time limit.</p> <p><u>OR</u></p> <p>A failed HGA is discharged from the reactor prior to the time limit.</p>	<p>-----NOTE----- Performance of A.1 activities shall not preclude completion of actions to establish a safe conservative condition. -----</p> <p>A.1 Suspend discharge operations.</p>	<p>IMMEDIATELY</p>

INFORMATION ONLY

Irradiated Assemblies
3.9.1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1	Verify that the time after shutdown is $\geq$ the limit defined in the CORE SHUTDOWN REPORT.	Prior to discharge of HGAs AND Prior to discharge of failed HGAs

CROSS-REFERENCES - None.

INFORMATION ONLY

Reactor Level
3.9.2

3.9 COMPONENTS HANDLING

3.9.2 Reactor Level

LCO 3.9.2 The reactor level shall be $\geq$ overflow.

APPLICABILITY: CHARGE AND DISCHARGE OPERATIONS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor level < overflow.	-----NOTE----- Performance of A.1 activities shall not preclude completion of actions to establish a safe conservative condition. ----- A.1 Suspend CHARGE AND DISCHARGE OPERATIONS.	IMMEDIATELY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.2.1	Verify that the reactor level is $\geq$ overflow.	12 hours

CROSS-REFERENCES - None.

3.9 COMPONENTS HANDLING

3.9.3 Fissile Components - Handling and Storage

LCO 3.9.3

- A. No more than 4 assemblies shall be permitted in a process or handling zone at the same time.
- B. The storage configurations of FISSILE MATERIAL in the reactor area shall be limited to arrays of undamaged assemblies of the same assembly types and to those configurations permitted in Tables 3.9.3-1 through -5, except that:
 - 1. MARK 16B and MARK 22 fuel tubes and assemblies may be stored in borated concrete racks if the presence of ≥ 0.25 wt% boron in the concrete is confirmed and if the assemblies contain no more than 480 g ^{235}U /ft.
- C. A separation of at least 12 inches shall be provided between groups of fissile targets and groups of fuel tubes under water.

-----NOTES-----

No configuration limits are placed on uranium targets provided the equivalent enrichment does not exceed 0.65%, where equivalent enrichment is defined as $\text{wt}\% \text{ } ^{235}\text{U} + 2.0$ times $\text{wt}\% \text{ Pu}$.

No configuration limits are placed on neptunium oxide targets provided the Al/NpO_2 molecular ratio is greater than 7 ($\leq 60\% \text{ NpO}_2$ by weight).

APPLICABILITY: During storage, handling and processing of FISSILE MATERIAL in the 105 Buildings outside the nuclear reactors.

Fissile Components - Handling and Storage
3.9.3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. More than 4 assemblies exist in a process or handling area at the same time.	A.1 Stop operations in the affected area(s).	IMMEDIATELY
	<u>AND</u> A.2 Initiate action to perform an engineering evaluation to determine the corrective actions.	IMMEDIATELY
B. Fissile components are stored or exist in configurations not permitted by Tables 3.9.3-1 through 3.9.3-5.	B.1 Stop operations in the affected area(s).	IMMEDIATELY
	<u>AND</u> B.2 Initiate action to perform an engineering evaluation to determine the corrective actions.	IMMEDIATELY
C. Borated rack boron content < 0.25 wt%.	C.1 Stop movement of fuel into or out of the borated racks.	IMMEDIATELY
	<u>AND</u> C.2 Initiate action to perform an engineering evaluation to determine the corrective actions.	IMMEDIATELY
D. 70 ton cask cadmium plates fail acceptance criteria. <u>OR</u> 70 ton cask cadmium plates not installed.	D.1 Stop operations in the affected area(s).	IMMEDIATELY
	<u>AND</u> D.2 Initiate action to perform an engineering evaluation to determine the corrective actions.	IMMEDIATELY

Fissile Components - Handling and Storage
3.9.3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Verify that no more than 4 assemblies exist in a process or handling area at the same time.	During movement or processing of FISSILE MATERIAL.
SR 3.9.3.2	Verify that FISSILE MATERIAL stored in the reactor area is stored in configurations permitted by Tables 3.9.3-1 through 3.9.3-5.	1 month <u>AND</u> During movement or processing of FISSILE MATERIAL.
SR 3.9.3.3	Verify that the cadmium plates are installed in the 70 ton cask.	Prior to performance of SR 3.9.3.4. <u>AND</u> Prior to use of the 70 ton cask for fuel movement.
SR 3.9.3.4	Perform a neutron transmission test on the 70 ton cask cadmium plates to determine cadmium thickness reduces neutron transmission to $\leq 50\%$.	Within 12 months prior to use.
SR 3.9.3.5	Perform a neutron transmission test on the borated storage racks to determine boron content. ≥ 0.25 wt%.	60 months

CROSS-REFERENCES - None.

ONLY

Fissile Components - Handling and Storage
3.9.3

Table 3.9.3-1

Maximum Number of Fuel Tubes in Isolated Arrays
(Note a)

FUEL TUBE	FUEL TYPE, MARK	
	22	16B
Whole assemblies (Note b)	4	4
Inner tubes only	7	12
Middle Tubes only	-	7
Outer tubes only	7	11

NOTES

- a. The number of assemblies or components are critically safe in an isolated grouping of any geometrical arrangement and array, whether moderated or not, as long as each grouping is isolated from another grouping by at least 12 inches of water or at least 3 feet of air.
- b. With or without targets.

Table 3.9.3-2

Limits for Final Storage
(Notes a and b)

Fuel Type Mark	Density Square Pitch	Mass
22	≥ 12.0 in.	≤ 924 assemblies
16B	≥ 12.0 in.	≤ 924 assemblies

NOTES

- a. These limits are for a single isolated array. A 20 ft separation is sufficient to isolate an array.
- b. No more than 10 persons are permitted in Final Storage with FISSILE MATERIAL present.

Table 3.9.3-3

Vertical Tube Storage
(Note a)

Fuel Type Mark	Average Square Pitch in inches	Maximum Permissible Number in a Cluster at Smaller Pitch	Minimum Permissible Square Pitch in Cluster inches
22	≥ 6.0	4	5.5
	≥ 8.0	4	3.2
16B	≥ 6.5	4	5.8
	≥ 8.0	4	3.7

NOTE

- a. Immersed in water with no voids. No limit to size of array.

INFORMATION ONLY

Table 3.9.3-4

Limits for Horizontal Storage

Fuel Type Mark	Average Pitch Between Rows inches	Height of Row in Assemblies
22	≥ 12.0	≤ 8
16B	≥ 12.0	≤ 8

Table 3.9.3-5

Limits for 70-Ton Cask

Fuel Type Mark	Cask Positions Permitted	Height Per Position in Assemblies
22	1, 5	≤ 4
16B	1, 5	≤ 4

-----NOTE-----
All four separator plates shall be installed
and compartments 2, 3 and 4 shall be filled
with water in each cask.

Disassembly Basin Operation - Water Temperature
3.9.4.1

3.9 COMPONENTS HANDLING

INFORMATION ONLY

3.9.4.1 Disassembly Basin Operation - Water Temperature

LCO 3.9.4.1 The temperature of the water in the disassembly basins shall be $\leq 45^{\circ}\text{C}$.

APPLICABILITY: Whenever the reactor and/or the disassembly basins contain HGAs.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Machine basin temperature. > 45°C .	A.1 Initiate action to establish cooling.	IMMEDIATELY
<u>OR</u>	<u>AND</u>	
Vertical tube storage (VTS) basin temperature > 45°C .	-----NOTE----- Performance of A.2 activities shall not preclude completion of actions to establish a safe conservative condition. -----	
	A.2 Stop discharge of assemblies to the disassembly basins.	IMMEDIATELY

INFORMATION ONLY

Disassembly Basin Operation - Water Temperature
3.9.4.1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.4.1.1 Verify machine basin and vertical tube storage basin temperatures $\leq 45^{\circ}\text{C}$.	12 hours during discharge operations if basin temperatures $\geq 36^{\circ}\text{C}$ <u>OR</u> 24 hours, if basin temperatures $< 36^{\circ}\text{C}$

CROSS-REFERENCES - None.

Disassembly Basin Operation - Water Level

3.9.4.2 Disassembly Basin Operation - Water Level

APPLICABILITY: Whenever the reactor or the disassembly basin contains HGAs or irradiated assemblies.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Disassembly basin level > -11.5 inches.	A.1 Initiate action to restore level within limits.	IMMEDIATELY
B. During REACTOR SHUTDOWN: Disassembly basin level < -18.0 inches.	<p>-----NOTE----- Performance of C.1 activities shall not preclude completion of actions to establish a safe conservative condition. -----</p> <p>B.1 Suspend CHARGE AND DISCHARGE OPERATIONS.</p> <p><u>AND</u></p> <p>B.2 Initiate action to restore basin level to $\geq$ -18.0 inches.</p>	<p>IMMEDIATELY</p> <p>IMMEDIATELY</p>

(continued)

INFORMATION ONLY

Disassembly Basin Operation - Water Level 3.9.4.2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. During REACTOR STARTUP and REACTOR OPERATION: Disassembly basin level < -18.0 inches.	C.1 Restore basin level to $\geq$ -18.0 inches.	2 hours
D. Required Action for Condition C not met within required Completion Time.	D.1 Be in REACTOR SHUTDOWN.	90 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.4.2.1 Verify machine basin and VTS level between -11.5 inches and -18.0 inches.	12 hours during discharge operations if basin is outside the -13.5 to -16.5 inch interval <u>OR</u> 24 hours if basin level is between the -13.5 and -16.5 inch interval

CROSS-REFERENCES - None.

INFORMATION ONLY

CHARGE AND DISCHARGE OPERATIONS - Process Water to Cooling Water AT 3.9.5

3.9 COMPONENTS HANDLING

3.9.5 CHARGE AND DISCHARGE OPERATIONS - Process Water to Cooling Water AT

LCO 3.9.5

The process water plenum inlet temperature shall be $\leq 5^{\circ}\text{C}$ above the cooling water temperature during forced circulation of moderator.

OR

The bulk moderator temperature shall be $\leq 5^{\circ}\text{C}$ above the cooling water temperature during no forced circulation of the moderator.

APPLICABILITY: During CHARGE AND DISCHARGE OPERATIONS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. With forced circulation of process water, and the plenum inlet temperature is $> 5^{\circ}\text{C}$ above the temperature of the cooling water supply to the heat exchangers.	-----NOTE----- Performance of A.1 activities shall not preclude completion of actions to establish a safe conservative condition. ----- A.1 Suspend CHARGE AND DISCHARGE OPERATIONS.	IMMEDIATELY
B. With no forced circulation, the bulk moderator temperature is $> 5^{\circ}\text{C}$ above the temperature of the cooling water supply to the heat exchangers.	-----NOTE----- Performance of B.1 activities shall not preclude completion of actions to establish a safe conservative condition. ----- B.1 Suspend CHARGE AND DISCHARGE OPERATIONS.	IMMEDIATELY

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CHARGE AND DISCHARGE OPERATIONS - Process Water to Cooling Water ΔT
3.9.5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify that the plenum inlet temperature of the process water $\leq 5^{\circ}\text{C}$ above cooling water supply header temperature.	6 hours with forced circulation
SR 3.9.5.2	Verify that the bulk moderator temperature is $\leq 5^{\circ}\text{C}$ above cooling water supply header temperatures.	6 hours with no forced circulation

CROSS-REFERENCES

TITLE	NUMBER
Process Water System - Shutdown	3.4.3.3

3.9 COMPONENTS HANDLING

3.9.6 Discharge Machine Water Sources

LCO 3.9.6 Two sources of H₂O, one primary and one secondary, and one source of D₂O, either a primary or a secondary shall be available. Each primary source shall be capable of providing ≥ 8 gpm and each secondary source shall be capable of providing ≥ 13 gpm.

AND

The plenum inlet, pump suction, and vertical tube storage temperatures shall be $\leq$ the limits defined in the CORE SHUTDOWN REPORT.

APPLICABILITY: Whenever the discharge machine is handling HGAs.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. Less than two sources of H₂O are available</p> <p>OR</p> <p>Less than one source of D₂O is available.</p>	<p>-----NOTE----- Performance of A.1 activities shall not preclude completion of actions to establish a safe conservative condition. -----</p> <p>A.1 Suspend discharge operations.</p>	IMMEDIATELY
<p>B. The plenum inlet, pump suction, or vertical tube storage (VTS) temperature is $>$ the limits defined in the CORE SHUTDOWN REPORT.</p>	<p>-----NOTE----- Performance of B.1 activities shall not preclude completion of actions to establish a safe conservative condition. -----</p> <p>B.1 Suspend discharge operations.</p>	IMMEDIATELY

IN: ONLY

Discharge Machine Water Sources
3.9.6

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.9.6.1 Verify that the required D₂O and H₂O sources are capable of supplying a primary flow $\geq$ 8 gpm through the load rod path and a secondary flow $\geq$ 13 gpm through the seal sleeve path.</p>	<p>Prior to initial discharge of the HGAs.</p> <p><u>AND</u></p> <p>Following a 24 hour delay between the discharge of HGAs.</p> <p><u>AND</u></p> <p>Once every 7 days during continuous discharge of HGAs.</p>
<p>SR 3.9.6.2 Verify that plenum inlet, pump suction, and vertical tube storage temperatures are $\leq$ the limits defined in the CORE SHUTDOWN REPORT.</p>	<p>Every 30 minutes when D machine cooling wait time exceeds horizontal in-basin wait time</p>

CROSS-REFERENCES - None.

ONLY

Charge and Discharge Equipment Power 3.9.8

3.9 COMPONENTS HANDLING

3.9.8 Charge and Discharge Equipment Power

LCO 3.9.8 The following power sources and related equipment shall be OPERABLE:

- a. Rod equipment room (RER) 125 volt floating emergency battery and one motor generator set driving a motor-alternator set including the automatic voltage and frequency control.

AND

- b. DC control power for the supply breakers to the four transformer rooms.

AND

- c. The secondary H₂O pump automatic transfer switch (ATS).

APPLICABILITY: During the charge and discharge of HGAs

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Secondary H ₂ O pump ATS inoperable.	A.1 Restore ATS to OPERABLE status.	8 hours
B. RER 125 volt floating emergency battery inoperable.	B.1 Restore RER 125 volt floating emergency battery to OPERABLE status.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. No RER motor-alternator set is being driven by a motor-generator set.	C.1 Restore at least one motor-alternator set driven by a motor-generator set to an OPERABLE in-service condition.	8 hours
D. DC control power not available to one or more breakers in one or more transformer rooms.	D.1 Restore transformer room breakers to OPERABLE status.	4 hours
E. Required Action(s) and associated Completion Time(s) of Conditions A, B, C, or D not met.	<p>-----NOTE----- Performance of E.1 and E.2 activities shall not preclude completion of actions to establish a safe conservative condition. -----</p> <p>E.1 Suspend CHARGE AND DISCHARGE OPERATIONS.</p> <p><u>AND</u></p> <p>E.2 Suspend operations involving POSITIVE REACTIVITY CHANGES.</p>	<p>IMMEDIATELY</p> <p>IMMEDIATELY</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.8.1 Verify that the RER battery bus voltage is ≥ 129 and ≤ 131 vdc.	12 hours

(continued)

ONLY

Charge and Discharge Equipment Power
3.9.8

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.9.8.2	Verify that the alternate power source is available for the secondary H ₂ O pump ATS.	12 hours
SR 3.9.8.3	Verify that one motor-generator set is driving a motor-alternator set including the voltage and frequency control.	12 hours
SR 3.9.8.4	Verify DC control power to the breakers in each of the transformer rooms.	24 hours
SR 3.9.8.5	Perform Automatic Transfer Switch (ATS) functional checks for secondary H ₂ O pump backup power.	6 months <u>AND</u> Within 7 days prior to discharge of HGAs

CROSS-REFERENCES

TITLE	NUMBER
Control Rod Actuator Power Supply	3.8.1

Emergency Cooling Water Pump Well
3.9.9

3.9 COMPONENTS HANDLING

3.9.9 Emergency Cooling Water (ECW) Pump Well

LCO 3.9.9 The ECW pump well shall not be isolated from the machine basin.

APPLICABILITY: Whenever the reactor contains HGAs

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ECW pump well machine basin sluice gate(s) are closed.	A.1 Open sluice gate(s).	IMMEDIATELY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.9.1	Verify that the emergency pump well machine basin sluice gates are open.	Prior to cooling water startup <u>AND</u> 24 hours

CROSS-REFERENCES - None.

INFORMATION ONLY

3.10 SPECIAL TEST EXCEPTIONS

3.10.1 Safety Rod Testing

LCO 3.10.1 During the performance and upon successful completion of safety rod testing per SR 3.1.2.3 or SR 3.1.2.4, the reactor shall be considered in REACTOR STARTUP for all APPLICABILITY determinations provided:

- a. The control rods remain FULLY INSERTED;
- b. The control rod drive power supplies are de-energized; and
- c. Total test completion time does not exceed 7 days.

APPLICABILITY: During the performance of safety rod testing per SR 3.1.2.3 or SR 3.1.2.4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. All full length and all partial length control rods <u>not</u> FULLY INSERTED.	A.1 Fully insert all safety rods.	IMMEDIATELY
B. All control rod drive power supplies <u>not</u> de-energized.	B.1 Verify that all control rods are FULLY INSERTED and de-energize all control rod drive power supplies.	IMMEDIATELY
C. Total test completion time > 7 days.	C.1 Be in REACTOR SHUTDOWN.	90 minutes

(continued)

INFORMATION ONLY

3.10 SPECIAL TEST EXCEPTIONS

3.10.1 Safety Rod Testing

LCO 3.10.1 During the performance and upon successful completion of safety rod testing per SR 3.1.2.3 or SR 3.1.2.4, the reactor shall be considered in REACTOR STARTUP for all APPLICABILITY determinations provided:

- a. The control rods remain FULLY INSERTED;
- b. The control rod drive power supplies are de-energized;
and
- c. Total test completion time does not exceed 7 days.

APPLICABILITY: During the performance of safety rod testing per SR 3.1.2.3 or SR 3.1.2.4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. All full length and all partial length control rods <u>not</u> FULLY INSERTED.	A.1 Fully insert all safety rods.	IMMEDIATELY
B. All control rod drive power supplies <u>not</u> de-energized.	B.1 Verify that all control rods are FULLY INSERTED and de-energize all control rod drive power supplies.	IMMEDIATELY
C. Total test completion time > 7 days.	C.1 Be in REACTOR SHUTDOWN.	90 minutes

(continued)

INFORMATION ONLY

Safety Rod Testing
3.10.1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.10.1.1	Verify that all full length and all partial length control rods are FULLY INSERTED.	Within 1 hour prior to initiating safety rod testing per SR 3.1.2.3 or SR 3.1.2.4 <u>AND</u> 12 hours
SR 3.10.1.2	Verify that the control rod drive power supplies are de-energized.	Within 1 hour prior to initiating safety rod testing per SR 3.1.2.3 or SR 3.1.2.4 <u>AND</u> 12 hours
SR 3.10.1.3	Verify that safety rod testing has been terminated or completed per the requirements of SR 3.1.2.3 or SR 3.1.2.4.	Within 7 days after initiating safety rod testing per SR 3.1.2.3 or SR 3.1.2.4

CROSS-REFERENCES

TITLE	NUMBER
Safety Rod OPERABILITY	3.1.2

4.0 DESIGN FEATURES

4.1.1 Heat Flux Limits

The calculated operating heat flux for the Safety Rods, Control Rods and Sparger Inserts shall be within the limits defined in the COR.

4.1.2 Lithium-to-Tritium Atom Ratio

This design feature specifies that for all lithium bearing components in the reactor, at the location of maximum lithium burnup, the number of lithium atoms must always be $\geq$ the number of tritium atoms. The lithium-to-tritium atom ratio shall be calculated from the following equation:

$$\text{Li/T} = \text{g mol Li present/g mol } ^6\text{Li burned} \geq 1.0.$$

4.1.3 Gas Volume Ratio (GVR)

This design feature specifies that for all lithium bearing components in the reactor, at the location of maximum lithium burnup, the GVR be ≤ 130 . The GVR shall be calculated from the following equation:

$$\text{GVR} = (1.5) (\text{g mol } ^6\text{Li burned}) (22,400)/\text{core volume, cc.}$$

4.1.4 Control and Safety Computers

a.

The maximum control rod withdrawal rate for a single control rod shall be ≤ 550 vu/min

b.

The maximum control rod withdrawal rate for a gang shall be ≤ 92.5 vu/min.

c.

Neither the control computer system nor the safety computer system shall adversely affect signals or the processing of signals in the other system. Neither computer system shall significantly reduce the protection provided by any other safety system.

(continued)

4.0 DESIGN FEATURES

4.1.4 Control and Safety Computers (continued)**d.**

Safety and control computer systems shall be designed and programmed with internal testing and auditing to automatically detect and annunciate failures of surveillance, data processing, and protective action capabilities.

4.1.5 Heat Exchanger Heat Removal Capacity

Sufficient cooling water flow shall be provided to the PWS heat exchangers to remove decay heat from the reactor under normal conditions in the event of a loss of normal power.

4.1.6 Ultimate Strength

The ultimate strength of any stressed component member, determined at the maximum calculated surface temperature, shall be at least a factor of 3 greater than the operating stress.

4.1.7 Fissile Components**a. Composition**

Fissile components shall meet the composition limits specified in Table 4.0-1, during storage, handling and processing of FISSILE MATERIAL in the 105 Buildings.

b. Storage Facility

The Storage Facility for fissile components shall be designed to ensure that the criticality limits described in DPSTM-100 CRIT, Technical Manual for 100-Area Criticality Safety, will not be exceeded in normal operation or in the event of credible contingencies.

4.1.8 Temperature Coefficients

The temperature coefficients shall be within the limits defined in Technical Procedure TP-90-012, "Reactor Core Reload Physics Calculations."

(continued)

4.0 DESIGN FEATURES (continued)

4.1.9 Shutdown Charge Reactivity

The maximum reactivity of a fully loaded Mark 22 charge in the reactor shall be less than a k_{eff} of 0.979 with:

- a. The reactor level $\geq$ 19 feet 1 3/8 inches measured from the bottom of the reactor tank, and
 - b. All control and safety rods FULLY INSERTED.
-

Table 4.0-1¹

Assembly Material Specifications

VARIABLE	ASSEMBLY TYPE		
	MK22	MK16B	MK16B HEAVY
^{235}U conc. (g/ft)	≤ 281.6	≤ 316	≤ 335
$^{236}\text{U}/^{235}\text{U}$	≥ 0.09	≥ 0.19	≥ 0.28
$^{239}\text{Pu} + ^{241}\text{Pu}$ conc. (g/ft)	0	0	0

NOTE:

1. The contents specified in this table are the as-manufactured contents of the unirradiated assemblies and are the contents used for the criticality analyses referenced in Technical Specification 4.1.7.

5.0 ADMINISTRATIVE CONTROLS

5.1 INTRODUCTION

This K-Reactor Administrative Control Technical Specification supersedes the Generic Reactors Administrative Technical Specification (DPST-TS-105-5.0) for K-Reactor. It specifies the administrative framework for safe operation of the Savannah River Site's (SRS) K-Reactor. It also provides an overview of the principal control documents used to maintain safe reactor operations.

The Westinghouse Savannah River Company (WSRC) is a full-scope, self-sufficient, site management and operations organization designed to accomplish the Department of Energy (DOE) production mission efficiently and cost-effectively while ensuring safe, secure, and environmentally sound operations. The company is a wholly-owned subsidiary of Westinghouse Electric Corporation with total authority, responsibility, and accountability for all phases of the Savannah River Site Management and Operating Contract.

A single, onsite executive, the WSRC President, has the responsibility and accountability for managing WSRC. All personnel within the WSRC, and those within Bechtel's subsidiary, Bechtel Savannah River, Inc. (BSR), who are required for regular, day-to-day effort in connection with the Savannah River Site Management and Operating Contract are located at or near the site. This organization provides enhanced communications within the SRS.

The major functions of the SRS are assigned to divisions, each under the direction of a Vice President. The Vice Presidents report directly to the WSRC President. Chapter 13 of the *Savannah River Site K Production Reactor Safety Analysis Report (SAR)*, WSRC-RP-89-383, contains additional details on the conduct of operation.

The control documents for the SRS K-Reactor are prescribed to ensure that basic and important decisions are made only after appropriate review and that decisions that could significantly affect safety are reviewed by a competent technical organization other than the one charged with direct responsibility for the operation.

5.2 MANAGEMENT POLICIES

5.2.1 General management policies and guidance shall include the following specific policies:

1. Decisions that have significant safety or production implications shall be reviewed by at least one organization not directly responsible for operation, before consideration and final approval by management. Safety-related operations shall be carried out by written procedures approved in advance.
2. The overall activity shall be considered as a system involving the interaction of process, people, facilities, and procedures.
3. Activities that have reactor safety significance shall have the review and input of the operating organization and the advisory organizations that are independent of reactor operation.

5.0 ADMINISTRATIVE CONTROLS

5.2.1 Continued

4. Hierarchies of administrative control shall correspond reasonably with organizational structures, i.e., detailed procedures originate at the operational level, and concepts and principles originate at managerial levels. Authority shall be delegated throughout the organization to match functional needs, but higher management shall retain full responsibility.
5. After safe boundaries for operation are carefully defined, they shall be approved by management with full recognition and acceptance of residual risks. This process shall be well documented, and the acceptable safe boundaries communicated to affected parties.
6. The principles of defense-in-depth and redundancy shall be applied as necessary to the operation including people, process, and equipment. The intent is not only to prevent accidents but to make provisions to limit consequences should accidents occur.

- 5.2.2 Management policies shall be implemented through WSRC management approved procedures.

5.3 REACTOR OPERATIONS DEPARTMENT RESPONSIBILITIES

5.3.1 Reactor Operations Department shall be responsible for:

1. Operation of the SRS K-Reactor in accordance with approved Technical Specifications.
2. Facilitation of physical changes in plant configuration and coordination of the activities of all work groups in the K-Reactor area.
3. Ensuring that each On-Duty shift shall be composed of at least the minimum shift-crew composition shown in Table 5.3.1-1.

5.0 ADMINISTRATIVE CONTROLS

TABLE 5.3.1-1
Minimum Shift Crew Composition

Reactor Condition	Minimum Certified CR Operators in CR (1)	Minimum Qualified Building Operators in Limited Area (1)	Minimum Certified Reactor Supervisors in CR (2)
1. REACTOR STARTUP (3) up to 75 MW	5	4	1
2. Power above 75 MW (3)	3	4	1
3. C&D OPERATION (3) (Fuel & Moderator in Reactor)	3	2	1
4. REACTOR SHUTDOWN (3) (Fuel & Moderator in Reactor)	2	2	1
5. REACTOR SHUTDOWN (4) (Fuel or moderator out of Reactor)	1	1	1

- (1) Except for the Control Room Supervisor, the Shift Crew Composition may be one less than the minimum requirements identified in Table 5.3.1-1 (including Note (3) personnel) for a period of time not to exceed 2 hours in order to accommodate unexpected absence (i.e., illness or injury) of on duty shift crew members, provided immediate action is taken to restore the Shift Crew Composition within the minimum requirements of Table 5.3.1-1 (including Note (3) personnel). This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.
- (2) The Certified Reactor Supervisor position may be filled by any certified supervisor including, but not limited to, the Shift Manager or the Control Room Supervisor.
- (3) A Shift Technical Advisor (STA) is required to be within the Limited Area and be able to return to the Control Room within ten minutes to provide technical guidance whenever heat generating assemblies are present in the Reactor or reactivity changes are being made.
- (4) If both fuel and moderator are out of the K-Reactor and fuel and moderator are present in P-Reactor or L-Reactor, consistent with the Remote Monitoring and Control System applicability, the minimum complement of personnel in the Control Room shall be a certified Reactor Supervisor or a certified Control Room Operator.

5.0 ADMINISTRATIVE CONTROLS

5.3.1 Continued

4. Ensuring all reactor operations are performed under the direct supervision of a certified reactor supervisor or a certified Control Room Operator who has direct responsibility for reactor safety. Reactor control is carried out by certified Control Room Operators according to written procedures.
5. Ensuring that prior to certification, a trainee operates the reactor controls only under the direct supervision of a certified Control Room Operator.
6. Ensuring all CHARGE AND DISCHARGE OPERATION(S) (C&D) are performed under the direct supervision of a certified components handling supervisor. The supervisor will be present in the crane control room during all C&D OPERATIONS where any reactor components are being handled. C&D are carried out by components handling qualified Operators according to detailed written procedures.
7. Ensuring that prior to qualification, a trainee operates the C&D machine controls only under the direct supervision of a components handling qualified Operator. The minimum complement of components handling qualified Operators in the crane control room during CHARGE or DISCHARGE is three.
8. Ensuring that administrative procedures shall limit the working hours of specified plant staff as specified in 5.3.2.

5.3.2 Reactor Operations Department certified supervisors and operators are subject to the following limitations when being assigned work outside their regular schedule. These limits do not include time for shift turnover or safety/security meetings.

1. 16 hours in 24 hours.
2. 72 hours in 7 days.
3. 19 consecutive days followed by 2 days off.
4. A single day's break interrupts the consecutive day sequence.
5. Deviations from these requirements may only be authorized in writing by the Plant Manager.

5.0 ADMINISTRATIVE CONTROLS

5.4 PROCEDURAL SYSTEM OF CONTROL DOCUMENTS

5.4.1 System Approach

1. A formalized system of procedures shall be employed to ensure K-Reactor is operated and maintained as prescribed by Technical Specifications, Safety Analysis Report (SAR), and management policies.
2. The SAR and Technical Specifications shall provide the criteria and basis for safe reactor operation. These documents in turn are implemented by procedures and other lower-tier documents. The procedures and other lower-tier documents contain limits on variables and system operation that are at least as restrictive as those in the SAR and Technical Specifications. Proposed procedures, Test Authorizations, and other document changes are reviewed to ensure control of safety margins.

5.4.2 Relationship of Controls

1. The primary SRS K-Reactor control documents (SAR and Technical Specifications) shall:
 - a. Prescribe nuclear safety control in the reactor area.
 - b. Be prepared, and then reviewed by a competent technical organization other than the one charged with direct responsibility for the operation.
 - c. Be approved as specified in the company level procedures. The preparation is usually on a lower level, but the review of a control document usually spans many levels within a division of WSRC. Appropriate divisions review and concur on a control document before it is approved by upper management of WSRC. Acceptable resolution of review comments is indicated by the approval signature of the appropriate manager.
2. Other control documents are approved as specified in Section 5.5 and/or implementing procedures.

5.5 DESCRIPTION OF CONTROLS

5.5.1 Contractual Agreement

The primary control document is the contract between WSRC and the Department of Energy, DE-AC-09-88SR18035. The contract explicitly describes obligations with regard to nuclear reactor safety on the part of both the contractor (WSRC) and the contracting officer (DOE).

INFORMATION ONLY

5.0 ADMINISTRATIVE CONTROLS

5.5.2 Safety Analysis Report

The Safety Analysis Report (SAR) shall:

1. Document the principal evaluations made to determine K-Reactor can be operated without undue risk to the public.
2. Outline the design objectives.
3. Describe the organizational responsibilities, operation, and equipment.
4. Identify potential hazards and parameters affecting reactor safety.
5. Describe the steps taken to avoid exceeding safety limits.
6. Indicate the predicted response of the facility or process to postulated failures.
7. Determine with reasonable assurance K-Reactor has the capacity for preventing accidents or mitigating their effects sufficiently to preclude undue risks to health and safety.
8. Identify boundaries between acceptable and unacceptable conditions. Limiting values for these boundaries are specified in Technical Specifications and procedures.
9. Be revised at least annually to include technical information that involves safety considerations, major changes in the reactor complex, or significant changes in charge design and operation.

5.5.3 Technical Specifications

The Technical Specifications shall:

1. Define the envelope of authorized operations (or safe operating envelope) of the K-Reactor. The Technical Specifications formally document the requirements in the following categories:
 - a. Use and Application
 - b. Safety Limits
 - c. Limiting Conditions for Operation and Surveillance Requirements Applicability
 - d. Design Features
 - e. Administrative Controls

5.0 ADMINISTRATIVE CONTROLS

5.5.3 Continued

2. Require the following actions be taken in the event a Technical Specification Safety Limit is violated:
 - a. REACTOR SHUTDOWN and DOE notified.
 - b. Before subsequent REACTOR STARTUP, submit to DOE for approval a complete root cause analysis of the circumstances leading to and resulting from the incident together with recommendations of measures to be taken to prevent recurrence of the incident.
 - c. DOE approval obtained prior to subsequent REACTOR STARTUP.
3. Be complied with, except reasonable action may be taken that departs from a Technical Specification in an emergency when this action is immediately needed to protect the public health and safety and no action consistent with the Technical Specifications that can provide adequate or equivalent protection is immediately apparent.
4. Be procedurally controlled to require changes:
 - a. Be made only after preparation, review, and approval of the proposed rewording of the Specification, a description of the revision, and justification for the revision.
 - b. Be independently reviewed by the Reactor Operations Safety Committee (ROSC) and the Reactor Independent Safety Review Committee (RISRC) prior to DOE submittal.
5. Have violations reviewed by the RISRC.

5.5.4 Unreviewed Safety Question Determination

1. A proposed activity (Change, Test, or Experiment) or discovery of new information associated with the design basis shall be considered to involve an Unreviewed Safety Question (USQ) if the proposed activity results in any of the following:
 - a. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety which was previously evaluated in the SAR may be increased, or
 - b. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created, or

5.0 ADMINISTRATIVE CONTROLS

5.5.4 Continued

- c. The margin of safety as defined in the basis for any Technical Specification is reduced.

2. Unreviewed Safety Question Determinations (USQDs) shall:

- a. Be controlled by procedure(s) that describe how USQDs are performed, approved, and documented.
- b. Be performed on proposed activities (changes, tests, or experiments) or discovery of new information associated with the design basis to determine if a USQ is involved. Proposed activities are reviewed (USQ Screening) to determine if a USQ Evaluation is necessary.
- c. When a USQ Evaluation is completed, it shall be reviewed by a Qualified Reviewer and the Reactor Operations Safety Committee (ROSC).
- d. When a positive USQ is involved in the proposed activity, it shall be reviewed by the ROSC, the Reactor Independent Safety Review Committee (RISRC), and be approved by DOE prior to implementation of the proposed activity.

5.5.5 Reactor Startup

- 1. DOE approval of Reactor Startup Authorization (RSAs) is required prior to REACTOR STARTUP when modifications have been installed which involve a USQ.
- 2. DOE Assistant Secretary for Defense Programs approval is required for REACTOR STARTUP following a shutdown as a result of a Seismic Event.
- 3. All other REACTOR STARTUPS require approval of the Plant Manager or designee and DOE concurrence.

5.5.6 Test Authorizations

Test Authorizations shall:

- 1. Have scopes confined to:
 - a. Test Prototype instruments or equipment.
 - b. Conduct tests using reactor system equipment.
 - c. Irradiate special materials.
 - d. Conduct preoperational tests of safety related equipment following major modifications.

5.0 ADMINISTRATIVE CONTROLS

5.5.6.1 Continued

- e. Tests or experiments on or involving safety related equipment or systems.
- 2. Be evaluated for USQs pursuant to Specification 5.5.4.
- 3. Not be used to authorize operation outside Technical Specifications.

5.5.7 Procedures

1. Procedural Activities

- a. Procedures shall be established, implemented, and maintained to address the activities specified in Table 5.5.7-1.
- b. Procedures required per Specification 5.5.7.1.a shall be reviewed and approved in conformance with the following:
 - (1) Procedures and changes thereto, shall be prepared, reviewed, and approved. Each such procedure or procedure change shall be reviewed by a qualified individual other than the individual who prepared the procedure or procedure change, but who may be from the same organization as the individual who prepared the procedure or procedure change. Procedures shall be approved by the appropriate responsible Manager as designated in writing by the Plant Manager or Manager, Reactor Operations as appropriate.
 - (2) Individuals responsible for reviews performed in accordance with Specification 5.5.7.1.b (1) shall be designated in approved written procedures. Each such review shall be performed by qualified personnel of the appropriate discipline and shall include a determination of whether or not additional cross-disciplinary review is necessary. Each review shall also include determination of whether an Unreviewed Safety Question is involved pursuant to Specification 5.5.4.
 - (3) Reactor Engineering shall perform a cross disciplinary review and concurrence on safety related procedures, and other procedures as specified within implementing procedures prior to approval.
 - (4) Reactor Quality shall preform a cross disciplinary review and concurrence on procedures prescribed by the Quality Assurance Program and other procedures as specified within implementing procedures prior to approval.

5.0 ADMINISTRATIVE CONTROLS

TABLE 5.5.7-1
Procedural Activities

A.	Administrative Procedures to govern:
	1. Authority and responsibility for Plant safe operation and shutdown 2. Equipment Control (e.g., locking and tagging, etc.) 3. Procedure adherence 4. Procedure review and approval 5. Conduct of Operations 6. Control of maintenance and modification work
B.	Operating Procedures to govern:
	1. Surveillance requirements 2. Major plant evolutions 3. Startup, operation, and shutdown of Plant systems and equipment 4. Refueling operations 5. Safety computer operations
C.	Maintenance Procedures to govern:
	1. Control of routine maintenance, inspection, calibration and test activities 2. Preventive and Corrective Maintenance 3. Surveillance Requirements
D.	Procedures governing Alarm Response, Abnormal Operating, and Emergency Conditions Procedures
E.	Implementation of Measuring and Test Equipment (M&TE) Program
F.	Implementation of the Reactor Fire Control Plan and Fire Control Preplan
G.	Implementation of the Site and Reactor Emergency Plans
H.	Implementation of the Security Plan
I.	Implementation of Radioactivity Control Program to limit materials released to the environment and personnel exposure

5.0 ADMINISTRATIVE CONTROLS

5.5.7 Continued**2. Temporary Approval of Procedure Changes**

Temporary approval of procedure changes shall:

- a. Ensure the intent of the original procedure is not altered.
- b. Be approved by two members of the plant management staff, at least one of whom holds certification as a reactor supervisor.
- c. Be approved in accordance with Specification 5.5.7.1.b within 14 days of implementation.

3. Online Computers

Online computers used to control or assist in the control of reactor operations shall:

- a. Be subject to the same basic system of controls and approval as related items of reactor control equipment.
- b. Have necessary program changes reviewed and approved to the same level as operating procedure revisions.
- c. Have no significant change in safety or control equipment or programming without appropriate:
 - (1) Documentation.
 - (2) Approval.
 - (3) Offline testing.

4. Startup and Test Procedures

Startup and Test Procedures shall be controlled according to applicable implementing procedures.

5.5.8 Process Technical Manuals

The Process Technical Manuals and their revisions shall:

1. Be used for major process development.
2. Show the technical bases of the development including:
 - a. Experimental results.

5.0 ADMINISTRATIVE CONTROLS

5.5.8.2 Continued

- b. Analytical methods.
- c. A detailed description of the process.
- 3. Contain the basic technical information for:
 - a. Design of the facilities.
 - b. Preparation of the Safety Analysis Report.

5.5.9 Qualification and Training Requirements

The Qualification and Training Requirements shall:

- 1. Identify the selected reactor personnel to which the requirements apply.
- 2. Identify the applicable qualification requirements to the specific reactor personnel.
- 3. Identify the applicable training requirements to the specific reactor personnel.
- 4. Be procedurally controlled to require major revisions in policy or content receive:
 - a. Management review and concurrence.
 - b. DOE approval.

5.5.10 Emergency Plan - Reactor Plan

The Reactor Emergency Plan shall define specific policies and plans to prevent or minimize injuries and property damage caused by accidents, natural disasters, or deliberate damage in the reactor area.

5.5.11 Emergency Plan - Site Plans

- 1. The Site Emergency Plan shall define program direction for implementing emergency preparedness requirements.
- 2. The Site Emergency Plan shall address onsite and offsite emergency preparedness requirements.

5.0 ADMINISTRATIVE CONTROLS

5.5.12 Security Plan

The Security Plan(s) shall:

1. Provide primary safeguards and security planning.
2. Identify reporting requirements.
3. Identify the acceptable levels of risk for safeguards and security interests.
4. Project the upgrades and operating resources required to address the vulnerabilities and risks.
5. Describe facility protection systems and programs.

5.5.13 Fire Control Preplan

The Fire Control Preplan shall:

1. Specify the organization to manually suppress fires.
2. Specify the action to manually suppress fires.
3. Specify the development of plant area fire protection preplans that adequately cope with any emergency that might occur during firefighting.

5.5.14 Reactor Fire Control Plan

The Reactor Fire Control Plan shall:

1. Define the reactor firefighting:
 - a. Groups.
 - b. Responsibilities.
 - c. Training material.
2. Ensure 24 hour firefighting coverage.
3. Provide special information by building location concerning:
 - a. Catalog of fire and other hazards.
 - b. Fire protection equipment provided.

5.0 ADMINISTRATIVE CONTROLS

5.5.14 Continued

4. Provide guidance to govern the use of water in fire emergencies that may involve fissionable materials.
5. Ensure detailed firefighting techniques are provided in training sessions.

5.5.15 Core Operation Report

The Core Operation Report shall:

1. Be produced prior to each reload cycle.
2. Be produced or revised prior to restart of any remaining portion of a reload cycle which results from core component changes (subcycle).
3. Establish and document the core operating limits in the Core Operation Report - Thermal-Hydraulic Limits and Physics Volumes:
 - a. The Thermal-Hydraulic Limits Volume shall contain core operating limits and parameters that are within the applicable limits of the safety analysis, and
 - b. Be produced using DOE approved analytical methods to determine the core operating limits.
 - c. The Physics Volume shall document the physics core design and provide information for core reactivity management operations and assessment.
 - d. The Core Operation Report shall be submitted to DOE for approval upon initial issuance or initial issuance following any revision to the DOE approved analytical methods. Subsequent cycle reports do not require DOE approval, but shall be submitted each reload cycle to DOE for information, and
 - e. Be supported by a reload evaluation.

5.5.16 Core Shutdown Report

1. The Core Shutdown Report shall contain:
 - a. Shutdown cooling time limits established by analytical methods and parameters approved by DOE.

5.0 ADMINISTRATIVE CONTROLS

5.5.16 Continued

- b. Shutdown cooling limits determined so that applicable limits of the safety analysis are met.
 - c. Shutdown cooling limits documentation prior to the discharge of any HEAT GENERATING ASSEMBLIES.
2. If the Core Shutdown Report is not complete and approved at the time of the cycle (or sub-cycle) shutdown, forced circulation shall be maintained and discharge operation shall not commence, until such time as the report has been issued and approved by DOE.

5.6 EVENTS, CONDITIONS AND CONCERN INVESTIGATIONS, AND REPORTS

5.6.1. Events, conditions, and concerns which may involve safety, health, quality, safeguards, security, or environmental implications shall be controlled by administrative procedures that ensure:

- 1. Identification.
- 2. Reporting.
- 3. Evaluation.
- 4. Corrective action determination and implementation.
- 5. Disposition.
- 6. Prioritization.
- 7. Close out.
- 8. Line and independent reviews.

5.6.2 Operation outside Technical Specification limitations shall:

- 1. If proposed, have DOE approval and issuance of a Technical Specification Revision prior to implementation.
- 2. If a safety limit is violated, when discovered, require the actions of Specification 5.5.3.2.
- 3. Be determined based on evidence that implies the limits or requirements were not met.

5.0 ADMINISTRATIVE CONTROLS

5.7 REVIEW, AUDIT, AND SURVEILLANCE

A system of review, audit and surveillance is maintained. Operating, Maintenance and Component Handling Procedures are audited, and the facilities are surveyed, for adherence to and adequacy of procedures, and for indications of abnormal behavior in the process, equipment, and personnel performance. A system is maintained to provide internal review of safety considerations by competent technical personnel other than those having direct responsibility for the subject being reviewed.

An integral part of the internal review system includes:

- (1) A semiannual review of reactor operations by WSRC staff, and
- (2) An independent annual review of the adequacy of the training program for certified Reactor Supervisors, certified Component Handlers, and certified Control Room Operators.

At least annually, an independent overview of the safety of operation of the Savannah River Site production reactors is performed. Reactor Safety Evaluation Section provides an ongoing process of independent review, audit, and surveillance of activities related to reactor safety. The internal review system is evaluated every three years for adequacy of performance by WSRC Staff Management. Auditing, inspection, and review records are available for examination by DOE on request.

5.7.1 Reactor Operations Safety Committee (ROSC)

1. Function

The ROSC shall function to advise the Plant Manager (PM) on all matters affecting the operation of the reactor and associated activities that affect safety.

2. Composition

The ROSC shall consist of individuals having, as a minimum, diversified background in the areas of reactor operations, maintenance, engineering, and radiological protection. Committee permanent members and alternates must possess sufficient education, experience, expertise, and safety analysis and technical training to undertake the reviews and evaluations which the committee is intended to perform. Six years of related industrial, military, and/or SRS experience with four being in nuclear technology is the minimum experience requirement. The ROSC shall be organized as a Committee with the following membership.

- a. The Plant Manager shall in writing appoint a Chairman.

5.0 ADMINISTRATIVE CONTROLS

5.7.1.2 Continued

b. Membership appointments to the Committee shall be approved by the Chairman, and shall represent:

1. Reactor Operations.
2. Health Protection.
3. Reactor Engineering.
4. Reactor Maintenance.
5. Reactor Support Services.
6. Reactor Quality Assurance and Assessment.
7. Reactor Safety Evaluation Section.

3. Alternates

The Plant Manager may designate, in writing, an alternate Chairman who may serve in the absence of the Chairman. Other alternate members shall be approved by the Chairman. No more than two alternates may participate as voting ROSC members at any one time.

4. Meeting Frequency

The ROSC shall meet once a month, or more frequently as may be required by the Chairman.

5. Quorum

The quorum for meetings of the ROSC shall be as follows:

- a. The Chairman, or his designated alternate; and,
- b. At least four other members (2 alternates maximum). These other members shall be concurred with by the Chairman to bring to the Committee a sufficient spectrum of technical competence to address meeting subject material.
- c. The ROSC may delegate reviews to subcommittees at the direction of the ROSC Chairman. Subcommittees may consist of one or more individuals. Subcommittee findings shall be formally submitted to the ROSC. Subcommittee members are not required to be committee members.

5.0 ADMINISTRATIVE CONTROLS

5.7.1 Continued

6. Responsibilities

The ROSC shall be responsible for:

- a. Review of proposed revisions to:
 - (1) Procedures which establish administrative controls to implement the QA program within Reactor Operations Department and the Technical Specifications surveillance program.
 - (2) Safety-related emergency and abnormal operating procedures.
- b. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.
- c. Review of all Reportable Events, such as violations of the Technical Specifications including investigations and recommendations for preventing or reducing recurrences of these events.
- d. Review of all proposed changes to the Technical Specifications.
- e. Review of plant operations to detect potential hazards to nuclear safety.
- f. Review of Reactor Emergency Plans and implementing procedures, and submittal to the PM of any recommended changes.
- g. Review of any accidental, unplanned, or uncontrolled radioactive releases.
- h. Review of the Reactor Fire Protection Program and implementing procedures.
- i. Forwarding recommendations, or requests for assistance, to the RISRC on specific items identified by the ROSC Chairman.
- j. Performance of special reviews, investigations, or analyses and reports as requested by the Plant Manager.
- k. Review all proposed tests and experiments that affect nuclear safety.

5.0 ADMINISTRATIVE CONTROLS

5.7.2 Reactor Independent Safety Review Committee (RISRC)

1. Function

The RISRC shall have the expertise in the area of and shall function to provide independent review and assessment in the areas of:

- a. Reactor and supporting systems operations.
- b. Reactor engineering.
- c. Chemistry and radiochemistry.
- d. Metallurgy.
- e. Instrumentation and control.
- f. Radiological safety.
- g. Mechanical and electrical engineering.
- h. Quality Assurance.
- i. Administrative procedures.
- j. Organization training programs.
- k. Appropriate training programs.
- l. Initial and subsequent qualification and certification requirements and procedures.

The RISRC shall report to and advise the Vice President and General Manager, Reactor Restart Division (VP&GMRRD) on those areas of responsibility specified in Specification 5.7.2.7. All recommendations of the Committee shall be made at formal meetings.

2. Composition

The committee shall be composed of at least five individuals, including the chairman. The chairman shall be appointed in writing, by the VP&GMRRD of WSRC. Members, including designated alternates, shall be recommended by the chairman and appointed in writing by the VP&GMRRD of WSRC. Members, including designated alternates, shall meet the qualification requirements specified by American National Standard Institute/American Nuclear Society Standard 3.1-1987 Section 4.7.

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5.0 ADMINISTRATIVE CONTROLS

5.7.2.2 Continued

The RISRC may delegate reviews, appraisals and audits to subcommittees, provided that such subcommittees formally report their findings to the RISRC.

3. Alternates

Alternate members shall serve when principal members are unable to attend a Committee function. However, no more than a minority of voting members in RISRC activities at any one time shall be alternates.

4. Consultants

When circumstances warrant, the services of consultants to provide special expertise may be utilized as is determined appropriate by the Chairman.

5. Meeting Frequency

Formal meetings of the RISRC shall be convened as necessary, but not less frequently than quarterly. During preparation for restart, and for a period of one year thereafter, the Committee shall convene at least once per month.

6. Quorum

The quorum necessary for the performance of the committee functions shall consist of the chairman, or his/her designated alternate, and at least the majority of regular committee members. No more than a minority of the quorum shall have line responsibility for operation of the reactors.

7. Review Responsibilities

The RISRC shall be responsible for ensuring that reviews are conducted of:

- a. Safety evaluations performed for proposed activities pursuant to Specification 5.5.4 to verify that such completed changes or actions did not involve an Unreviewed Safety Question (USQ), including Unreviewed Safety Question Determinations.
- b. Proposed changes to procedures, equipment, or systems which involve a USQ, prior to submittal to DOE for approval.
- c. Proposed tests or experiments which involve a USQ prior to submittal to DOE for approval.
- d. Proposed changes to Technical Specifications prior to submittal to DOE for approval.

5.0 ADMINISTRATIVE CONTROLS

5.7.2.7 Continued

- e. Violations of orders, regulations, codes, Technical Specifications, or of internal procedures or instructions having nuclear safety significance.
- f. Significant operating abnormalities, or deviations from normal and expected performance of reactor plant equipment and systems that affect nuclear safety.
- g. All Reportable Events, conditions, or concerns, including investigations and recommendations for preventing or reducing recurrences of these events.
- h. Recommendations of independent/oversight review groups and dispositions thereof.
- i. All recognized indications of an unanticipated deficiency in some aspect of the design or the operation of structures, systems, or components that could affect nuclear safety.
- j. Items referred to the Committee by the ROSC or the VP&GMRRD.
- k. Organizational responses to identified concerns, adverse performance trends, and problems which could affect nuclear safety.
- l. Selected reports, meeting minutes, and other records of the activities of the ROSC, to ensure that the quality and effectiveness of evaluations are maintained, appropriate topics are reviewed, and reviews are sufficient in scope and depth to identify and address issues involved. All positions and concerns are considered, reviewed, and dispositioned; appropriate conclusions are reached based on an adequate technical analysis; action items are tracked through to completion; and that actions are completed in a timely manner. RISRC shall provide recommendations as deemed appropriate.

8. Audits

Independent audits or appraisals of reactor operation activities shall be performed by appropriately qualified personnel at least once within a 12 month period or as otherwise specified. The RISRC shall review the results of audits. The audits or appraisals shall encompass as a minimum:

- a. The conformance of reactor operations to the Technical Specifications.

5.0 ADMINISTRATIVE CONTROLS

5.7.2.8 Continued

- b. The performance, training, and qualifications of reactor operations personnel. Appropriate training programs, initial and subsequent qualification and certification requirements and procedures.
- c. The results of actions taken to correct deficiencies occurring in reactor equipment, structures, systems, or methods of operation that affect nuclear safety. The program for identification, evaluation, reporting, tracking, trending, dispositioning, implementing proposed corrective actions, and closing of events, conditions and concerns (semi-annually for two years, and annually thereafter).
- d. The performance of activities required by the Quality Assurance (QA) Program.
- e. Biennial review of the fire protection programmatic controls including the implementing procedures, and review of the fire protection equipment and program implementation once within a 12 month period. An outside independent fire protection consultant shall be used at least every third year.
- f. The Radiological Environmental Monitoring Program and resulting issues. The performance of activities for effluent and environmental monitoring.
- g. The Emergency Plan and implementing procedures.
- h. The Offsite Dose Calculation Manual, or equivalent document, and implementing procedures.
- i. The Process Control Program or equivalent and implementing procedures for processing and packaging of radioactive waste.
- j. The Security Plan and implementing procedures.
- k. The program for:
 - (1) reviewing proposed activities to determine if they involve a USQ, and
 - (2) dispositioning USQ items.
- l. Organization and staffing.

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5.0 ADMINISTRATIVE CONTROLS

5.7.2.8 Continued

- m. Any other area or site activity considered appropriate by the VP&GMRRD of WSRC.

9. Records

Records of RISRC activities shall be prepared, approved, distributed, and stored in accordance with the following:

- a. Minutes of each RISRC meeting shall be prepared, approved, and forwarded to the Vice President and General Manager, Reactor Restart Division (VP&GMRRD), within 14 days following the meeting. Minutes shall include notations of any dissenting professional opinions expressed within the Committee, if such occur.
- b. Reports of reviews encompassed by Specification 5.7.2.7 shall be prepared, approved, and forwarded to the VP&GMRRD, within 14 days of the completion of the review.
- c. Audit or appraisal reports encompassed by Specification 5.7.2.8 shall be forwarded to the VP&GMRRD, and to the management positions responsible for the areas within 30 days after the completion of the audit or appraisal.
- d. RISRC records shall be retained in accordance with a Records Retention Schedule.

5.7.3 Site Safety Review Committee (SSRC)

The SSRC advises the President of WSRC, on oversight of sitewide implementation of Environmental, Safety, Health, Safeguards, Security, and Quality Assurance programs. To fulfill its function, the SSRC will review the related operations of the RISRC and maintain an overview of the related operations.

5.7.4 ESH&QA/Reactor Safety Evaluation Section (RSES)

- 1. The ESH&QA Division provides independent oversight of the safety of operations of the Savannah River Site production reactors. Within ESH&QA, the Reactor Safety Evaluation Section (RSES) performs this function through an ongoing program of selective independent review, surveillance and assessment of activities related to reactor safety, including, but not limited to:
 - a. Programs and activities related to reactor design, construction, and operation.
 - b. Reactor safety-related events and lessons learned.

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5.0 ADMINISTRATIVE CONTROLS

- c. Commercial nuclear industry events and regulatory correspondence from the Nuclear Regulatory Commission to ensure that commercial industry experience is considered to enhance SRS reactor safety.
 - d. Indicators of reactor safety performance.
 - e. Post-trip independent reviews and providing concurrence with the resolution of the trip, prior to REACTOR STARTUP.
 - f. The functioning of nuclear safety committees, such as the RISRC and the ROSC.
 - g. Nuclear safety documents of the types defined in Section 5.5 of these Specifications, including review and approval prior to submittal to DOE of Safety Analysis Reports, Technical Specifications, Reactor Startup Authorizations, Core Operation Reports, and other documents as requested by the Vice President and General Manager, RRD.
2. The RSES is responsible for performing portions of the Independent Safety Review function and conducting annual safety reviews/appraisals per DOE Orders 5480.6, "Safety of DOE-Owned Reactors" and 5482.1B, "Environment, Safety, and Health Appraisal Program." RSES is also responsible for independently appraising WSRC management of reactor safety issues.

5.8 TRAINING

Initial training, continuing training and retraining of certified Reactor Supervisors and certified Control Room Operators are carried out by formal classroom instruction, simulator training, computer assisted instruction, and on-the-job experience. Component Handling Supervisors receive similar training with the exception of simulator training. Initial certification of Reactor Supervisors, Component Handling Supervisors and Control Room Operators is based on a demonstrated acceptable level of competence and performance. Initial certification depends on satisfactory completion of comprehensive written, oral, and operating examinations, satisfactory physical condition, general health, and higher supervision's judgment of general qualifications.

The training program addresses the positions identified for accreditation in DOE Order 5480.18. Performance-based training, as defined in the Department of Energy Training Accreditation Program (TAP) documents, is used for the design and implementation of all reactor training. Continuing training and reexamination on procedures for handling abnormal reactor conditions is done annually, and biennially for other procedures important to the safe operation of the reactors.

5.0 ADMINISTRATIVE CONTROLS

5.8 Continued

Recertification is biennial (biennial is two years or 24 months, with the maximum interval not to exceed 24 months) and depends in part on the satisfactory completion of written job performance measure evaluations and simulator examinations on both normal and abnormal operating conditions. The bases for both initial certification and recertification are documented. This documentation includes a copy of the most recent test results and test grades. Documentation on testing is available to DOE on request. An independent review of the adequacy of the training program is made annually.

A summary of the content of the training program shall be distributed to DOE for approval prior to implementation.

5.9 PLANT OPERATING RECORDS

5.9.1 Records retention practices shall be in accordance with the Savannah River Site Quality Assurance Plan and Records Management directive.

5.9.2 Specifically, the following documents shall be retained as records for the period specified by the Organization's Record Retention Schedule:

1. Records and logs of plant operation covering time interval at each power level.
2. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety.
3. All reportable events.
4. Records of surveillance activities, inspections, and calibrations required by Technical Specifications.
5. Records of changes made to procedures.
6. Records and drawing changes reflecting plant design modifications made to systems and equipment described in the Safety Analysis Report.
7. Records of radiation exposure for all individuals entering radiologically controlled areas.
8. Records of gaseous and liquid radioactive material released to the environs.
9. Records of transient or operational cycles of those plant components sensitive to fatigue.
10. Records of reactor tests and experiments.

5.0 ADMINISTRATIVE CONTROLS

5.9.2 Continued

11. Records of training and qualification for current members of the Reactor Operations staff.
12. Records of Unreviewed Safety Question screening and reviews performed for changes made to procedures or equipment or reviews of tests and experiments.
13. Records of meetings of the Reactor Independent Safety Review Committee and Reactor Operations Safety Committee.
14. Records of moderator water sampling and water quality.

5.10 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with DOE orders and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

5.10.1 Radiological Effluent Control Program

A Radiological Effluent Control Program, Environmental Monitoring Program, and an Offsite Dose Calculation Manual shall be documented and controlled consistent with DOE Orders and the Environmental Impact Statement. Annual Reports shall be submitted to DOE.

5.10.2 High and Very High Radiation Areas

Procedures for High and Very High Radiation Areas shall be prepared consistent with the following:

1. A High Radiation Area in which the intensity of radiation is equal to or greater than 100 mrem/hr but less than 5000 mrem/hr at 30 centimeters from the radiation source, shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Radiation Work Permit.
2. A Very High Radiation Area in which the intensity of radiation is equal to or greater than 5000 mrem/hr at 30 centimeters from the radiation source shall be subject to the provisions of Specifications 5.10.2.1 above, and in addition locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under administrative control, except during periods when access to the area is required, with positive controls over and radiation surveys made for the initial entry and periodically as necessary.

5.0 ADMINISTRATIVE CONTROLS

5.11 RESPIRATORY PROTECTION PROGRAM

Procedures for the use of respiratory equipment shall be prepared consistent with ANSI Z88.2.-1980.

5.12 WATER CHEMISTRY CONTROL PROGRAM

Procedures shall be established to control reactor water chemistry within limits that minimize the potential for radioactive material leakage.

5.13 QUALITY ASSURANCE

The Quality Assurance program shall:

- 5.13.1 Satisfy DOE requirements.
- 5.13.2 Be modeled after the National Consensus Standards referenced in DOE orders.
- 5.13.3 Be implemented through written procedures and instructions.
- 5.13.4 Specify those procedures which require review and concurrence by Quality Assurance.
- 5.13.5 Be applicable to construction, operation, maintenance, research, development and design.
- 5.13.6 Require sufficient records be maintained to preserve:
 - 1. Design basis tests.
 - 2. Experimental results including computational data used to derive safety limits, limiting safety system settings, limiting conditions for operation, and design features for operation.
- 5.13.7 Include line management responsibilities and audit requirements to determine compliance.
- 5.13.8 Require WSRC Staff Management and DOE approval prior to initial implementation or subsequent changes.

5.14 CHARGE and DISCHARGE OPERATIONS (C&D)

CHARGE AND DISCHARGE OPERATIONS (C&D) shall:

- 5.14.1 Be performed using appropriately approved operating procedures.
- 5.14.2 Decrease reactivity before increasing reactivity except as specified by approved operating procedures.

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Assembly Effluent Temperatures - TECHNICAL LIMITS B 2.1

B 2.0 SAFETY LIMITS (THERMAL-HYDRAULIC LIMITS)

B 2.1 Assembly Effluent Temperatures - TECHNICAL LIMITS

BASES

The primary protection against damage to fuel/target assemblies, or the reactor structure is provided by operation within limits on the effluent temperature of coolant from individual fuel/target assemblies. These limits are designed to prevent flow instability in an assembly. Flow instability can reduce coolant flow so that the coolant passages are filled with steam rather than liquid. Because of impaired heat transfer, fuel or target elements could melt and/or the pressure could increase in the reactor tank from the generation of steam. Impaired heat transfer provides a mechanism whereby a large fraction of an assembly or many assemblies could melt.

Flow instability (Reference 1) will not occur until: (1) the onset of significant voiding with the resultant dryout of fuel tube surfaces and increase in fuel temperature (OSV limit) or (2) the exit temperature exceeds saturation for the local pressure in the fluid stream exiting through the shell holes (TSAT limit). TECHNICAL LIMITS are established so that during normal operation at the limit, the effluent coolant temperature in the bottom end fitting shell holes does not exceed the saturation temperature. The TECHNICAL LIMIT is the saturation temperature reduced to allow for a margin of safety and is specified in terms of the temperature measured by monitor pin thermocouples. Therefore, flow instability in the coolant is avoided at the TECHNICAL LIMIT and protection against flow instability in any part of the assembly is provided.

REFERENCES

1. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 15.1.2.3.1, "Flow Instability and Assembly Effluent Temperature." (U)
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B 2.0 SAFETY LIMITS (THERMAL-HYDRAULIC LIMITS)

B 2.2 Normal Reactor Burnout Risk - TECHNICAL LIMITS

BASES

BOR limits are specified to avoid significant release of fission products caused by local film boiling burnout of fuel and target surfaces during normal operation or during specified transients. Film boiling burnout may occur when the critical heat flux is exceeded. Heat transfer coefficients beyond the point of critical heat flux rapidly decrease and could cause the surface temperature to rise above the melting point of the material.

A TECHNICAL LIMIT is established so that for operation at the limit, the calculated risk of fission product release to the process water caused by film boiling burnout in all assemblies of the charge will not be significantly greater than that already experienced from other mechanisms of releasing activity (e.g., fuel cladding defects, vibration, corrosion, etc.). The release corresponds to the total inventory of a core operated continuously at 0.03 MW. This risk corresponds to melting a volume of irradiated fuel that would contain about 10E5 Ci of fission products. This activity level has been encountered in operation and is such that continuous operation can be maintained without undue hazard or significant interference with normal operation.

REFERENCES

1. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 4.4.1.2.2, "Thermal - Hydraulic Limits." (U)
 2. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 15.1.2.3.2, "Burnout Risk (BOR)." (U)
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Reactor Effluent Temperatures - TECHNICAL LIMITS
B 2.3

B 2.0 SAFETY LIMITS (THERMAL-HYDRAULIC LIMITS)

B 2.3 Reactor Effluent Temperatures - TECHNICAL LIMITS

BASES

Limits on reactor effluent temperature provide protection against abnormal cavitation damage to the Process Water System and/or loss in pumping capacity. Cavitation can result when fluid velocity and/or temperature increases sufficiently to cause vaporization in low pressure locations. Additionally, sudden reductions in reactor tank pressure alone can cause cavitation. Boundary materials in the immediate vicinity are subjected to rapidly repeated stress reversals and may eventually fail through fatigue, the first sign of which is surface pitting.

In the K reactor, cavitation occurs in the elbows just downstream from the reactor outlet nozzles, at a lower temperature than in the pumps. Reactor effluent temperature, as a controlled variable, differs from assembly effluent temperature and BOR in that excessive reactor effluent temperature is not directly related to fuel/target damage or to sudden and significant reactor damage. Reactor operation at an excessive reactor effluent temperature can produce progressive mechanical damage to the process water circulation pumps and their suction lines, but will not result in direct damage to the fuel/target assemblies. Excessive reactor effluent temperature can contribute to fuel/target damage only insofar as cavitation can reduce process water flow and thus cause limits on assembly effluent temperature and BOR to be exceeded.

The TECHNICAL LIMITS on reactor effluent temperature are established so that, during normal operation at the limit, detectable cavitation will not occur in the Bingham pumps or their suction lines. This limit protects against abnormal cavitation damage to the process water circulating system. The TECHNICAL LIMITS are based on experimental determinations in the reactors and by the pump manufacturer (Refs. 1 and 2).

(continued)

INFC

ONLY

Reactor Effluent Temperatures - TECHNICAL LIMITS
B 2.3

BASES (continued)

REFERENCES

1. Hamm, L. L., "Revised Reactor Effluent Characteristics for PLK Reactors," DPST-84-625, October 26, 1984. (U)
 2. Bingham Pump Company Performance Tests, Pumps No. 1 through 7, AXC 22156-1/2, DPF 2092325.
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B 3.0 LIMITING CONDITIONS FOR OPERATION (LCO) - APPLICABILITY

BASES

LCOs 3.0.1 - 3.0.6	LCOs 3.0.1 through 3.0.6 establish the general requirements applicable to all Specifications unless otherwise stated.
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LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when (i.e., in which reactor states or other specified conditions) conformance to the LCO is required.
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LCO 3.0.2	<p>LCO 3.0.2 establishes that upon discovery of a failure to meet a Limiting Condition for Operation, the associated ACTIONS shall be met. The Completion Times of the Required Actions are applicable from the point in time 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:</p> <ol style="list-style-type: none"> 1. completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification, and 2. completion of the Required Actions is not required when an LCO is met within the specified Completion Time unless otherwise specified.
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There are two basic types of Required Actions. The first type of Required Action has an associated 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 for restoring parameters within specified limits. If these Required Actions are not completed within the specified Completion Time, a shutdown may be required to place the reactor in a reactor state or condition in which the Specification no longer applies. The second type of Required Action specifies the remedial measures that

(continued)

BASES

LCO 3.0.2
(continued)

permit continued operation of the reactor which is not further restricted by the Completion Time. In this case, conformance to the Required Actions provides an acceptable level of safety for continued operation.

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 Surveillance Requirements, preventive maintenance, corrective maintenance, or investigation of operational problems. It is not intended that intentional entry into shutdown Required Actions (i.e., Actions requiring a change in reactor state) be made for operational convenience. This is to limit routine voluntary removal of redundant equipment from service in lieu of other alternatives that would not result in redundant equipment being inoperable. Individual Specifications may include a specified Completion Time of a Surveillance Requirement when equipment is removed from service, or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this limit expires if the Surveillance has not been completed.

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

It is allowable to be in more than one ACTIONS Condition of a given LCO at the same time. When in multiple ACTIONS Conditions the Required Actions of each Condition entered must be completed within the specified Completion time.

LCO 3.0.3

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

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

(continued)

BASES

LCO 3.0.3
(continued)

2. The Condition is not specifically addressed by the associated ACTIONS.

This Specification delineates the time limits for placing the reactor in a safe reactor state or other specified condition when operation cannot be maintained within the limits for safe operation defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience which 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.

One hour is allowed to prepare for an orderly shutdown before initiating a change in plant operation. The time limits specified to reach lower reactor states 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 reactor assuming only the minimum required equipment is OPERABLE. This reduces the potential for a scram that could challenge safety systems under conditions for which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 shall be consistent with the discussion of Section 1.3, Completion Times.

A reactor shutdown required per LCO 3.0.3 may be terminated, and LCO 3.0.3 exited if any of the following occurs:

1. The LCO is now met.
2. A Condition exists for which the Required Actions have now been performed.
3. ACTIONS exist which do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition was initially entered and not from the time LCO 3.0.3 is exited.

(continued)

REACTOR ONLY

BASES

**LCO 3.0.3
(continued)**

The time limits of Specification 3.0.3 allow 90 minutes for the reactor to be in REACTOR SHUTDOWN when a shutdown is required during REACTOR OPERATION.

During REACTOR OPERATION LCO 3.0.3 provides actions for Conditions not covered in Specifications 3.1.1 through 3.9.9. The requirements of LCO 3.0.3 do not apply in REACTOR SHUTDOWN because the reactor 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 in the Applicability, unless in REACTOR OPERATION because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

The exceptions to LCO 3.0.3 are provided in instances where requiring a reactor shutdown in accordance with LCO 3.0.3 would not provide appropriate remedial measures for the associated condition. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

LCO 3.0.4 precludes entry into the reactor Conditions specified in the Applicability unless the associated ACTIONS permit continued operation of the reactor for an unlimited period of time. LCO 3.0.4 does not prevent changes in reactor Conditions which are necessary to comply with Required Actions.

Compliance with Required Actions that permit continued operation of the reactor for an unlimited period of time in the reactor state or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the reactor before or after

(continued)

BASES

LCO 3.0.4 (continued)

a reactor state change. Therefore, in this case, entry into a reactor state or other condition specified in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not, however, be interpreted as endorsing the failure to exercise good practice in restoring systems or components to OPERABLE status before reactor startup.

The provisions of LCO 3.0.4 shall not prevent changes in reactor states or other specified conditions in the Applicability which are required to comply with ACTIONS.

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

When changing reactor states or other specified conditions while in an ACTIONS Condition in compliance with LCO 3.0.4, or where an exception to LCO 3.0.4 is stated, the ACTIONS define the remedial measures that apply. Surveillance Requirements do not have to be performed on the associated inoperable equipment. Therefore a change in reactor state or other specified condition in this situation does not violate SR 3.0.1 or SR 3.0.4 for those Surveillance Requirements that do not have to be performed on the associated inoperable equipment. However, Surveillance Requirements must be met to demonstrate OPERABILITY prior to declaring the associated equipment OPERABLE and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service when it has been removed from service or declared inoperable to comply with ACTIONS. This Specification provides an exception to LCO 3.0.2 to allow performance of testing to:

1. Demonstrate the OPERABILITY of the equipment being returned to service, or
2. Demonstrate the OPERABILITY of other equipment.

It is necessary to allow such testing to be conducted to prove OPERABILITY and return the equipment to service. Pl inoperable equipment in service for this testing does not affect its Completion Time.

BASES (continued)

LCO 3.0.6

LCO 3.0.6 establishes which ACTIONS are applicable when support systems are inoperable, depending on whether or not they have an LCO specified in the Technical Specifications. The supported system is not required to be declared inoperable solely due to support system inoperability. Only the support system LCO's ACTIONS are required to be entered. The supported systems ACTIONS are only required to be entered where directed to do so by the support systems ACTIONS. This is a clarification of the definition of OPERABILITY but is necessary to establish the relationship between the support systems and the supported system in order to preclude cascading to multiple supported system ACTIONS and to eliminate the confusion associated with entering multiple LCOs' ACTIONS. Examples of support systems with LCOs specified in the Technical Specifications include plant cooling water, diesel generators and AC and DC distribution.

When a support system is inoperable and there is not an LCO for that support system specified in the Technical Specifications, the impact of the inoperability or degradation of the support system function on the OPERABILITY of the supported system shall be evaluated. It is intended that this evaluation be performed as soon as practical and also consider the time required to obtain the information required to support the evaluation. This is because the inoperability or degradation of the support system function may or may not affect the OPERABILITY of the supported system, depending upon the intended function of the supported system and the level of support the support system provides. Upon determination that the supported system is inoperable, the ACTIONS of its LCO shall apply.

INFORMATION ONLY

SR Applicability
B 3.0

B 3.0 SURVEILLANCE REQUIREMENTS (SR) - APPLICABILITY

BASES

SRs
3.0.1 - 3.0.4

SRs 3.0.1 through 3.0.4 establish the general requirements applicable to all Specifications unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that Surveillance Requirements must be met during the reactor states or other specified conditions in the Applicability for which the requirements of the LCO apply unless otherwise stated in an individual Surveillance Requirement. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components and that parameters are within specified limits. Failure to meet a Surveillance Requirement 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 Surveillance Requirements have been met. However, nothing in this Specification is to be construed as implying that systems or components are OPERABLE when either:

1. they are known to be inoperable although still meeting the Surveillance Requirements, or
2. the requirements of the Surveillance(s) are known not to be met between required Surveillance performances.

Surveillance Requirements do not have to be performed when the reactor is in a reactor state or other specified condition for which the requirements of the associated LCO do not apply unless otherwise specified. The Surveillance Requirements associated with a Test Exception are only applicable when the Test Exception is used as an allowable exception to the requirements of a Specification.

Surveillance Requirements, including Surveillance Requirements required by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillance Requirements have to be met 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 meeting applicable Surveillance Requirements in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current reactor state or other specified conditions in the Applicability due to the necessary reactor 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 reactor state or other specified condition where other necessary post maintenance tests can be completed.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances, Required Actions which require the performance of a Surveillance Requirement, and any Required Action with a Completion Time requiring the periodic performance of an action on a "once per..." interval. SR 3.0.2 permits an extension of the nominal surveillance frequency. This facilitates Surveillance scheduling and considers reactor operating conditions that may not be suitable for conducting the Surveillance; e.g., transient conditions or other ongoing Surveillance or maintenance activities.

The extension does not significantly degrade the reliability which results from performing the Surveillance at its specified frequency. This is based on the recognition that the most probable result of any particular Surveillance Requirement being performed is the verification of conformance with the Surveillance Requirements. The exceptions to SR 3.0.2 are those Surveillances where the extension of the nominal surveillance frequency does not apply. These exceptions are stated in the individual Specifications. An example of where SR 3.0.2 does not apply is a Surveillance with a frequency of "In accordance with 10 CFR 50 Appendix J, and approved exemptions." The requirements of regulations take precedence over the Technical Specifications. The Technical Specifications cannot in and of themselves extend a test interval specified in the regulations.

(continued)

INFORMATION ONLY

BASES (continued)

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring equipment inoperable when Surveillances have inadvertently not been completed within the specified Frequency. An allowed deferral of up to 24 hours applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2 and not at the time that the specified Frequency was not met.

This deferral provides an adequate time limit to complete Surveillances that have been missed. This allowance permits the completion of a Surveillance before compliance with Required Actions or other remedial measures would be required that may preclude completion of a Surveillance. The basis for this allowance includes consideration of reactor 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 Surveillance Requirements. This provision also provides a time limit for completion of Surveillance Requirements that become applicable as a consequence of reactor state changes imposed by Required Actions.

If a Surveillance is not completed within the allowed deferral time, the equipment is considered inoperable and the Completion Times of the Required Actions begin immediately upon expiration of the allowed deferral time. If a Surveillance is failed within the allowed deferral time, the equipment is considered inoperable and the Completion Times of the Required Actions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance Requirement in accordance with the allowed deferral time of this Specification or within the Completion Time of the ACTIONS restores compliance with the requirements of SR 3.0.1.

The intent of SR 3.0.3 is to allow performance of a missed surveillance for up to 24 hours depending on planning and time required to perform the surveillance, not to allow the deference of all missed surveillances for up to 24 hours.

(continued)

BASES

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable Surveillances must be met before entry into a reactor state or other specified condition in the Applicability. This Specification ensures that system and component OPERABILITY requirements or parameter limits are met before entry into a reactor state or specified condition in the Applicability for which these systems and components ensure safe operation of the reactor. This provision applies to changes in reactor states or other specified conditions in the Applicability associated with reactor shutdown as well as startup.

The provisions of SR 3.0.4 shall not prevent changes in reactor states or other specified conditions in the Applicability which are required to comply with ACTIONS.

Exceptions to SR 3.0.4 are stated in the individual Surveillance Requirements. Exceptions to this Specification are allowed in order to perform Surveillance testing when the prerequisite condition(s) specified in a Surveillance test procedure require entry into the reactor state or specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance test. An exception to SR 3.0.4 automatically allows 24 hours for completion of a Surveillance Requirement from the time the prerequisite test condition(s) are met. If 24 hours is not sufficient to complete the Surveillance Requirement, an appropriate time interval is stated with the exception in the individual Surveillance Requirement within which the Surveillance Requirement must be met after reaching the prerequisite condition(s). Unless otherwise inoperable, equipment or systems are not considered inoperable, and ACTIONS are not entered, solely as a result of complying with an exception to SR 3.0.4. The associated Required Actions become applicable only if the Surveillance Requirements are not met within the time allowed by the exception.

INTENTION ONLY

B 3.1 REACTOR

B 3.1.1 Safety Rod Withdrawal Limit

BASES

BACKGROUND

The Savannah River Site (SRS) K reactor contains two different types of reactivity control assemblies: safety rods and control rods. The safety rods provide sufficient negative reactivity for an immediate shutdown of the reactor from any operating condition. The control rods provide for power control and help offset the reactivity increase caused by xenon decay following REACTOR SHUTDOWN. Additional information relative to the function of control rods during REACTOR SHUTDOWN can be found in Bases B 3.1.3, Reactivity Requirements - Shutdown.

The reactor contains 66 safety rods. The safety rods ensure a rapid shutdown of the reactor when a reactor scram is initiated. The safety rods are fully withdrawn from the core prior to achieving criticality and are held above the core via magnetic clutches. Upon receipt of a scram signal, the magnetic clutches are de-energized and then rapidly demagnetized by a negative pulse voltage (approximately -125 volts). De-energization and demagnetization of the safety rod clutches allow the safety rods to free-fall into the core by gravity (Refs. 1 and 2). Details of safety rod operation are provided in Bases B 3.1.2, Safety Rod OPERABILITY.

A sufficient negative reactivity margin must always be available when operating the reactor, to enable safe shutdown. The safety rods function as an available negative reactivity source to rapidly shut down the reactor upon receipt of a reactor scram signal. When the reactor is critical, only the worth of the safety rods contribute to the shutdown margin. When the reactor is subcritical, the combined worth of the safety rods and inserted control rods keep the reactor subcritical and comprise the shutdown reactivity.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

Reactor scram capability is an essential element in the mitigation of the transients and accidents associated with the TRANSIENT LIMITS and Emergency Cooling Limits. The safety rod withdrawal limit ensures that sufficient negative reactivity is available prior to achieving criticality to shut down the reactor and help maintain the reactor shutdown following a reactor scram.

LCO

All safety rods are required to be completely withdrawn prior to positioning control rods during REACTOR OPERATION. This requirement ensures that prior to reactor criticality, and during subsequent REACTOR OPERATION, the worth of all OPERABLE safety rods is available to shut down the reactor.

APPLICABILITY

The safety rods must be fully withdrawn prior to positioning any control rods for REACTOR OPERATION.

ACTIONS

A.1

Condition A is entered when any safety rod is not fully withdrawn. Operation of the reactor with any safety rod not fully withdrawn or withdrawal of a safety rod when the reactor is critical are unanalyzed conditions and are not permitted. If a safety rod is discovered not fully withdrawn after control rod positioning has begun, the reactor must be in REACTOR SHUTDOWN within 90 minutes. The 90 minute completion time is required to facilitate a CONTROLLED SHUTDOWN and is not provided to attempt withdrawal of the inserted safety rod.

SURVEILLANCE
REQUIREMENTS

SR 3.1.1.1

This surveillance ensures that all safety rods are fully withdrawn from the core prior to control rod positioning for REACTOR OPERATION. There are several methods of verifying that all of the safety rods are withdrawn from

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.1.1 (continued)

the core. Examples of possible surveillance methods include safety rod position lights in the central control room, latch loaded lights in the actuator tower, or the "safety rod unloaded or off up limit alarm" and visual verification of latch positions.

The intent of this surveillance is to periodically verify that all of the safety rods are fully withdrawn from the core and detect any abnormal conditions which might reduce the available shutdown margin provided by the safety rods. Physical entries into the pattern to visually verify safety rod latch positions are limited when safety rods are off their down limit, to minimize the potential for damaging safety rod latch mechanisms. Verification that all safety rods are fully withdrawn from the core must be performed within 1 hour prior to positioning control rods for REACTOR OPERATION, and at least once per 12 hours during REACTOR OPERATION. The surveillance note specifies that SR 3.0.4 is not applicable, to permit entry into the Applicability (REACTOR OPERATION) prior to performing the surveillance.

The 12 hour surveillance frequency provides adequate assurance that all safety rods are fully withdrawn since safety rods are not repositioned during normal REACTOR OPERATION, and since the safety rod unloaded or off up limit alarm provides an alarm if a safety rod is detected in the reactor during REACTOR OPERATION.

REFERENCES

1. RTR-2210, "Improved Safety Rod System - Safety Rod Insertion Time Measurements," July 13, 1984. (U)
 2. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 15.1.3.4, "Reactor Physics Considerations." (U)
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INFORMATION ONLY

B 3.1 REACTOR

B 3.1.2 Safety Rod OPERABILITY

BASES

BACKGROUND

The Savannah River Site (SRS) K reactor contains two different types of reactivity control assemblies: safety rods and control rods. The safety rods provide sufficient negative reactivity for an immediate shutdown of the reactor from any operating condition. The control rods provide for power control and help offset the reactivity increase caused by xenon decay following REACTOR SHUTDOWN. Additional information relative to the function of control rods during REACTOR SHUTDOWN can be found in Bases B 3.1.3, Reactivity Requirements - Shutdown.

The reactor contains 66 Safety Rods. The Safety Rods are used to ensure rapid shutdown of the reactor when a scram is initiated. The Safety Rods are fully withdrawn from the core prior to achieving criticality and are suspended above the core via magnetic clutches. Upon receipt of a scram signal, the positive holding voltage is interrupted and the magnetic clutches are quickly demagnetized by a negative pulse (200 msec pulse duration) voltage (approximately -125 volts) and the safety rods free-fall into the core by gravity. The safety rods are allowed to free-fall for at least 75 inches into the reactor, where upon a DC snubbing voltage (110 ± 2 volts) is applied. The DC snubbing voltage slows the rod descent until the rod is 14 to 29 inches from the bottom limit position (snubbing height, clutch fully engaged) and the B motor package then completes the safety rod insertion, at a rate of 8 feet per minute.

A sufficient negative reactivity margin must always be available when operating the reactor to enable safe shutdown (Ref. 8). The safety rods function as an available negative reactivity source to rapidly shut down the reactor upon receipt of a reactor scram signal. When the reactor is critical, only the worth of the safety rods are credited for shutdown. When the reactor is subcritical, the combined worth of the safety rods and inserted control rods keep the reactor subcritical, and comprise the total core shutdown reactivity.

(continued)

INFORMATION ONLY

Safety Rod OPERABILITY
B 3.1.2

BASES (continued)

APPLICABLE SAFETY ANALYSES

Calculations (Ref. 1) used to establish Safety Rod System Worth assume that 57 safety rods will provide negative reactivity in time to be effective for mitigating the transient and that 6 additional rods will insert within 2 to 30 seconds to provide additional negative reactivity. Calculations used to establish the Flow Instability portion of the Emergency Cooling Limits also assume that 57 safety rods will provide negative reactivity in time to be effective for preventing Flow Instability.

The timing assumed in the safety analysis for insertion of safety rods allows for degradation of safety rod performance such that 57 are assumed to meet the 1.05 second insertion time, 6 will not meet the 1.05 second insertion time but will contribute to the shutdown margin at a later time (i.e., much sooner than required for later transients at 50 to 200 seconds), and 3 adjacent, highest worth safety rods will not insert at all (Ref. 1). The above timing assumed in the safety analysis will be referred to as 57/6/3 in the remaining discussion.

Safety rod OPERABILITY ensures that sufficient negative reactivity is available to shut down the reactor following a reactor scram from full power and maintain the reactor subcritical during the resulting temperature decrease. Assumptions made in the safety analysis relative to safety rod OPERABILITY include (Refs. 1, 2 and 3):

The safety rods will not begin to snub until they have traveled halfway between the fully withdrawn position (safety rod tips located at the bottom of the poison plate) and the snubbing reference height (25 inches above electrical down limit). With the safety rod tips located at the bottom of the poison plate, snubbing will not begin until the safety rods have traveled at least 75 inches from the poison plate into the core upon receipt of a scram signal (Refs. 4 and 5).

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

OPERABLE safety rods will travel at least 75 inches from the bottom of the poison plate into the core within 1.050 seconds following magnetic clutch relay de-energization. The requirement that the safety rods insert 75 inches into the core (as measured from the bottom of the poison plate) within 1.05 seconds following drop-out of the magnetic clutch relay, is sufficient to satisfy the assumptions made concerning safety rod worth versus time used for calculating the Flow Instability portion for Emergency Cooling Limits, and for calculating the Transient-Protection Limits.

LCO

At least 64 safety rods are required to be OPERABLE during REACTOR OPERATION. The safety analysis assumes (based on historical data (Refs. 6 and 7)) that if at least 64 of the safety rods are OPERABLE after entering REACTOR OPERATION, degradation of the safety rod system during operation will be limited such that at least 57 of the safety rods will meet the assumed 1.050 second insertion time, 6 rods will insert after 1.050 seconds but within 2 to 30 seconds, and that 1 (in addition to the 2 inoperable safety rods) safety rod will not insert at all due to a single failure. The (57/6/3) margin is provided to allow for degradation associated with safety rod OPERABILITY during REACTOR OPERATION. This LCO requirement ensures that prior to positioning control rods for REACTOR OPERATION, the safety rods will be capable of providing shutdown margin consistent with assumptions made in the safety analysis relative to safety rod worth and timing.

APPLICABILITY

At least 64 safety rods must be demonstrated OPERABLE prior to positioning control rods for REACTOR OPERATION. The safety analysis accommodates operation of the reactor with up to 2 safety rods assumed to be inoperable.

(continued)

BASES (continued)

ACTIONS

A.1

Condition A is entered when three safety rods are inoperable. A Completion Time of 12 hours is allowed to restore at least one safety rod to OPERABLE status. The 12 hour Completion Time accommodates repairs of safety rod components without significantly reducing reactor safety. Reactor safety is maintained because the likelihood of requiring the inoperable safety rods for mitigation of a transient or accident (coincident with an additional assumed failure) during the very short outage period is small.

B.1

Condition B is entered when more than 3 safety rods are inoperable or when the Required Actions for Condition A have not been satisfied. More than 3 safety rods inoperable exceeds the margin assumed in the safety analysis for safety rod inoperability and could possibly jeopardize safe shutdown of the reactor during a transient or accident. If more than 3 safety rods are inoperable, the reactor must be shut down within 90 minutes. The 90 minute completion time is required to conduct a controlled shutdown of the reactor.

**SURVEILLANCE
REQUIREMENTS**

SR 3.1.2.1

This surveillance demonstrates that the safety rod negative pulse voltage used to demagnetize the magnetic clutches following de-energization of the clutches during a scram is available. The negative pulse voltage must be available to ensure (i.e., initiate release within 0.30 second from scram relay de-energization) that the safety rods will be inserted into the reactor in less than 1.050 seconds. Loss of negative pulse voltage could cause the safety rods to be released too slowly and exceed the 1.050 second insertion time. This surveillance must be performed every 12 hours during REACTOR OPERATION, and is only required during REACTOR OPERATION since the timing of safety rod insertion is an important assumption made in the safety analyses. The 12 hour surveillance frequency is supported by historical performance (negative pulse outage is uncommon, Refs. 6 and 7) and continuous monitoring is provided by alarm.

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RTI ONLY

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.2.2 (continued)

This surveillance demonstrates that the safety rod variable snubbing switches do not initiate snubbing until the safety rods have traveled at least 75 inches. Reference 4 identifies requirements which must be performed to ensure that the snubbing switches are properly calibrated. The snubbing switch actuates at the midpoint between the rod position at the time that a scram is initiated (full up limit with the rod tip at the bottom of the poison plate) and an arbitrarily chosen reference point. It is essential that the reference point for the snubbing be no more than 25 inches above the electrical down limit to ensure that the safety rods begin snubbing at the location assumed in the safety analysis. Premature application of the snubbing voltage can cause incorrect insertion times and snubbing heights. This surveillance must be performed at least every 60 months, after repair, replacement, adjustment or modification to the magnetic clutch/switch assembly, and prior to installing a replacement B motor package. The 60 month surveillance frequency ensures that the mechanical cams and switches comprising the magnetic clutch/switch assembly are properly calibrated and checked for evidence of wear. The surveillance frequency is also based on the nature of the mechanical timing device, which is only subject to wear during safety rod positioning and reactor scrams.

SR 3.1.2.3

This surveillance demonstrates that during a reactor scram under simulated operating conditions, the safety rods will be inserted into the reactor to the extent and within the time assumed in the safety analysis. The blanket gas plenum must be pressurized during performance of this surveillance to simulate actual conditions of REACTOR OPERATION, because the pressure forces moderator up into the safety rod guide tube and increases the SAFETY ROD INSERTION TIMES.

This surveillance frequency is designed to ensure that the safety rods meet the required 1.050 second SAFETY ROD INSERTION TIME subsequent to latching the actuator and within 7 days prior to control rod positioning for REACTOR OPERATION.

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BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.2.3 (continued)

Additional performance of this surveillance is required after modification or repairs are made to the safety rod B motor packages and when more than 2 safety rod insertion times measured during a scram are not within the 1.050 second insertion time requirement. The intent of this surveillance and the frequency is to ensure that the reactor always begins its operating cycle with a complement of at least 64 OPERABLE safety rods so that any degradation that occurs during REACTOR OPERATION will be bounded by the assumptions made (57/6/3) in the safety analysis.

SR 3.1.2.4

This surveillance demonstrates that the safety rods will travel to the bottom limit position within 30 seconds after scram relay de-energization. The safety analysis for the Emergency Cooling System portion of LOCA and for LOPA assumes that all except the 3 highest worth safety rods will contribute to the shutdown reactivity. The 30 second requirement ensures that at least 63 of the safety rods will be contributing to the shutdown reactivity assumed available in the safety analysis (approximately 20 seconds) before they are required. This surveillance must be performed with the same frequency required in SR 3.1.2.3, and is normally conducted in conjunction with SR 3.1.2.3. The same 64 safety rods which meet the insertion time requirements of SR 3.1.2.3 must also satisfy the requirements of this surveillance.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.1.2.5

This surveillance demonstrates that the external, exposed components of the safety rods are free of encumbrances or other visible evidence of abnormal conditions. Since the safety rod mechanisms are exposed in the actuator tower, they are susceptible to many physical obstructions or other conditions which could interfere with proper operation. Specific items which have been identified in Reference 7 as potential problems include:

- Tangled safety rod cables.
- Misalignment of safety rod latch loaded switch.
- Debris (washers) wedged between sleeve and latch.
- Unlocked cable guards out of position

The extent to which visual inspection is practical and prudent, is determined by the operating condition of the reactor and the extent of maintenance performed on the safety rod drive mechanisms during shutdown. The items listed as example problems above may not be detectable at all times during the reactor operating cycle.

This surveillance must be performed within 24 hours prior to control rod withdrawal for REACTOR OPERATION. And this surveillance must be performed within 12 hours following the performance of any safety rod maintenance in the actuator (i.e., replacement of B motor package(s)).

SR 3.1.2.6

This surveillance demonstrates that the mechanical snubbers on all of the safety rods are not deformed. The mechanical snubbers ensure that the safety rods and reactor components are not damaged if the safety rod snubbing voltage should fail during a reactor scram and subsequent safety rod insertion, by absorbing the kinetic energy released when the safety rods free-fall and impact upon the upper portion of the safety rod thimble tubes. Damaged mechanical snubbers must be replaced prior to entering REACTOR OPERATION to prevent possible safety rod binding and to ensure that failure of the snubbing voltage circuitry will not damage the reactor, safety rods, or safety rod thimbles. The surveillance must be performed after performance of SR 3.1.2.3, SAFETY ROD INSERTION TIME testing, and prior to positioning control rods for REACTOR OPERATION.

(continued)

ONLY

Safety Rod OPERABILITY
B 3.1.2

BASES (continued)

REFERENCES

1. DPST-88-865, "Estimates Of The Safety Rod Timing, Worth And Post-Power Peaking In Mark 22 Charges," January, 1989. (U)
 2. DPSP-62-1-3, "Works Technical Monthly Report," March. 1962, pages R332-335 (SECRET/RD).
 3. RTR-2210, "Improved Safety Rod System - Safety Rod Insertion Time Measurements," July 13, 1984. (U)
 4. DPSP-55-25-32, "AH Control System," pages 205-208 (U)
 5. RTR-2029, "Improved Performance Of Safety Rod System," May 3, 1982. (U)
 6. DPST-86-707, "Failure Probability of Individual Safety Rods," October 9, 1986. (U)
 7. RTR-2606, "Safety Rod Information For Technical Specifications," November 15, 1988. (U)
 8. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 15.1.3.4, "Reactor Physics Considerations." (U)
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REACTOR SHUTDOWN ONLY**B 3.1 REACTOR****B 3.1.3 Reactivity Requirements - Shutdown****BASES**

BACKGROUND

The SRS reactors contain two different types of reactivity control assemblies; safety rods and control rods. The safety rods provide a sufficient amount of reactivity for an immediate shutdown of the reactor from any operating condition, and maintain it shutdown during the resulting temperature decrease. The full length control rods provide reactor power control and radial flux shaping, and the partial length control rods provide axial flux shaping and fine tuning of the radial flux. The control rods, in addition to the safety rods, also maintain the reactor subcritical as xenon decays during REACTOR SHUTDOWN, and following any postulated reactivity addition transients or accidents analyzed during shutdown.

The reactor contains 66 safety rods and 61 control rod clusters. The safety rods contain boron carbide as the neutron absorbing material. Each control rod cluster has seven locations for control rods, and is made up of five full length control rods which contain neutron absorbing material over the entire length of the rod, and two partial length control rods which contain neutron absorbing material over the bottom half of the rod only. Three of the full length control rods contain lithium (one "strong" rod and two "weak" rods) as the neutron absorbing material. The other two full length control rods contain cadmium as the neutron absorbing material. The two partial length control rods (one "strong" rod and one "weak" rod) contain lithium as the neutron absorbing material. There are a total of 122 cadmium control rod locations, however, two of these locations contain Cf-252 sources, so the total number of full length cadmium control rods is 120.

The worth of 63 safety rods (section 15.4.4.2 of the Safety Analysis Report (SAR) assumes that the three highest worth safety rods are not inserted into the core upon receipt of a scram signal) is sufficient to shut down the reactor from any operating condition and to maintain it shutdown during the resulting temperature decrease. During REACTOR OPERATION, the shutdown margin is maintained by ensuring

(continued)

BASES

BACKGROUND
(continued)

that all of the safety rods (66) are fully withdrawn. The minimum worth of 63 safety rods is 0.0183 delta k effective/k effective (Mark 22 charges). Following REACTOR SHUTDOWN, the safety rods alone may not be sufficient to maintain the shutdown margin as xenon decays. If 63 safety rods are FULLY INSERTED and no control rods are inserted, the reactor will not return to critical due to xenon decay for at least 4 hours. However, the insertion of control rods and safety rods is sufficient to maintain the reactor subcritical indefinitely provided no more than three rods (total combined safety rods and control rods) are not FULLY INSERTED.

APPLICABLE
SAFETY ANALYSES

The safety analyses assumes that a reactor containing a MK22 charge scrams from 1500 MWs (a value that is, conservatively, higher than is achievable under current operating limits) with 432 kWd reactor exposure and is subsequently restarted at the peak of the post-shutdown xenon transient. The reactor IMMEDIATELY scrams again, with no appreciable xenon burn-out. Additional assumptions made in the analysis are:

- 1) The MARGIN OF CONTROL (MOC) of the reactor is positive when the scram occurs. If a positive MOC is verified at the beginning of a Mark 22 cycle, then insertion of the withdrawn portion of the control rods will always be sufficient to shut down the reactor and maintain the reactor subcritical during xenon decay and the resulting temperature decrease at any future point in the cycle. A positive MOC is verified at the beginning of a Mark 22 cycle by compliance with Technical Specification 3.1.4, Reactivity Anomaly.
- 2) only 63 safety rods fail during the scram (the three highest worth rods are assumed to fail),
- 3) all of the control rods fail to drive into the reactor at the time of the scram,

If 63 safety rods are FULLY INSERTED and no control rods are inserted, the reactor will not return to critical for at least 4 hours due to xenon decay.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES
(continued)

The full insertion of safety rods and full length and partial length control rods is sufficient to compensate for the resulting positive reactivity addition due to complete xenon decay, and maintain the reactor subcritical indefinitely, provided no more than three rods (total combined safety rods and control rods) are not FULLY INSERTED.

LCO

The requirement to have no more than three rods (total combined safety rods and control rods) not FULLY INSERTED ensures that subcriticality will be maintained as xenon decays.

APPLICABILITY

During REACTOR SHUTDOWN, the safety rods and full length and partial length control rods are required to be inserted such that no more than three rods (total combined safety rods and control rods) are not FULLY INSERTED prior to shutdown time exceeding 4 hours. This ensures that a sufficient amount of negative reactivity is available to ensure subcriticality as xenon decays.

ACTIONS

A.1

If more than three rods (total combined safety rods and control rods) are not FULLY INSERTED within 4 hours after shutdown, this action requires that the SSS be actuated IMMEDIATELY. The worth of the SSS is more than sufficient to restore the shutdown margin.

It has been demonstrated in Reference 3 that the worth of the safety rods, plus the worth of the full length and partial length control rods is sufficient to maintain subcriticality as xenon decays, provided no more than three rods (total combined safety rods and control rods) are not FULLY INSERTED.

(continued)

INTENTION ONLY

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.1.3.1

Verifying that no more than three rods (total combined safety rods and control rods) are not FULLY INSERTED, within 4 hours after entering REACTOR SHUTDOWN and each 24 hours thereafter with the actuator latched and power available to the control and safety rod drive motors, ensures that the safety rods' and control rod's contribution to the shutdown margin is available and maintained. This surveillance Frequency will preserve the safety analyses assumptions. The surveillance frequency of 24 hours is consistent with those used in PWR Technical Specifications for similar surveillances. This Surveillance Requirement does not have to be performed when the actuator is unlatched and raised, since the requirements of the LCO would be met prior to unlatching and raising the actuator, and no other mechanism would be available to withdraw the rods.

REFERENCES

1. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 15.4.4.2, "Other Reactivity Addition Accidents During Shutdown." (U)
 2. WSRC-TR-91-036, "Control Rod Insertion Requirements For Reactor Shutdown", March 1991. (U)
 3. SRT-APG-920076, "Physics Analysis of Three Missing Rods", June 1992. (U)
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INFORMATION ONLY

B 3.1 REACTOR

B 3.1.4 Reactivity Anomaly

BASES

BACKGROUND

The charge specific core design includes estimates of core reactivity as indicated by predictions of expected critical positions. Comparison of expected critical positions with actual critical positions provides an indication of the accuracy of the analytical model of the core and a validation of the precision of the manufacturing process. When the actual and predicted critical positions agree within an allowable uncertainty interval, the as-built core can be presumed to behave as predicted by the analytical model. Conversely, if the actual core reactivity is outside of the expected reactivity uncertainty interval, the actual core performance as projected in the safety analysis is suspect until the consequences of the off-nominal reactivity condition are evaluated and found to be acceptable.

In order to ensure that the actual core reactivity characteristic is completely within the bounds of the safety analysis, the reactivity deviation from nominal is measured at both xenon free and equilibrium xenon conditions at the beginning of the charge lifetime and periodically as the charge is depleted. The initial xenon free reactivity deviation is "normalized out" of the subsequent reactivity comparisons so that only the component of the reactivity deviation associated with the temperature and neutron poison and burnup are measured.

APPLICABLE SAFETY ANALYSES

Initial conditions assumed in safety analyses are selected to bound the consequences of postulated events. One of the assumptions implicit in the safety analysis is that the predicted core response will accurately characterize the response of the as-built core to operating transients, and postulated events. In order for this assumption to be valid, some assurance that the as-built core will respond as modeled in the safety analysis is required. One of the parameters used to ensure that the core will respond as predicted is the as-built core reactivity deviation from the nominal core.

(continued)

INFORMATION ONLY

Reactivity Anomaly
B 3.1.4

BASES

**APPLICABLE
SAFETY ANALYSES
(continued)**

The reactivity characteristic of the core is important in all transients that either involve the inadvertent addition of positive reactivity, take credit for negative reactivity feedback prior to scram or are terminated by a reactor scram.

LCOs

An analysis of the initial critical positions of cores with Mark-22 charges has shown that the actual critical position of a nominal charge will be within $\pm 1.57\%$ ΔK of the expected critical position at a 95/95 confidence level (Refs. 1 and 2).

In addition to the initial critical position check above, the LCO requires the initial temperature and neutron poison reactivity and the reactivity change with burnup to be monitored and compared to predicted values. The limit for these reactivity deviations is less than the limit for the initial xenon free reactivity deviation because there are fewer parameters available to account for the change. For example, any reactivity deviation caused by fuel assembly content or location would not be included in the temperature and neutron poison or burnup reactivity change because it would have already been included in the initial xenon-free critical position and would not impact the temperature and neutron poison reactivity and the reactivity change with burnup (Ref. 3).

The LCO limit of 1.0% ΔK for the temperature and neutron poison reactivity and the reactivity change with burnup is based on engineering judgement and is consistent with values used in the commercial industry.

The actual reactivity change with burnup will be compared to the predicted core reactivity change with burnup. The actual value will be within 1.0% ΔK of the predicted value. The 1.0% ΔK limit is based on operating experience at SRS.

Finally the MARGIN OF CONTROL will be $> 0\%$ ΔK . Maintaining MARGIN OF CONTROL will ensure that the plant is operating within the boundary set forth by safety analysis.

(continued)

INFORMATION ONLY

BASES (continued)

APPLICABILITY

During REACTOR OPERATION the reactor is vulnerable to reactivity addition events and other transients that are mitigated by a reactor scram. To ensure that the assumptions of the safety analyses remain valid, the actual core reactivity deviation is required to be within the LCO limits during REACTOR OPERATION.

ACTIONS

A.1

This condition applies when the initial xenon free reactivity deviation is not within the LCO limit. In this event the reactor is required to be in REACTOR SHUTDOWN within 90 minutes. The Completion Time of 90 minutes is a reasonable time, based on operating experience, to reach REACTOR SHUTDOWN from full power without challenging safety systems or operators.

B.1 and C.1

These Conditions apply when the temperature and neutron poison reactivity (Condition B) and the reactivity change with burnup (Condition C) is not within the LCO limit. In either case, an Events Report must be prepared in accordance with Administrative Controls section 5.6. Required Actions B.1 and C.1 require that the impact of the reactivity deviation on the safety analysis must be evaluated. It is assumed that this evaluation can be completed within a period of 7 days and that continued reactor operation is acceptable until the evaluation is completed. Ideally, at the conclusion of the evaluation, the current thermal-hydraulic limits would continue to be applicable. A new predicted temperature and neutron poison reactivity and the reactivity change with burnup would be available and the actual core reactivity characteristic would be within the LCO limit.

(continued)

INFORMATION ONLY

BASES

ACTIONS (continued)

D.1 and D.2

With the MARGIN OF CONTROL less than the limit, Required Actions D.1 and D.2 require that the reactor be placed in REACTOR SHUTDOWN within 90 minutes and following entry into shutdown that action be initiated to fully insert all safety and control rods.

E.1 and E.2

If the Required Actions of Condition B or C cannot be met within the required Completion Times, the reactor must be placed in a condition where the LCO does not apply. The Completion Time of 90 minutes is a reasonable time, based on operating experience, to reach REACTOR SHUTDOWN from full power without challenging safety systems or operators. Required Action E.2 ensures that actions are taken to insert all safety and control rods following entry into REACTOR SHUTDOWN. This action ensures that the reactor will remain shutdown following the addition of positive reactivity due to temperature decrease and xenon decay.

SURVEILLANCE REQUIREMENTS

A note is provided for each of the following Surveillance Requirements which states that SR 3.0.4 is not applicable. This note is required to permit entry into the Applicability (REACTOR OPERATION) prior to performing the surveillances. This exception to SR 3.0.4 is required to achieve the required conditions for performing the surveillances.

SR 3.1.4.1

This surveillance verifies that the actual core reactivity change with burnup is within an acceptable allowance of the predicted core reactivity change. The frequency of once each 50 KMWd is based on engineering judgement and is similar to the value used in the commercial nuclear reactor industry in the Standard Technical Specifications.

(continued)

INFORMATION ONLY

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.2

This surveillance verifies that the actual reactivity required to compensate for temperature and neutron poisons is within an acceptable tolerance when compared to the predicted value. The surveillance is only required to be performed once per REACTOR SUBCYCLE during the initial power ascension. With the initial temperature and neutron poison reactivity is correctly predicted, it is assumed that subsequent temperature and neutron poison reactivity will also be correctly predicted.

SR 3.1.4.3

This surveillance verifies that the initial xenon free critical position is within an acceptable tolerance when compared to the predicted value. The surveillance is only required to be performed once during the initial REACTOR SUBCYCLE startup.

SR 3.1.4.4

This surveillance verifies that the MARGIN OF CONTROL is greater than the acceptable value. The surveillance must be performed once during the initial REACTOR SUBCYCLE startup and every 50 KWWDs.

REFERENCES

1. DPST-88-878, "Band of Acceptability on Deviation Between Predicted and Observed Initial Critical Rod Configurations in Mark 22 Charges", Nov 14, 1988. (U)
 2. DPST-88-859, "Uncertainties and Bias in Mark 22 MARCO Calculations", Nov. 1988. (U)
 3. RTR-2593, "Mark 22 Start-up Reactivity Uncertainty", Dec. 5, 1988. (U)
 4. WSRC-SAR-RX-K; Savannah River Site Production Reactor Safety Analysis Report, Section 15.1.3.4, "Reactor Physics Considerations." (U)
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B 3.1 REACTOR

INFORMATION ONLY

B 3.1.5 Assembly Effluent Temperatures - TRANSIENT LIMITS

BASES

TRANSIENT LIMITS are provided to limit consequences of nuclear or hydraulic transients that might be caused by certain incidents. Specifically, TRANSIENT LIMITS on the coolant assembly effluent temperatures are specified to avoid: (1) the onset of significant voiding with the resultant dryout of fuel tube surfaces and increase in fuel temperature (OSV limit), or (2) the exit temperature exceeds saturation for the local pressure in the fluid stream exiting through the shell holes (TSAT limit) (Ref. 1).

The hypothetical incidents analyzed in calculating TRANSIENT LIMITS are selected on the basis of (Ref. 2):

- o Ability to define the hypothetical incident and likelihood of the incident.
- o Magnitude of potential consequences.
- o Ability to specify boundary conditions that provide protection against a wide range of incidents within the framework of the incidents selected.

Exceeding a TRANSIENT LIMIT during a process fluctuation would not in itself produce any measurable consequences (provided that the TECHNICAL LIMIT is not exceeded for a significant duration); however, such a fluctuation could increase the vulnerability to damage because less margin would be available to allow for occurrence of a serious incident which has the potential for bulk boiling.

REFERENCES

1. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 15.1.2.3.1, "Flow Instability and Assembly Effluent Temperature." (U)
2. DPSTM-110. Technical Manual - Effluent Temperature and BOSF_N Limits for Fuel and Target Assemblies and Reactor Effluent Temperature Limits. (U)

B 3.1 REACTOR

INFORMATION ONLY

B 3.1.6 Transient Reactor Burnout Risk - TRANSIENT LIMITS

BASES

Burnout risk (BOR) limits are specified to avoid significant release of fission products caused by local film boiling burnout of fuel and target surfaces during normal operation or during specified incidents. Local film

boiling burnout could occur when the critical heat flux is reached in the nucleate boiling regime. Beyond the critical point heat transfer coefficients rapidly decrease and this could cause the surface temperature to rise above the material's melting point.

The TRANSIENT LIMITS lessen consequences of nuclear or hydraulic transients that might be caused by certain incidents. Specifically, TRANSIENT LIMITS applied to BOR is specified to avoid significant release of fission products to the moderator because of film boiling burnout that occurs during specified incidents.

The degree of protection afforded by limits on BOR depends upon the particular incident involved, the effectiveness of scram circuits, and the calculation of damage from film boiling. The acceptable consequences for operation at the TRANSIENT LIMIT are, therefore, defined in terms of the specific incident and the operability of scram circuits without regard to the mechanism for damage or to the parameter that must be limited to achieve compliance with the damage restrictions. Applicable limits other than those on BOR for individual fuel and target assemblies are discussed in Reference 2.

The BOR is a statistical possibility subject to manufacturing tolerances. Although BOR is reduced to a negligible risk by use of the Flow Instability (FI) limit, the possibility of local burnout exists during normal operation or transients. Therefore, the BOR limit is conservatively being retained with its historical basis and it is analogous to the limits placed on leaks in fuel rods in commercial reactors.

The BOR TRANSIENT LIMITS of 3 MW corresponds to melting a volume of fuel that would contain about 10^7 Ci of fission products.

(continued)

INFORMATION ONLY

BASES (continued)

Fission product activity releases of about 10^5 curies to the moderator have been experienced with SRS charges with no undesirable safety effects. This amount of activity release corresponds to about 0.5 % of the fission product activity in one SRS fuel assembly. Analysis of radiation levels from below-grade piping shows that total activity releases to the moderator of 10^7 Ci will impose no undue safety hazards or significant interruption of reactor operation from excessive downtime due to high radiation levels which prevent maintenance and/or operation over periods greater than that experienced for other types of maintenance. Releases up to 10^8 curies to the moderator would impose no undue safety hazards and would be tolerable from the standpoint of downtime; i.e., several weeks to several months would be required because of the need to replace purification equipment during the cleanup operation and because of greater delays in fuel discharge activities. If the volatile fraction of this activity were released from the reactor hydraulic system, offsite thyroid and whole body doses are calculated to be < 3 R under adverse meteorological conditions.

BOR is monitored for every assembly every few minutes rather than continuously monitoring one assembly. The maximum BOR of any one assembly is limited only by its contribution to total allowable risk. The minimum $BOSF_N$ in this reactor is not controlled.

The hypothetical incidents analyzed in calculating TRANSIENT LIMITS are selected on the basis of (Ref. 1):

- o Ability to define the hypothetical incident and likelihood of the incident.
- o Magnitude of potential consequences.
- o Specifying boundary conditions that provide protection against a wide range of incidents within the framework of the incidents selected.

Process fluctuation beyond a TRANSIENT LIMIT during normal operation would not in itself produce any measurable consequences (provided that the duration above the TECHNICAL LIMITS is not significant); however, such a fluctuation could increase the vulnerability to damage because less margin would be available to allow for occurrence of a serious incident.

(continued)

INFORMATION ONLY

BASES (continued)

REFERENCES

1. DPSTM-110. Technical Manual - Effluent Temperature and BOSF_N Limits for Fuel and Target Assemblies and Reactor Effluent Temperature Limits. (U)
 2. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 15.1.2.3.2, "Burnout Risk." (U)
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INFORMATION ONLY

B 3.1 REACTOR

B 3.1.7 Reactor Effluent Temperatures - TRANSIENT LIMITS

BASES

Transient limits on reactor effluent temperature provide protection against cavitation damage to the Process Water System during transients causing excessive reactor effluent temperatures. Cavitation can result when fluid velocity and/or temperature increases sufficiently to cause vaporization in low pressure locations. Additionally, sudden reductions in reactor tank pressure alone can cause cavitation. Boundary materials in the immediate vicinity are subjected to rapidly repeated stress reversals and may eventually fail through fatigue, the first sign of which is surface pitting.

In the K reactor cavitation occurs in the elbows just downstream from the reactor outlet nozzles, at a lower temperature than in the pumps. Reactor effluent temperature, as a controlled variable, differs from assembly effluent temperature and BOR in that excessive reactor effluent temperature is not directly related to fuel/target damage or to sudden and significant reactor damage. Reactor transients causing excessive reactor effluent temperature can produce mechanical damage to the process water circulation pumps and their suction lines, and may result in direct damage to the fuel/target assemblies. Excessive reactor effluent temperature can contribute to fuel/target damage only insofar as cavitation can reduce process water flow and thus cause limits on assembly effluent temperature and BOR to be exceeded.

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BASES

INFORMATION ONLY

(continued)

Although a reactor effluent temperature limit is established to preclude cavitation, it is less restrictive than the FI assembly effluent temperature limit. Section 4.4.1 of the Safety Analyses Report (Ref. 1) contains a plot of the cavitation reactor effluent temperature limit as a function of total process water flow. The higher the flow, the lower the reactor effluent temperature limit. For the FI limit, the Onset of significant voiding (OSV) component increases with decreasing flow while the TSAT component increases with decreasing flow. Therefore, a comparison of the FI limit and the cavitation limit, shown in Table 15.1-3 (Ref. 2), is made at two flow rates, 300 and 350 gpm, which is representative of the flow zones in the center of the reactor. The table shows that the FI limit is always less than the cavitation limit for a given inlet temperature, gas pressure and flow rate. If the design limit is met, so is the cavitation limit.

REFERENCES

1. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 4.4.1, "Thermal - Hydraulic Limits." (U)
2. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 15.1.2.3.3, "Cavitation and Reactor Effluent Temperature." (U)

B 3.2 REACTOR

INFORMATION ONLY

B 3.2.1 Full Length Control Rod Positioning (Radial Flux Control)

BASES

BACKGROUND

The SRS reactors are designed to ensure the health and safety of the public and the workers at the plant. One of the methods used to satisfy this criteria is reactivity control. At SRS reactivity is controlled by three systems: control rods, safety rods, and the Supplementary Safety System.

The three systems perform different functions. Control rod functions are dependent on reactor state. During REACTOR OPERATION the control rods are used to control overall power, local power, and axial power distributions. During REACTOR SHUTDOWN the control rods counter the effect of the reactivity insertion caused by xenon decay. The safety rods are discussed in B 3.1.1 and B 3.1.2, SSS is discussed in B 3.5.2.1 and B 3.5.2.2

To control reactivity the control rods are inserted into or withdrawn from the core. The two types of control rods, full length and partial length, control reactivity in different ways. The function of the full length control rods is to control radial flux distribution. The control rods are grouped together and motorized to allow precise radial flux control.

Seven control rods are grouped together to form a cluster. Each cluster consists of five full length rods and two partial length rods. Each control cluster is housed in a septifoil.

Each control rod has a drive motor and can be moved individually. However, during normal operation rod clusters are grouped together into three gangs. Gang I clusters are located in the center circular region of the core. Gang II clusters are in a concentric circular region about gang I. Gang III clusters are in a concentric circular region about gang II. This arrangement allows symmetrical movements of clusters within a gang. More precise and independent control of rod movement is achieved by trim controls. Each cluster has two trim controls: one for full length rod movement and one for partial length rod movement. The trim controls allow the operator to control flux distortions.

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BASES (continued)

INFORMATION ONLY

**BACKGROUND
(continued)**

The full length rods control three radial flux distortions: tilt, hump, and dish. Tilt flux distortion is the situation where flux on one side of the core is significantly higher than the flux on the exact opposite side. Trimming the full length rods in where the high flux occurs and trimming the full length rods out where the low flux occurs would reduce tilt distortion. Hump flux distortion is the situation where the flux at the central region of the core is significantly higher than the normal flux level for that region. To control hump distortion gang I would be trimmed into the core and gang III would be trimmed out of the core. Dish flux distortion is the situation where the flux at the central region of the core is significantly lower than the normal flux level for that region. To control dish distortion gang I would be trimmed out of the core and gang III would be trimmed into the core.

The use of cadmium control rods as INTEGRAL CONTROL RODs is prohibited. This ensures consistency of operation with analysis assumptions about worth of cadmium rods to help ensure long term REACTOR SHUTDOWN. Cadmium control rods are assumed to be available to insert after safety rod scram for shutdown. Cadmium rods inserted in sequence do not violate this requirement.

The use of up to six full-length Lithium control rods as INTEGRAL CONTROL RODs is permitted provided: 1) Only strong control rods are used as INTEGRAL CONTROL RODs, 2) No more than two INTEGRAL CONTROL RODs are inserted in any one REACTOR SYSTEM, 3) No INTEGRAL CONTROL RODs are inserted in gang 1, 4) All INTEGRAL CONTROL RODs are FULLY INSERTED, and 5) No strong partial rods are used without a weak partial in the same cluster. The results of analyses performed show that use of INTEGRAL CONTROL RODs with the above restrictions will not cause an increase in reactivity worth for any accident assumed in the Safety Analysis Report (SAR).

**APPLICABLE
SAFETY ANALYSES**

Restrictions on the positioning of full length control rods are required under defined high power conditions to ensure that the uncertainty in monitoring, and thus in controlling, assembly axial power distributions is within the safety analysis assumptions. Restrictions are also required at lower power and during startup to ensure that the axial power shapes, transient and final worth of safety rods, post-scrum power peaking, and the ability of the control rods to maintain shutdown are within the safety analysis assumptions.

(continued)

BASES (continued)

INFORMATION ONLY

APPLICABLE SAFETY ANALYSES (continued)

Loss of coolant accident (LOCA) analyses are discussed in Chapter 15 of Reference 2. BURNOUT RISK (BOR) safety limits, applicable during steady state operation, are discussed in Reference 2. BOR analyses are performed for Transient-Protection Limit (TPL) events, which are discussed in Chapter 15 of Reference 2. For all of those events, the analyses of Reference 2 assume that the reactor is shut down by a scram. Control rod and gang withdrawal events are included in the TPL events in Chapter 15 of Reference 2.

Short-term shutdown following scram, and long-term shutdown following insertion of control rods, are inherent assumptions in the safety analysis.

This Technical Specification requires rod movement to correct control rod mispositioning when it occurs. The safety analysis bounds all cases of such corrective rod motion, including the effects of power maldistribution due to the rod mispositioning, when the corrective action is initiated under otherwise normal operating conditions. Control rod misalignments are discussed in SAR Chapter 15.1.3.4.

The use of INTEGRAL CONTROL RODS could potentially affect reactivity in one of two ways: changing the reactor flux shape in a manner which would worsen reactivity effects in other clusters or, due to the increase in Lithium in the affected cluster, increase the reactivity worth of the Loss of Control Rod Cooling accident (LCRC).

For the first case, the perturbations to the flux shape are either caused by integral rods near the cluster of interest (in which case they depress the flux and thus reduce reactivity), or they are in the diagonally opposed REACTOR SYSTEM from the point of interest (in which case they increase the tilt). Flux perturbations in any REACTOR SYSTEM causing flux shapes outside of those specified in the Technical Specifications would place the plant in conditions for which ACTIONS are specified. As a result, there are no additional effects which must be accounted for specifically due to the use of INTEGRAL CONTROL RODS.

For the second case, calculations were performed concerning the introduction of a strong integral rod to a cluster which then underwent a LCRC event. The results of these calculations showed that there is no increase in the reactivity values for any of the safety analysis accidents due to the presence of strong integral rods provided LCO 3.2.1 requirements are met.

INFORMATION ONLY

BASES (continued)

LCOs

Adherence to the requirements for full length control rod positioning, limits operation to conditions which have been analyzed. This ensures that the core and assembly axial power shape measurement uncertainty is within the assumptions of the safety analyses for flow instability during LOCA and loss of pumping accident (LOPA), for reactor BOR, and for all "Transient-Protection Limit" events which occur with reactor conditions above the Power Shape Applicability Limits (LCO 3.2.1 Requirements A through J and L). Adherence to the requirements also ensures that the axial power shapes and safety rod effectiveness for events occurring during operation with conditions below the Power Shape Applicability Limits and during startup are within the safety analysis assumptions (LCO 3.2.1 Requirement A).

APPLICABILITY

Adherence to LCOs 3.2.1 is required during REACTOR OPERATION. The specific bases of the applicability of different requirements are discussed below.

Control rod positioning limits are required to ensure the accuracy of assembly radial power distribution measurements.

The safety analyses on which these specifications are based (Ref. 3 and 4) were performed for Mark 22 charges. Thus LCO 3.2.1 is applicable only to Mark 22 charges.

(continued)

INFORMATIONAL ONLY

BASES (continued)

ACTIONS

General Bases

Gross violation of the full length rod positioning requirements could produce power distributions which are outside the safety analysis assumptions, and at the same time introduce unaccounted-for uncertainty into the measurement of those assembly power distributions.

Restoring the control rods to within the limits within the required Completion Times will limit the time of operation with a power distribution which may be outside the bounds assumed in the safety analysis.

Completion Times

The defined completion times of 8 hours for rod motions are based on operating experience and permit carefully planned and deliberate action to avoid erroneous or unnecessary further perturbation of the power distribution.

If correction cannot be made within the 8 hour time frame, REACTOR SHUTDOWN is required in 90 minutes.

Multiple Conditions

A Condition requiring an Action corresponds to violation of one of the requirements of the LCO.

If the Required Actions are not completed within the required Completion Times, then placing the reactor in REACTOR SHUTDOWN within an additional 90 minutes will limit the time of operation with a power distribution or available SHUTDOWN MARGIN which may be outside the bounds assumed in the safety analysis. Operation without the ability to properly manage control rods is unanalyzed.

SURVEILLANCE
REQUIREMENTS

SR 3.2.1.1

Visually verify that every full length control rod is latched to the cable connected to a motor. This inspection will be done within 4 hours after latching the actuator and prior to REACTOR STARTUP.

The visual check is done to confirm the control rod will move when the motor is energized. The time of 4 hours was selected based on operator experience.

(continued)

INFORMATION ONLY

BASES (continued)

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.2.1.2

TILT RATIOS are verified to be within their limits at the frequencies of 1 hour following a ROD MOVE and every 12 hours.

This test alerts operators of unexpected core power distribution changes that may, in time, challenge the core thermal-hydraulic limits. When TILT RATIOS are not within limits the operator enters Condition L.

Monitoring TILT RATIO at the frequencies above limits the time of operation with the radial power shape measurement uncertainty beyond the safety analysis assumptions. Dramatic deviations of the TILT RATIOS would result in operator alarms and/or a reactor scram. The uncertainties in the tilt ratios have been taken into account.

SR 3.2.1.3, SR 3.2.1.4, SR 3.2.1.5, SR 3.2.1.6, SR 3.2.1.7,
SR 3.2.1.8, SR 3.2.1.9, SR 3.2.1.10, and SR 3.2.1.11

Monitoring of the position of affected clusters or rods, during the frequencies in SR 3.2.1.3 through SR 3.2.1.11 which could result in violation of the rod positioning requirements, limits the time of operation with the axial power shape measurement uncertainty beyond the safety analysis assumptions. One hour is acceptable because of the unlikelihood of an erroneous rod motion sufficiently adverse to violate the safety analysis assumptions.

For those cases in which the trim of an inserted rod is changed, these surveillances are normally performed using information available in the control room (SR 3.2.1.4, SR 3.2.1.5, SR 3.2.1.6, SR 3.2.1.7, and SR 3.2.1.7).

For those cases in which the control status of an individual rod is changed, i.e. the rod is changed from fully withdrawn to inserted, or from inserted to fully withdrawn, or from an INTEGRAL CONTROL ROD to not an INTEGRAL CONTROL ROD or vice versa, these surveillances are normally performed by direct observation of the control rod upper latch positions at the +66 foot level (SR 3.2.1.9, SR 3.2.1.10, SR 3.2.1.12). Such control rod motions have a higher potential for introduction of position errors.

(continued)

Full Length Control Rod Positioning (Radial Flux Control)
B 3.2.1

BASES (continued)

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.2.1.3, SR 3.2.1.4, SR 3.2.1.5, SR 3.2.1.6, SR 3.2.1.7,
SR 3.2.1.8, SR 3.2.1.9, SR 3.2.1.10, and SR 3.2.1.11
(continued)

Verifying that each control rod positioning requirement is met, by surveillance of all rods controlled by that requirement at an appropriate frequency, limits the time of operation with the axial power shape measurement uncertainty or available shutdown reactivity beyond the safety analysis assumptions, should the requirement be violated for any reason. The appropriate frequency is determined based on experience of the probability of an error or malfunction, considering the other surveillances triggered by specific actions (as discussed in the foregoing sections).

Surveillances concerned with LCO 3.2.1 requirement A (SR 3.2.1.1) have been discussed. The following discussion addresses fixed-frequency surveillances which cover the remaining requirements, in order of decreasing frequency.

These surveillances are normally performed using information available in the control room.

SR 3.2.1.3, SR 3.2.1.4, and SR 3.2.1.5

Full rod motion by gangs using the N unit is frequent, and often virtually continuous. A surveillance frequency of 12 hours for SR 3.2.1.3. A frequency of every 12 hours and within 1 hour following ROD MOVE for SR 3.2.1.4 is required to ensure that the rod insertion remains within the limits of Table 3.2.1-1.

The frequencies are adequate because full rod motion is a slow, gradual process and exceeding full rod insertion limits would be a slow event.

SR 3.2.1.6 and SR 3.2.1.7

Motion of individual CONTROLLING RODs in gangs 1 and 2 is not a normal procedure. It would occur only in the unlikely case of operator error or through equipment malfunction. Most equipment malfunctions would be indicated by the rod fault system, if it is OPERABLE. Errors in placement of rods within a gang 3 group would be detected during performance of SR 3.2.1.7. Motion of individual partial rods or pairs of rods takes place every shift when partial rods are i.e. trim is adjusted.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.2.1.6 and SR 3.2.1.7 (continued)

A surveillance frequency of 12 hours ensures that there is no difference in insertion of inserted full length rods within any gang, or of CONTROLLING RODs within gang 1, gang 2, or any cluster within any gang 3 cluster group.

The frequencies of 1 hour following a ROD MOVE and 12 hours is adequate because control rod motion is a slow, deliberate process. By procedure, individual control rod positions are always verified following motion and the control rod fault system (if OPERABLE) will alarm for any control rod which is not at its CONTROL ROD DEMAND POSITION.

SR 3.2.1.8

When untilt trim is required, motion of full rods in a gang 3 cluster group relative to the rest of gang 3 is required; this is done by trimming the position of each rod in the group in turn. A surveillance frequency of 12 hours ensures that the difference between rod insertions within gang 3 is the same.

A frequency of 12 hours is adequate because gang 3 full rod trim motion is a slow, deliberate process and exceeding the trim limit rod insertion limits would be a slow event.

SR 3.2.1.9, SR 3.2.1.10, and SR 3.2.1.11

Seven days prior to REACTOR STARTUP the verification of control rods consisting of their required CONTROL ROD SEQUENCE must be preformed. This surveillance verifies the CONTROL ROD SEQUENCE transformers for all clusters are wired properly for each corresponding control rod, thus ensuring safe REACTOR OPERATION.

(continued)

BASES (continued)

INFORMATION

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.2.1.12

The inadvertent insertion of a control rod resulting in flux shapes outside of those specified in the Technical Specifications would place the plant in conditions for which ACTIONS are specified. A strong full length control rod may be used out of sequence (inserted as an INTEGRAL CONTROL ROD) for power tilt control. Verification that no more than six full length control rods are inserted as INTEGRAL CONTROL RODS is made 1 hour following a ROD MOVE and every month. Verification is also required at the same frequency to show that 1) any rod inserted as an INTEGRAL CONTROL ROD is a strong control rod, 2) no more than two of the rods inserted as INTEGRAL CONTROL RODS are in any one REACTOR SYSTEM, 3) none of the rods inserted as INTEGRAL CONTROL RODS are in gang 1, and 4) All INTEGRAL CONTROL RODS are FULLY INSERTED. The surveillance frequencies are sufficient to limit the time of operation with possible violation of the safety analysis assumptions caused by either inserting a rod other than a strong rod as an INTEGRAL CONTROL ROD or by inserting a strong rod as an INTEGRAL CONTROL ROD in a manner outside the bounds of the safety analysis (i.e., greater than six rods inserted, more than two rods inserted in a single REACTOR SYSTEM or any rod inserted in gang 1).

SR 3.2.1.13

The control rod Fault Alarm System provides visual indication at the T-amplifier of the control rod and in the control room of a deviation greater than 4 vu of a rod position from its CONTROL ROD DEMAND POSITION. Surveillance on a 1 month frequency provides adequate verification of its OPERABILITY because of the relatively low failure rate of these systems. Control rod fault system failures have historically occurred less than one per reactor year.

(continued)

INFORMATION ONLY

BASES (continued)

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.2.1.14

This surveillance is required only when power is less than or equal to ISAL, but not required when controlling RSF per SR 3.2.1.15. This surveillance ensures when power is increasing during power ascension that all three gangs are on the same CONTROLLING ROD (i.e., if gangs 1,2, and 3 all have a CONTROL ROD SEQUENCE number of 3, they have a rod content of 2 gm/ft Li-6). The Correlation between Sequence Number and Rod description is presented in the Definition of CONTROL ROD SEQUENCE. The other requirement which satisfies the surveillance states the CONTROLLING ROD(s) in any gang are within 100 vu of any other gang CONTROLLING ROD when transitioning to the next rod in the CONTROL ROD SEQUENCE. The latter requirement allows safe transitioning when the operator passes through a control rod tip (a control rod is nearly out of the core when another begins to pull up off its down limit). The requirement is presented in two parts due to mechanical interlocks concerning control rod withdrawal, the 3 gangs CANNOT have the same CONTROLLING ROD when transitioning to the next rod. Engineering judgement has deemed that the surveillance frequency of 12 hours is acceptable to ensure operation is within the safety analysis bounds.

(continued)

BASES (continued)

INFORMATION ONLY

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.2.1.15

This surveillance is not required if maintaining the same CONTROLLING ROD in each gang per SR 3.2.1.14. To ensure the flux shape is within safety analysis bounds, a surveillance of the RSF must be within the stated bands and greater than or equal to the INITIAL RSF. The INITIAL RSF should not be obtained before Low Power (approx. 75 MW) has been established, since previous operating experience has shown unrepresentative RSF values at Initial Low Power (approx. 40 MW). Engineering judgement has deemed that the surveillance frequency of 12 hours is acceptable to ensure operation is within the safety analysis bounds.

SR 3.2.1.16

This surveillance is required only when power is greater than ISAL. The surveillance ensures that every cluster with a FULL ROD DEMAND POSITION of greater than 350 vu has a partial rod inserted, which ensures that operating conditions will be bounded by the conditions assumed in the safety analysis. Engineering judgement has deemed that the surveillance frequency of 12 hours is acceptable to ensure operation is within the safety analysis bounds.

SR 3.2.1.17

This surveillance is required only when power is above ISAL. The verification of reactor power level ensures that the reactor power level after exceeding ISAL does not then decrease below ISAL. The control rods for power ascension (the process of taking the reactor from a subcritical state, to critical then proceeding to ISAL for each REACTOR STARTUP) are procedurally controlled to be removed by their SEQUENCE NUMBER, the CONTROLLING ROD(S) in any other gang remain within 100 vu of any other gang CONTROLLING ROD or the RSF is greater than or equal to the INITIAL RSF to ensure operation is bounded by the safety analysis until ISAL is reached. Operation above ISAL is bounded by the safety analysis flux shape limits. However, after power level exceeds ISAL, power operation can not decrease below ISAL since possible control rod configurations for the power level are unbounded by the safety analysis. REACTOR OPERATION after the power level decreases below ISAL is prohibited. Engineering judgement has deemed that the surveillance frequency of 12 hours is acceptable to ensure operation is within the safety analysis bounds.

INFORMATION ONLY

BASES (continued)

REFERENCES

1. RTM-3580, "Thermal Hydraulic Limits and Flow Design for Subcycle (reactor subcycle dependent)". (U)
 2. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 4.3, "Nuclear Design." (U)
 3. RTM-4533, Revision 1, "Mark 22 Lattice Control Rod Replacement Criteria," M. L. Nadeau, August 7, 1987. (U)
 4. RTR-2633, "Sensitivity of Reactor Power Limits to Operability of the Axial Power Shape Monitors," D. C. Losey, R. H. Lowery, March, 1989. (U)
 5. DPST-88-865, "Estimates of the Safety Rod Timing, Worth and Post-Power Peaking in Mark 22 Charges", P. B. Parks, W. E. Graves, N. H. Kuehn, January, 1989. (U)
 6. DPST-88-865, Add. 1, "Estimates of the Safety Rod Timing, Worth and Post-Power Peaking in Mark 22 Charges, Addendum 1," P. B. Parks, N. H. Kuehn, S. D. Painter, and W. M. Massey, March, 1989. (U)
 7. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 15.1.3.1, "Safety System Configuration." (U)
 8. SRT-APG-920033, Effect of Integral Rods on Control Rod Analyses in Chapter 15 of the SAR, E. F. Trumble April 1992.
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REVISION ONLY

B 3.2 REACTOR OPERATING CONDITIONS

B 3.2.2 Partial Length Control Rod Positioning (Axial Flux Control)

BASES

BACKGROUND

Restrictions on the positioning of partial length control rods are required to ensure that the uncertainty in monitoring, and thus in controlling, assembly axial power distributions is within the safety analysis assumptions. Restrictions are also required to ensure that the transient and final worth of safety rods, post-scrum power peaking, and the ability of the control rods to maintain shutdown are within safety analysis assumptions.

Requirements for axial power distribution monitoring

Restrictions on assembly axial power shapes (APSs) are required to ensure adherence of the core and assembly APS to safety analysis assumptions for transient analyses including Loss of Coolant Accidents (LOCA), Loss of Pumping Accidents (LOPA), reactor burnout risk (BOR), single control rod and gang rod withdrawal transients, and scram effectiveness.

Limits for the measured normalized APS have been defined, which ensure that the existing core and assembly power shapes are within those considered by the safety analysis assumptions.

Axial power distribution is measured by nine axial power monitor (APM) rods distributed throughout the core, inside the reactor tank. Each APM rod contains seven self-calibrating gamma thermometer sensors in a vertical array. Continuous readout from these nine APM's, through recorders and to the control computers, provide measurements of APS which are applied to all fuel assemblies in the reactor. However, the measurements represent only the conditions at the APM rod locations. As the distance between

(continued)

BASES (continued)

INTEGRATION ONLY

BACKGROUND

Requirements for axial power distribution monitoring
(continued)

an assembly and the nearest APM increases, the uncertainty in monitoring the assembly power shape increases. Maximizing symmetry and uniformity of power shape across the reactor tends to minimize the uncertainty. Although some non-uniformity and asymmetry are unavoidable, measures are taken to maximize the uniformity and symmetry of axial power distributions:

- a) The axial power distributions are restricted to a defined range,
- b) The charge design assures a 120-degree symmetry in the loading of partial rod complements,
- c) REACTOR OPERATION with safety rods inserted is restricted (Technical Specification 3.1.1, "Safety Rod Withdrawal Limit"),
- d) REACTOR OPERATION with control rods with different CONTROL ROD SEQUENCE within either gang 1 or gang 2 is restricted (Technical Specification 3.2.1),
- e) REACTOR OPERATION with CONTROLLING RODs with non-uniform insertion within any gang is restricted (Technical Specification 3.2.1),
- f) REACTOR OPERATION with non-symmetric partial rod complements is restricted (Technical Specification 3.2.2),
- g) REACTOR OPERATION with nonuniform insertion of inserted partial rods within any gang is restricted (Technical Specification 3.2.2),

Requirement to ensure long-term shutdown

The prohibition of the use of cadmium control rods as INTEGRAL CONTROL RODs ensures consistency of operation with analysis assumptions about worth of cadmium rods to help ensure long term reactor shutdown. Cadmium control rods are assumed to be available to insert after safety rod scram for shutdown. Cadmium rods inserted in sequence do not violate this requirement.

(continued)

BASES (continued)

INFORMATIONAL BASIS

BACKGROUND
(continued)

Requirement to ensure long-term shutdown
(continued)

The use of up to six full-length Lithium control rods as INTEGRAL CONTROL RODs is permitted provided: 1) Only strong control rods are used as INTEGRAL CONTROL RODs, 2) No more than two INTEGRAL CONTROL RODs are inserted in any one REACTOR SYSTEM, 3) No INTEGRAL CONTROL RODs are inserted in gang 1, 4) All INTEGRAL CONTROL RODs are FULLY INSERTED, and 5) No strong partial rods are used without a weak partial in the same cluster. The results of analyses performed show that use of INTEGRAL CONTROL RODs with the above restrictions will not cause an increase in reactivity worth for any accident assumed in the Safety Analysis Report (SAR).

APPLICABLE
SAFETY ANALYSES

Restrictions on the positioning of partial length control rods are required under defined full power conditions to ensure that the uncertainty in monitoring, and thus in controlling, assembly axial power distributions is within the safety analysis assumptions. Restrictions are also required at lower power and during startup to ensure that the axial power shapes, transient and final worth of safety rods, post-scam power peaking, and the ability of the control rods to maintain shutdown are within the safety analysis assumptions.

LOCA analyses are discussed in Chapter 15 of Reference 2. BOR safety limits, applicable during steady state operation, are discussed in Reference 2. BOR analyses are performed for "Transient-Protection Limit" (TPL) events, which are discussed in Chapter 15 of Reference 2. For all of those events, the analyses of Reference 2 assume that the reactor is shutdown by a scram. Control rod and gang withdrawal events are included in the TPL events in Chapter 15 of Reference 2. Short-term shutdown following scram, and long-term shutdown following insertion of control rods, are inherent assumptions in the safety analysis.

This Technical Specification requires rod movement to correct control rod mispositioning when it occurs. The safety analysis bounds all cases of such corrective rod motion, including the effects of power maldistribution due to the rod mispositioning, when the corrective action is initiated under otherwise normal operating conditions. Control rod misalignments are discussed in SAR Chapter 15.1.3.4.

(continued)

INFORMATION ONLY

BASES (continued)

LCOs

Adherence to the requirements for partial control rod positioning limits operation to conditions which have been analyzed. This ensures that the core and assembly axial power shape measurement uncertainty is within the assumptions of the safety analyses for flow instability during LOCA and LOPA, for reactor BOR, and for all TPL events.

APPLICABILITY

Adherence to LCO 3.2.2 is required during REACTOR OPERATION.

The safety analyses on which these specifications are based (Ref. 3 and 4) were performed for Mark 22 charges. Thus LCO 3.2.2 is applicable only to Mark 22 charges.

(continued)

BASES (continued)

ACTIONS

General Bases

Gross violation of the partial rod positioning requirements could produce power distributions which are outside the safety analysis assumptions, and at the same time introduce unaccounted-for uncertainty into the measurement of those assembly power distributions.

Restoring the control rods to within the limits within the required Completion Times will limit the time of operation with a power distribution, or control rod available for shutdown, which may be outside the bounds assumed in the safety analysis.

Completion Times

The defined completion times of 8 hours for rod motions are based on operating experience and permit carefully planned and deliberate action to avoid erroneous or unnecessary further perturbation of the power distribution.

If the correction cannot be made within the 8 hour time frame, REACTOR SHUTDOWN is required in 90 minutes.

Multiple Conditions

A condition requiring an Action corresponds to violation of one of the requirements of the LCO.

If the Required Actions are not completed within the required Completion Times, then placing the reactor in REACTOR SHUTDOWN within an additional 90 minutes will limit the time of operation with a power distribution or available shutdown reactivity which may be outside the bounds assumed in the safety analysis. Operation without the ability to properly manage control rods is unanalyzed.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.2.2.1

Visually verify that every partial length control rod is latched to the cable connected to a motor. This inspection will be done within 4 hours after latching the actuator and prior to REACTOR STARTUP.

The visual check is done to confirm the control rod will move when the motor is energized. The time of 4 hours was selected based on operator experience.

SR 3.2.2.2, SR 3.2.2.3, SR 3.2.2.4, SR 3.2.2.5,
SR 3.2.2.6, SR 3.2.1.7

Verifying that each control rod positioning requirement is met, by surveillance of all rods controlled by that requirement at an appropriate frequency, limits the time of operation with the axial power shape measurement uncertainty or available shutdown reactivity beyond the safety analysis assumptions, should the requirement be violated for any reason. The appropriate frequency is determined based on experience of the probability of an error or malfunction, considering the other surveillances triggered by specific actions (as discussed in the foregoing sections).

These surveillances are normally performed using information available in the control room.

SR 3.2.2.5, SR 3.2.2.6

Motion of individual CONTROLLING RODs in gangs 1 and 2 is not a normal procedure. It would occur only in the unlikely case of operator error or through equipment malfunction. Most equipment malfunctions would be indicated by the rod fault system. Motion of individual partial rods or pairs of rods takes place every shift when partial rods are set, i.e. trim is adjusted. A surveillance frequency of 12 hours ensures that rods are positioned correctly and will maintain the fuel assemblies within the DBA limits set by the safety analysis.

A frequency of 12 hours is adequate because control rod motion is a slow, deliberate process. By procedure, individual control rod positions are always verified following motion and the control rod fault system (if OPERABLE) will alarm for any control rod which is not at its CONTROL ROD DEMAND POSITION.

(continued)

BASES (continued)

INFORMATION ONLY

**SURVEILLANCE
REQUIREMENTS
(continued)**

SR 3.2.2.6. SR 3.2.2.7

Verifying that the partial rod complements are symmetric, i.e. that the strength (none, weak, or double) of partial rods inserted in all three of each symmetric trio of clusters is the same, and that no gang contains both a VOID PARTIAL ROD COMPLEMENT and a DOUBLE PARTIAL ROD, every 7 days, limits the time of operation with erroneously selected asymmetric partial rods or too wide a range of complements in a gang, which would be outside the safety analysis assumptions.

A frequency of 7 days is adequate as re-verification of the partial rod complement because the partial rod complement is verified within one hour of any individual partial rod insertion or withdrawal (SR 3.2.2.3). Verification every 7 days thereafter confirms that the partial rod complements have not changed.

No consideration has been given to a strong partial rod being in the reactor without a weak partial rod in the same cluster in the axial power measurement uncertainty analyses. Therefore Table 3.2.2-1 does not permit operation with a partial rod complement of only a strong partial rod in a cluster.

SR 3.2.2.8

Performing a calibration check of the APMs ensures that the reactor will be within the safety analysis APR limits (with uncertainties taken into account) as set in Table 3.2.2-3. A surveillance frequency of 1 month provides adequate verification of the low failure rate of those systems. APMs failing a calibration check have occurred less than one per reactor year.

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BASES (continued)

INFORMATION ONLY

**SURVEILLANCE
REQUIREMENTS
(continued)**

SR 3.2.2.9

This surveillance is required only when power is above ISAL. The verification of reactor power level ensures that the reactor power level after exceeding ISAL does not then decrease below ISAL. The control rods for power ascension (the process of taking the reactor from a subcritical state, to critical then proceeding to ISAL for each REACTOR STARTUP) are procedurally controlled to be removed by their SEQUENCE NUMBER, the CONTROLLING ROD(S) in any other gang remain within 100 vu of any other gang CONTROLLING ROD or the RSF is greater than or equal to the INITIAL RSF to ensure operation is bounded by the safety analysis until ISAL is reached. Operation above ISAL is bounded by the safety analysis flux shape limits. However, after power level exceeds ISAL, power operation can not decrease below ISAL since possible control rod configurations for the power level are unbounded by the safety analysis. REACTOR OPERATION after the power level decreases below ISAL is prohibited. Engineering judgement has deemed that the surveillance frequency of 12 hours is acceptable to ensure operation is within the safety analysis bounds.

SR 3.2.2.10

This surveillance ensures that no cluster contains a STRONG PARTIAL ROD COMPLEMENT. Analysis supporting the use of INTEGRAL CONTROL RODs assume that no strong partial rods are used without a weak partial in the same cluster. This restriction would ensure that there is no increase in the reactivity values for any of the safety analysis accidents due to the presence of strong integral rods.

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INFORMATION ONLY

BASES (continued)

SURVEILLANCE
REQUIREMENTS
(continued)

Further remarks on surveillance

Control rod position tolerance: The control rod fault system for each rod will normally provide audible and visual indication when the difference between actual rod position and CONTROL ROD DEMAND POSITION exceeds ± 4 vu. The difference between control rod insertions within a gang or a gang 3 cluster group, including any known deviation from indicated position, must be within the limit of 40 vu.

REFERENCES

1. RTM-3580, "Thermal Hydraulic Limits and Flow Design for Subcycle (reactor subcycle dependent)". (U)
 2. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 4.3, "Nuclear Design." (U)
 3. RTM-4533, Revision 1, "Mark 22 Lattice Control Rod Replacement Criteria," M. L. Nadeau, August 7, 1987. (U)
 4. RTR-2633, "Sensitivity of Reactor Power Limits to Operability of the Axial Power Shape Monitors," D. C. Losey, R. H. Lowery, March, 1989. (U)
 5. DPST-88-865, "Estimates of the Safety Rod Timing, Worth and Post-Power Peaking in Mark 22 Charges", P. B. Parks, W. E. Graves, N. H. Kuehn, January, 1989. (U)
 6. DPST-88-865, Add. 1, "Estimates of the Safety Rod Timing, Worth and Post-Power Peaking in Mark 22 Charges, Addendum 1," P. B. Parks, N. H. Kuehn, S. D. Painter, and W. M. Massey, March, 1989. (U)
 7. SRT-APG-920033, Effect of Integral Rods on Control Rod Analyses in Chapter 15 of the SAR, E. F. Trumble, April 1992.
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B 3.0 EQUIPMENT OPERABILITY

INFORMATIONAL GUIDE

B 3.3.1 Scram Circuits Setpoints and Response Times

BASES

The ability to terminate undesired transients prior to the occurrence of significant damage is a function of several variables (Reference 1) including:

- a) Pre-transient operating point (limiting values assumed in the Safety Analysis or specified in the COR).
- b) The margin between normal operating conditions and conditions that could result in significant fuel, target, or reactor damage (calculated by the methods specified in Reference 2).
- c) The nature of the undesired transient (calculated for transients specified in Reference 2).
- d) The magnitude of the departure from normal operation necessary to actuate scram circuits (specified by this LCO).
- e) Number and type of circuits OPERABLE.
- f) Scram circuit response time (specified by this LCO, Reference 3).
- g) Safety rod insertion time (specified by LCO 3.1.2, Reference 3).
- h) Safety rod worth (specified by LCO 3.1.1, Reference 3).

All of these variables share a degree of equivalence in that adjustment of one can be compensated for by appropriate values of the others so as to provide an equivalent degree of protection.

Ideally, variables d, f, and g should be minimized and variables e and h should be maximized so as to maximize variable a at a given degree of protection against undesired transients. In general, specifications on these

(continued)

INFORMATION ONLY

BASES

(continued)

variables are at least partially based on system capabilities so as to minimize the degree of departure from the optimum combination.

The bases of the safety circuit set points and response times listed in the Technical Specification are:

- o Consistency with safety analyses and derivations of TRANSIENT LIMITS (Ref. 2).
- o Scram circuit functional capabilities (Refs. 4 and 5).

Set points, bypass requirements, and response times must be consistent with safety and limits analyses to provide protection against the incidents specifically analyzed. In particular, certain scram circuits are designed to protect against damage from certain general classes of incidents shown in Table B 3.3.1-1. In these cases, providing the required protection is a necessary (although not necessarily sufficient) performance requirement for the particular scram circuit. In other cases (e.g., the seismometer scram circuit), the protection provided is considered necessary; however, the required set points and response times cannot be numerically defined. In such cases, the performance requirements are based on system capability.

The conditions for response time measurement listed in the Requirements are based on existing test equipment and procedures where applicable (Ref. 6). The Requirements specifically permit the substitution of improved tests and measurements as they are developed.

Bypass restrictions are designed to maintain uninterrupted scram capability for all applicable incidents. Ideally, each scram circuit is OPERABLE continuously, when it can contribute to the desired protection, and an effort is made to approach this goal insofar as practicable. However, maintenance, testing, adjustments in reactor power, or adjustment in setpoints requires that certain circuits be bypassed occasionally for brief intervals. In actual practice, scram circuit protection is normally far in excess of the requirements of this LCO.

(continued)

BASES

INFORMATION ONLY

(continued)

Stringent procedures are applied whenever the following circuits must be taken offline (Reference 7):

SCRAM CIRCUIT

1. High Level Flux Monitor (HLFM)
2. Period Monitor
3. Heat Exchanger Cooling Water Flow
4. Cooling Water Header Flow
5. Plenum Pressure
6. Seismometer
7. Blanket Gas pressure
8. Septifoil Supply Header pressure very low
9. Control system AC power failure
10. Process Water level

Typically, more than one circuit is capable of fulfilling the function of the primary circuits (e.g., loss of AC power to the process pumps could be sensed by two process water plenum pressure circuits as well as the safety computers).

Because the effectiveness of several of the scram circuits is dependent upon set points and charge design, designation of primary and secondary circuits is documented in the safety analysis report rather than in this LCO.

(continued)

Scram Circuits Setpoints and Response Times
B 3.3.1

BASES (continued)

INFORMATION ONLY
TABLE B 3.3.1-1

Bases for Scram Circuit Set Points and Response Time

<u>Scram Circuit</u>	<u>Bases (Note a)</u>
1. High Level Flux Monitor (HLFM)	General power surge
2. Period Monitor	Startup "transient"
3. Process Water Plenum Pressure Monitor	Sudden reduction of flow in one or more process water pumps
4. Blanket Gas Pressure	Loss of blanket gas pressure
5. Cooling Water Supply Header Flow	Loss of flow in one cooling water header
6. Individual Heat Exchanger Cooling Water Flow	Loss of flow in one cooling water heat exchanger
7. Control Rod Coolant Supply	Loss of control rod coolant flow
8. Moderator Level	Process water leak
9. Process Water Pump AC motor undervoltage	Loss of AC motor power
10. Control System Power Monitor	System capability
11. Seismic Switch	System capability

NOTES

- a. In most cases, the response times listed in the requirement are based on system capability.

(continued)

INFORMATION ONLY

BASES (continued)

REFERENCES

1. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 7.2.1.2.3.1.1, "Scram Circuit Setpoints and Response Times." (U)
 2. DPSTM-110, Technical Manual - Effluent Temperature and BOSF_N Limits for Fuel and Target Assemblies and Reactor Effluent Temperature Limits. (U)
 3. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 15.1.3.4, "Reactor Physics Considerations." (U)
 4. DPST-70-497, Analysis of the Savannah River Reactor Protective Instrument System. (UCNI)
 5. DPSTSY-100-6. E. Nomm. System Analysis of Process Control and Safety Computers, December 1975. (U)
 6. DPSOP-122, Reactor Instrumentation. (UCNI)
 7. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 7.2.1.2.3.1.2, "Scram Circuit Bypass Log." (U)
 8. WSRC-TR-91-0167
-

B 3.3 INSTRUMENTATION

B 3.3.2 Scram Circuit - BYPASS

BASES

INFORMATION ONLY

Bypass restrictions are designed to maintain uninterrupted scram capability for all applicable incidents. Ideally, each scram circuit is OPERABLE continuously when it can contribute to the desired protection, and an effort is made to approach this goal insofar as practicable. However, maintenance, testing, adjustments in reactor power, or adjustment in setpoints requires that certain circuits be bypassed occasionally for brief intervals. In actual practice, scram circuit protection is normally far in excess of the requirements of this LCO. Stringent procedures are applied whenever the following circuits must be taken offline (Reference 1):

SCRAM CIRCUIT

1. High Level Flux Monitor (HLFM)
2. 10-second period
3. Heat exchanger cooling water flow
4. Cooling water header flow
5. Plenum pressure
6. Process water pump AC power
7. Seismometer

(continued)

INTENTIONAL ONLY

BASES

(continued)

SCRAM CIRCUIT

8. Blanket gas pressure
9. Septifoil supply
header pressure very low
10. Control system AC power
failure
11. Process water level

Typically, more than one circuit is capable of fulfilling the function of the primary circuits (e.g., loss of AC power to the process pumps could be sensed by two process water plenum pressure circuits as well as the safety computers).

Because the effectiveness of several of the scram circuits is dependent upon set points and charge design, designation of primary and secondary circuits is documented in the safety analysis report rather than in this LCO.

REFERENCES

1. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 7.2.1.2.3.1.2, "Scram Circuit Bypass Log." (U)
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DISSEMINATION ONLY

B 3.3 INSTRUMENTATION

B 3.3.3 Scram Circuit - Neutron Flux Monitoring - Shutdown

BASES

To avoid inadvertent criticality during REACTOR SHUTDOWN the neutron flux will be monitored (Reference 1). The monitoring equipment shall include two High Level Flux Monitors (HLFMs) and two period monitors.

High Level Flux Monitoring System - The HLFM measures the neutron flux over a range of reactor power levels from about 0.0001 to 1.00 times full power. The high level flux monitors provide information for scram circuits to protect against excessive power. To provide redundancy, at least two operating HLFM's are required.

Log N - Period System - This system responds in proportion to the logarithm of the neutron flux level. It provides a signal over a very wide range of neutron flux levels with no scale changes on the instrument. The time derivative of the log N signal is equal to the reciprocal of the reactor period. To protect against rapid changes in neutron flux at levels below those where the high level flux monitor readings are on scale, a scram circuit derives its signal from the log N - period system. The scram set point must permit periods short enough to allow for normal increase in the flux during startup and changes in power at low power levels. However, the setpoint must protect against periods so short that manual or HLFM scram action is ineffectual in arresting the increase prior to core damage.

REFERENCES

1. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 15.4.4, "Reactivity Additions During Shutdown." (U)
-

B 3.3 INSTRUMENTATION

- B 3.3.4.1 Automatic Incident Action
B 3.3.4.2 Incident Control

INFORMATION ONLY

BASES

The monitoring and control systems necessary to the proper OPERABILITY of the Emergency Cooling and Activity Confinement systems shall be OPERABLE in both the reactor building and the remote control center. OPERABILITY of this equipment (with the exception of the automatic actuation apparatus known as AIA, Reference 1) is required whenever water is required in the reactor. The AIA shall be OPERABLE during REACTOR OPERATION, beginning with initiation of safety rod withdrawal and ending either with initiation of shutdown of AC pump motors or one hour after insertion of safety rods to effect nuclear shutdown. Limited exceptions are permitted for testing and maintenance as specified in applicable procedures.

To prevent an incident or minimize the consequences of an incident, the reactor operating personnel must have the facilities to:

1. Determine when emergency cooling is needed, and initiate it promptly.
2. Monitor key parameters and control the equipment necessary to maintain core cooling and minimize activity release from the reactor building, even if reactor area evacuation should be required.

Two systems assist the operator to prevent incidents, Automatic Incident Action (AIA) and Incident Control. Both systems are required to be OPERABLE.

Both automatic and procedural means are provided to determine the need for emergency cooling and to initiate it (Refs. 1 and 2). Procedural actuation is facilitated by provision of a single incident switch that actuates (or enables) all of the appropriate valves and pumps. Remote control and telemetry systems are provided to permit monitoring and control of vital reactor site processes even if the reactor building personnel must be evacuated.

(continued)

BASES

(continued)

OPERABILITY of emergency cooling and activity confinement systems is required under conditions with potential for melting of fuel and release of fission products. Automatic action is required under conditions that might require prompt reaction to a loss-of-coolant accident, i.e., both the large leak rates possible with AC motors driving the pumps and assembly power high enough to rapidly heat an assembly to the melting point. Decay heat after nuclear shutdown is sufficiently low that manual incident action is adequate to prevent fuel melting. The limited exceptions are permitted for maintenance and testing and prescribed in terms of lists of equipment that must be OPERABLE under various reactor operating conditions permitted by this specification.

A note is provided to the APPLICABILITY which states that the AIA may be inoperable for up to 1 hour at a time not to exceed a cumulative time of 24 hours per reactor-year which is the cumulative time that the reactor is critical. This exception to the APPLICABILITY will allow for the correcting of individual AIA problems without shutting down the reactor. The allowed outage time is based on a Probabilistic Risk Assessment (PRA) which indicates that the increase in the core melt frequency is extremely slight and acceptable.

REFERENCES

1. DPSPU-76-11-6, M-2 Containment Console. (U)
 2. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 7.3.2, "AIA Analysis." (U)
 3. DPSOL 105-1857-PKC, Containment System Operating Requirements and Testing Schedule. (UCNI)
 4. DPSOL 105-1186, Safety Circuit Bypass Rules. (UCNI)
-

"DELETED VERSION"

Remote Emergency Monitoring and Control System B 3.3.5

B 3.3 INSTRUMENTATION

B 3.3.5 Remote Emergency Monitoring and Control System

BASES

BACKGROUND

The Remote Emergency Monitoring and Control System (REMACS) provides the capabilities necessary to shut down the reactor from outside the K reactor central control room. The REMACS capabilities include remote monitoring and control of essential cooling, sump, electrical, and ventilation equipment for the reactor (Reference 1).

The system provides remote monitoring of significant parameters from the following systems:

- Process Water System
- Airborne Activity Confinement System
- Contaminated Water Removal and Disposal System
- Confinement Heat Removal System
- Emergency Cooling System
- Cooling Water System

The system provides remote control of equipment in the following systems:

- Automatic Incident Action System
- Process Water System
- Airborne Activity Confinement System
- Contaminated Water Removal and Disposal System
- Confinement Heat Removal System
- Emergency Cooling System
- River Water System
- Cooling Water System
- Containment Isolation System
- Control Room Heating, Ventilation and Air Conditioning

Remote monitoring and control of the K reactor via the REMACS is possible from the L reactor area or from the [REDACTED] building.

(continued)

"DELETED VERSION"

"DELETED VERSION"

Remote Emergency Monitoring and Control System
B 3.3.5

BASES

INFORMATION ONLY

**BACKGROUND
(continued)**

The K reactor contains two redundant slave computer systems (Slave 1 and Slave 2) and a master station. Input signals from field sensors are provided to both slave computers. The slave computers relay the information to the K, L, and [REDACTED] building master stations. The Emergency Operation Center (EOC) terminal is a monitor-only display of [REDACTED] data. A control link must be established between a master workstation and a slave computer. All input is issued over this controlling link, and from there to relays which control equipment operation. Power to the relays is provided from a 120 volt battery via the Power Security Switch. Communications between areas is accomplished using communication modems connected to telephone lines and Power Department supervisory control cables. Each slave computer system has an uninterruptible power supply that can operate the system for 20 minutes if power is lost.

In the event that the control room must be evacuated, the Permit button on the M-2 console is pressed. This action alerts the L control room and [REDACTED] building personnel to take control. The area responding first by pressing the Request For Control button on their master station keyboard will gain control of the K reactor. The L control room can request control and supersede the [REDACTED] building control. Once the L control room has obtained control, it can retain control even if another area requests control. The controlling area can grant control by pressing the Release Control button or deny control by pressing the Hold Control button. The originating area, K reactor, can always regain control by pressing a single Deny button (two of three Deny buttons are needed to override a failed PERMIT button).

(continued)

"DELETED VERSION"

BASES (continued)

**APPLICABLE
SAFETY ANALYSES**

The SRS Production Reactor SAR Chapter 15 Accident Analyses listed below assume the capability of operators to shut down the reactor or otherwise take action to mitigate the consequences of the accident. These capabilities (except actuate the process room spray system) are provided by the REMACS in the event the control room must be evacuated.

Loss of Cooling Water Supply to 186 Basin (15.2.2.4)
Operator action - shut down the reactor

Loss of Cooling Water Inventory With Resultant Loss of Process Water Pumping Power (Classical Loss of Pumping Accident) (15.2.2)
Operator action - ECS actuation

Inadvertent Actuation of the Emergency Cooling System (15.5.1)
Operator action - terminate ECS

Leaks and Failure of Small Lines (15.6.2)
Potential operator actions - shut down reactor, trip process water pump ac motors, align the PWS for once-through cooling, close 8 rotovalves

Loss of Coolant Accident (Heat Exchanger Tube Leak) (15.6.1)
Potential operator action - scram reactor

Loss of Coolant Accident (LOCA) (15.6.3)
Potential operator actions - initiate ECS flow to PWS loop 4, actuate the process room spray system

Radioactive Releases from a Subsystem or Component (fuel handling accident) (15.7)
Operator action - actuate the process room spray system

(continued)

INFORMATION ONLY

BASES (continued)

LCO

The LCO specifies that the REMACS shall be OPERABLE. The system design includes sufficient redundancy (e.g. two slave computers and two remote master stations) to accommodate a single failure and still perform its required function.

APPLICABILITY

The REMACS is required to be OPERABLE during REACTOR STARTUP, REACTOR OPERATION, and REACTOR SHUTDOWN whenever the reactor contains HGAs. This Applicability assures that the capability is available to remotely shut down the reactor or otherwise take action to mitigate the consequences of postulated accidents.

ACTIONS

A.1

With one of the slave computers inoperable Action A.1 requires that the affected slave be restored to OPERABLE status within 24 hours. The Completion Time of 24 hours is considered acceptable based on the ability of the remaining slave computer to provide the required remote monitoring and control capabilities and the low likelihood of both a control room evacuation and a failure of the remaining slave computer.

B.1 and B.2

With both slave computers inoperable Action B.1 requires that one slave computer be restored to OPERABLE status within 8 hours and both slaves within 24 hours. The Completion Times of 8 hours and 24 hours are considered acceptable based on the low likelihood of a control room evacuation during the restoration period.

(continued)

INFORMATION ONLY

BASES

ACTIONS (continued)

C.1, C.2, and C.3

If the Required Actions of Condition A or B are not met within the specified Completion Time, then the plant must be placed in the safest possible state. Therefore, Actions C.1 and C.2 require the reactor to be shut down and operations involving positive reactivity additions to be suspended. These actions place the plant outside of the Applicability requirements, except that the reactor may contain HGAs. The 90 minute Completion Time to be in REACTOR SHUTDOWN is reasonable based on operating experience and will not challenge plant systems. Required Action C.3 specifies that the Actions of Conditions A or B be continued until all Actions are completed. Although the possibility of an accident requiring the OPERABILITY of REMACS with the reactor shut down is remote, efforts to restore the slave computer(s) must be continued.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.3.5.1

This surveillance requires verification that the REMACS communication modems are OPERABLE each 12 hours. The 12 hour Frequency is considered necessary due to the importance of the communication modems to the function of the system.

SR 3.3.5.2

This surveillance requires verification that the Power Security Switch supplies power to the slave computer output relays. This surveillance must be performed each 12 hours. The 12 hour Frequency is considered necessary due to the importance of the Power Security Switch to the function of the system.

SR 3.3.5.3

This surveillance requires verification that each of the slave computers can perform the required monitoring and control functions. The OPERABILITY of signal input circuits (monitoring), computer control inputs from the master station, slave computer execution, and output circuits (control) is confirmed by this testing. The system complexity consisting of hardware, software and interfacing instrumentation justifies performance of this surveillance each 24 hours.

SR 3.3.5.4

This surveillance requires OPERABILITY verification of the system functions (Deny, Permit, Request, and Take Control) which control transfer of the controlling master station. This surveillance must be performed within 12 hours prior to REACTOR STARTUP and each 30 days. This Frequency provides assurance that the system control functions are OPERABLE prior to entering the most critical phases of operation (startup and operation) and each 30 days during all applicable operations. The 30 day Frequency is considered acceptable based on the demonstrated reliability of the system control functions of the REMACS.

(continued)

BASES (continued)

REFERENCES

1. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 7.4.1, "Shutdown From Outside the Control Room." (U)
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B 3.3 INSTRUMENTATION

B 3.3.6 Safety Computer

INFORMATION ONLY

BASES

The safety computer systems are designed to divide the temperature and the flow signals from the reactor assemblies between two independent computers. Each computer receives temperature signals from three of the six assemblies in every cluster in the reactor lattice and flow signals from the other three assemblies in the cluster. In the buckled zone region, groupings of six adjacent assemblies are defined to form the cluster for monitoring purposes. Thus, each computer system provides complete coverage of the reactor by monitoring either a temperature or a flow signal from each assembly. The intent of this LCO is to maintain this redundancy to the greatest extent practicable, therefore 2 safety computers must be OPERABLE.

Since the safety computers provide the primary scram protection against temperature increase incidents, reactor operation without minimum protection is not permitted and operation without redundant protection is limited. A single temperature input per cluster per safety computer provides the minimum protection required (termination of a cluster power increase within damage limits). Requiring both computers on-line at startup, and limiting time of operation with a single safety computer, are intended to implement the policy of maintaining redundant protection. The particular time limit of 24 hours was derived from estimates of the time required to diagnose and correct problems with an inoperable computer. Limiting the number of invalid temperature and flow inputs is intended to maintain a practical maximum number of valid inputs.

To ensure the assembly effluent temperature limits are within those described in the Core Operations Report (COR), assembly temperatures are monitored by the control computer using input from four thermocouples per assembly. The average assembly effluent temperature may also be monitored by an auxiliary recorder or a safety computer if the control computer is inoperable.

(continued)

INFORMATION ONLY

BASES (continued)

These thermocouples, each of which monitors one quadrant of a Mk22 assembly, are the primary means by which adherence to assembly effluent temperature limits are verified. Individual thermocouple signals are read by the control computer and a hard wired assembly average of the thermocouple signals is read by one or the other safety computer. There are various ways for temperature monitoring to become degraded during operation due to several different failure modes affecting different parts of the circuit train. The primary causes of degradation are computer hardware failures, open thermocouples, short circuits and junction faults.

To meet the requirements of the Technical Specifications, a distinction between inoperable thermocouples and invalid safety computer inputs is required. The distinction is as follows:

Inoperable Thermocouple: any malfunction in the circuit path between, and including, the monitor pin thermocouple and the solder panel.

Invalid Safety Computer Input: an assembly signal input to the safety computer is invalid if the signal has been been aborted or declared invalid by the safety computer.

During reactor operation the control computer can scan for inoperable thermocouples in two ways:

1. A program executed automatically by the control computer compares the average assembly effluent temperatures read by the safety computers to the temperatures calculated by the control computer, approximately once per hour per assembly. If the two values are significantly different, an alarm message is issued. The temperature comparison detects the most common type of failure, which is a computer hardware failure. In addition, it can also detect certain types of open thermocouple failures.
2. Another program that is required to be executed once per month compares each of the four quadrant temperature signals from an assembly with the others. Since Mk22 are quadrant monitored (as opposed to annulus monitored), each thermocouple in a given assembly should have approximately the same value with

(continued)

INFORMATION ONLY

BASES (continued)

minor differences arising from flux tilt, thermocouple uncertainties, etc. However, if one of the thermocouples is inoperable, it may register (via a control computer printout) a significantly different value than the other three thermocouple readings in that assembly.

Since certain types of open thermocouple failures cannot be detected during operation, a number of undetected inoperable thermocouples must be assumed to exist in the core during operation. The safety analysis bounds all cases of two inoperable thermocouples in one assembly, but due to the chance of also having an undetected thermocouple failure in that assembly, the Technical Specifications require the restoration of at least three thermocouples in each assembly to OPERABLE status in 12 hours. The time limit is conservative with respect to the PRA analysis and sufficient in length for operator completion. If in that time period, three thermocouples are not restored to OPERABLE, the reactor must shutdown in 90 minutes.

If three or more thermocouples in one assembly are found to be inoperable, the reactor must be shutdown in 90 minutes.

The current safety analysis supports 40 inoperable thermocouples in all assemblies requiring forced cooling. The Technical Specifications set an administrative limit of 30 inoperable thermocouples and a surveillance is performed at least once a month during REACTOR STARTUP and REACTOR OPERATION to ensure that there are less than or equal to 30 inoperable thermocouples. This ensures that the maximum number of detected inoperable thermocouples plus the maximum number of undetected inoperable thermocouples will not exceed the safety analysis limit of 40. The surveillance that identifies those thermocouple failures that are not detectable during operation is required prior to REACTOR STARTUP.

If 31 or more thermocouples are inoperable, 12 hours is allowed to restore at least 1698 monitor pin thermocouples to OPERABLE status. Twelve hours is allowed to verify the indication of an inoperable thermocouple. When verifying thermocouple OPERABILITY, certain failures (e.g. control computer, cross bar, capacitor) can provide indication of numerous inoperable thermocouples when in fact the thermocouples are not inoperable and continue to provide valid signal to the safety computer. The 12 hours permits a reasonable time to evaluate the indication, repair or

(continued)

BASES (continued)

INTENTION ONLY

replace equipment, or complete the surveillance with alternate equipment or by an alternate method. If the OPERABILITY of the thermocouples can not be verified by the end of 12 hours or if it is determined that 31 or more thermocouples are inoperable, the reactor is required to be shutdown in 90 minutes.

Bypassing a safety computer that does not meet prescribed requirements may be done procedurally or automatically.

The safety computer also verifies that the reactor has shutdown and then trips the 190 pumps.

REFERENCES

1. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 7.2.1.1.2, "Safety Computer System." (U)
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B 3.3 INSTRUMENTATION INFORMATION ONLY

B 3.3.7.1 Neutron Monitoring Requirements - Shutdown (No C&D)

B 3.3.7.2 Neutron Monitoring Requirements - C&D

BASES

BACKGROUND

The function of the Fission Counter (FC) System is to alert the operators of unexpected flux increases. These detectors include internal and external fission counters that detect neutrons leaking from the core, or within the core.

The installed fission counters are proportional fission ion chambers operated in the pulse mode. The chamber responds to radiation sources. An electronic discriminator conditions the output so only the fission events (caused by neutrons within the chambers) are detected. The observed count rate depends on the chamber position, neutron source strength, amount of fuel in the reactor, and the fuel exposure. The range of the chamber and electronics is 0 to 9,999,999 counts per minute (CPM).

During normal operation the chambers are moved to the full out position and the shutter is closed. Since the chambers contains fuel, this is done to prevent burning out the chambers (Ref. 1).

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the FC system is required to provide a signal to alert the operator to changes in core reactivity such as an improperly loaded fuel assembly. The operator could then initiate emergency action (activate the SSS).

LCO

During C&D the OPERABILITY of fission counters ensure that redundant monitoring capability is available to detect changes in core reactivity.

During shutdown one neutron monitor ensures the capability is available to detect changes in core reactivity.

(continued)

PRODUCTION ONLY

BASES (continued)

APPLICABILITY

To detect changes in core reactivity, LCO 3.3.7.1 requires one fission counter OPERABLE during REACTOR SHUTDOWN with no C&D in progress and moderator in the reactor.

Core reactivity changes are detected, in LCO 3.3.7.2, by two required fission counters OPERABLE during CHARGE AND DISCHARGE.

ACTIONS

A. (LCO 3.3.7.1)

When no fission counters are OPERABLE during REACTOR SHUTDOWN the function to detect reactivity changes has been lost. With that condition, positive reactivity additions are IMMEDIATELY suspended and actions are IMMEDIATELY initiated to restore a fission counter to OPERABLE status.

A. (LCO 3.3.7.2)

When less than two fission counters are OPERABLE during CHARGE AND DISCHARGE it is possible that either redundancy or function has been lost. With that condition, CHARGE AND DISCHARGE OPERATIONS should IMMEDIATELY be suspended.

**SURVEILLANCE
REQUIREMENTS**

SR 3.3.7.1.1 and SR 3.3.7.2.1

Performing a CHANNEL CHECK provides assurance that the channels have not drifted outside their limits. A CHANNEL CHECK is the comparison of the indicated parameter values for each of the functions. It is based on the assumption that the two channels of indication should be reading approximately the same. Agreement is based on a combination of the channel instrument uncertainties, including control isolation, indication, and readability. If a channel is outside of the match criteria, it may be an indication that the transmitter or the electronics have drifted outside of their limits. The frequency of 12 hours is based on engineering judgment, and is consistent with improved commercial industry Technical Specifications.

(continued)

INFORMATION ONLY

Neutron Monitoring Requirements - Shutdown (No C&D) B 3.3.7.1
Neutron Monitoring Requirements - C&D B 3.3.7.2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.7.1.2 and SR 3.3.7.2.2

The performance of a discriminator check provides assurance that the monitoring system is responding to neutron flux. The chamber discrimination circuit is set so that primarily fission events (caused in the chamber by neutron absorption in ^{235}U) are detected. The frequencies are based on engineering judgment, and are consistent with improved commercial industry Technical Specifications.

SR 3.3.7.1.3 and SR 3.3.7.2.3

The performance of an CHANNEL CALIBRATION provides assurance that the analog process control equipment and alarms are OPERABLE. This test is a periodic check of the monitoring equipment. When the channel is placed in the test condition, the input from the transmitter is removed and the trip output is isolated. This allows a test signal to be introduced into the instrument loop. The input can be measured, thus noting the accuracy of the signal conditioning of the process control modules upstream. The alarm OPERABILITY of the channel can be determined by varying the input and observing the alarm status. The frequency of 30 days is based on engineering judgment, and ensures systematic verification of the channels.

REFERENCES

1. Reactor Operator and Supervisor Study Guide, Instruments and Controls.
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INFORMATION ONLY

Process Water System Leakage Detection Instrumentation B 3.4.1

B 3.4 PROCESS WATER SYSTEM

B 3.4.1 Process Water System Leakage Detection Instrumentation

BASES

BACKGROUND

The Process Water System Leakage Detection Instrumentation (Airborne Tritium Monitoring System) has the capability to detect Process Water System Pressure Boundary (PWSPB) degradation shortly after occurrence to alert the operators, so that appropriate action can be taken before the degradation would result in a gross failure of the pressure boundary.

A limited amount of leakage is expected from the Process Water System and from auxiliary systems within the reactor building. Small amounts of leakage from valve stem packing and pump seals and other equipment that cannot practically be made 100% leaktight will occur. These leakages are considered IDENTIFIED LEAKAGE, and are collected in tanks so that flow rates can be established and monitored during plant operation. Normally, the valve stem packing and pump seal leakage does not enter the ventilation air stream and, therefore cannot be detected by the Tritium Monitoring System Instrumentation.

The process water contains tritium, which when released to the reactor building atmosphere can be detected by the Airborne Tritium Monitoring Instrumentation. The tritium in the process water is produced by neutron irradiation of deuterium in the process water. A measurement of the tritium concentration in the ventilation exhaust air from the reactor building is used to calculate the Process Water System leakage rate. Two types of instruments used to measure tritium-in-air are the Stack Tritium Monitor (STM) and the Berthold Tritium Monitor (BTM). The STM has a sensitivity of approximately 2 microcuries of tritium per cubic meter of air, and the BTM has a sensitivity of approximately 0.1 microcuries of tritium per cubic meter of air (corresponding to 0.6 counts per second) based on a sixty second measuring period (Refs. 1 and 3). The sensitivity of these instruments is sufficient to provide a rapid response to

(continued)

INFORMATION ONLY

Process Water System Leakage Detection Instrumentation B 3.4.1

BASES

BACKGROUND (continued)

leaks from the PWSPB. The sensitivities of these instruments are theoretical under idealized conditions. Uncertainties for these instruments, as well as for the air flow measurement and process moderator sampling are accounted for in the limit for UNIDENTIFIED LEAKAGE from the Process Water System (Specification 3.4.2).

The Airborne Activity Confinement System maintains a negative pressure in the process areas. The Far Side Process Room, Near Side Process Room, Purification, and -40' level ventilation exhaust air streams all pass through the 105 Building stack, which is monitored at the +148' level. Therefore, the 105 Building stack serves as a common monitoring point which would indicate leakage in any of the 105 Building areas.

The tritium monitoring instruments are installed with sampling manifolds that connect to the various 105 Building process areas:

- 40' level
- Near Side Process Room
- Far Side Process Room
- Combined Far Side Process Room and Purification Stack (+148' level)

Any monitor can be aligned to monitor ventilation air from any of the 105 Building process areas mentioned above. There is one dedicated BTM (BTM C) that monitors only the stack (+148' level). The STMs and BTMs indicate instantaneous and cumulative process water leakage rates.

APPLICABLE SAFETY ANALYSES

The safety significance of PWSPB leakage can vary widely depending on the source of the leakage, leakage rate, and change in the leak rate. Therefore, the detection and monitoring of process water leakage into the 105 Building is necessary. Separating the identified sources of leakage from unidentified sources is necessary to provide prompt and quantitative information to the operators to allow them to take immediate action should a leak occur that is detrimental to the safety of the plant.

(continued)

INFORMATION ONLY

Process Water System Leakage Detection Instrumentation. B 3.4.1

BASES (continued)

LCO

Rapidly detecting small process water leaks with instrumentation provides the capability for a controlled reactor shutdown before a gross failure of the PWSPB. The requirement to have three Tritium Monitoring Systems OPERABLE provides a high degree of confidence that extremely small leaks would be detected in time to allow appropriate actions to be taken prior to a potential gross failure of the PWSPB. Two of the Tritium Monitoring Systems are required to be monitoring the +148' level of the 105 Building stack because the stack serves as a common monitoring point which would indicate leakage into any of the 105 Building process areas mentioned previously. The third Tritium Monitoring System can be aligned to monitor any of the 105 Building process areas and provides the capability to determine which process area is the source of a leak detected by the stack monitors.

APPLICABILITY

The Tritium Monitoring Instrumentation is required to be OPERABLE at all times so that instrumentation will be available to detect any inadvertent process water system leakage.

ACTIONS

A.1

With one or more Tritium Monitoring Systems inoperable during REACTOR STARTUP or REACTOR OPERATION, action must be taken to monitor the +148 level ventilation exhaust air stream with two OPERABLE Tritium Monitoring Systems in 8 hours. One of the monitors must be a BTM. A Completion Time of 8 hours is provided because the +148' level of the 105 Building stack serves as the only common monitoring point which would indicate leakage into any of the process areas serviced by any of the ventilation exhaust streams. The Completion Time of 8 hours takes into account the time it would take to align an OPERABLE Tritium Monitoring System to monitor the exhaust stream at the +148' level of the 105 Building stack. An analysis has indicated that the crack growth rate during an 8 hour period is negligible (Refs. 2 and 3).

(continued)

INFORMATION ONLY

Process Water System Leakage Detection Instrumentation B 3.4.1

BASES

ACTIONS (continued)

B.1 and B.2

During REACTOR STARTUP or REACTOR OPERATION, if the third Tritium Monitoring System that is monitoring any of the process areas serviced by ventilation exhaust air streams in the 105 Building is inoperable, action B.1 requires the two Tritium Monitoring Systems which are monitoring the exhaust stream at the +148' level of the 105 Building stack to be verified OPERABLE within 8 hours. One of the monitors shall be a BTM. A Completion Time of 8 hours is provided because the +148' level of the 105 Building stack serves as the only common monitoring point which would indicate leakage into any of the process areas serviced by any of the ventilation exhaust streams. An analysis has indicated that the crack growth rate during an 8 hour period is negligible (Refs. 2 and 3).

Action B.2 requires monitoring of any 105 Building process areas ventilation exhaust air streams with an OPERABLE Tritium Monitoring System within 7 days. The 7 day Completion Time is acceptable based on the OPERABILITY of the building stack monitor, which will detect leakage in any of the 105 Building areas. The inability to identify the process area source of a leak detected by the building stack monitor is acceptable for a 7 day period due to the very low likelihood of a significant leak during the period.

C.1 and C.2

During REACTOR STARTUP or REACTOR OPERATION, if all three Tritium Monitoring Systems are inoperable, action C.1 requires monitoring of the exhaust stream at the +148' level of the 105 Building stack with two OPERABLE Tritium Monitoring Systems within 8 hours. One of the monitors must be a BTM. This action ensures that the building leak detection capability is restored within 8 hours. The 8 hour Completion Time is acceptable based on analysis which demonstrates that the crack growth rate during an 8 hour period is negligible (Refs. 2 and 3).

Action C.2 requires monitoring of any 105 Building process areas ventilation exhaust air streams with an OPERABLE Tritium Monitoring System within 7 days. The 7 day Completion Time is acceptable based on the OPERABILITY of the building stack monitors, which will detect leakage in any of the 105 Building areas. The inability to identify

(continued)

INFORMATION ONLY
Process Water System Leakage Detection Instrumentation
B 3.4.1

BASES

ACTIONS
(continued)

C.1 and C.2

the process area source of a leak detected by the building stack monitors is acceptable for a 7 day period due to the very low likelihood of a significant leak during the period.

D.1

During REACTOR STARTUP or REACTOR OPERATION, if the Required Actions of Conditions A, B, or C are not met within the required Completion Times, the reactor must be in REACTOR SHUTDOWN in 90 minutes. This Completion Time provides for a controlled reactor shutdown.

E.1

Any time except REACTOR STARTUP and REACTOR OPERATION if Conditions A, B, or C exist, in the event that any leak detector is inoperable, the Required Action is to restore the instrument as soon as possible by initiating action to restore the instrument IMMEDIATELY. This action will restore the plant to an acceptable safe condition in the earliest time practical and is the only viable ACTION since the only alternative would be to place the plant in a condition that is outside the LCO APPLICABILITY by discharging the core and draining the Process Water System.

SURVEILLANCE
REQUIREMENTS

SR 3.4.1.1

Surveillance Requirement 3.4.1.1 requires the performance of a CHANNEL CHECK of the Tritium Monitoring Instrumentation. A CHANNEL CHECK of the Tritium Monitoring Instrumentation shall be the comparison of a STM with a BTM, or a BTM with a BTM. A CHANNEL CHECK is based on the assumption that two channels monitoring the same parameter should be reading approximately the same. Agreement is based on a combination of the channel instrument uncertainties, including control isolation, indication, and readability. If a channel is outside of the match criteria, it may be an indication that the transmitter or

(continued)

BASES

**SURVEILLANCE
REQUIREMENTS
(continued)**

SR 3.4.1.1

the electronics have drifted outside of their limit. If the channels are within the match criteria, it is a reasonable assumption that the channels are within specification with respect to their Alarm Setpoints. The surveillance interval of 48 hours is based on the recommendation given in Reference 2.

SR 3.4.1.2

Surveillance Requirement 3.4.1.2 requires the performance of a CHANNEL TEST of the Berthold Tritium Monitoring Instrumentation. The CHANNEL TEST is the injection of a known quantity of elemental tritium gas into the monitor, and determining the output of the monitor from this known input. The output of the monitor is then compared to the input of the monitor. The surveillance interval of 120 days is based on the recommendation given in Reference 2.

SR 3.4.1.3

Surveillance Requirement 3.4.1.3 requires the performance of a CHANNEL CALIBRATION of the Berthold Tritium Monitoring Instrumentation. The CHANNEL CALIBRATION is the adjustment of the CHANNEL as necessary, such that its output corresponds with acceptable accuracy to known values of the parameter which the CHANNEL monitors. The surveillance interval is based on the results of the CHANNEL TEST as recommended by Reference 2.

SR 3.4.1.4

Surveillance Requirement 3.4.1.4 requires the calibration of the Stack Tritium Monitoring Instrumentation using the BTM Instrumentation. The surveillance interval of 120 days is based on the recommendation given in Reference 2.

REFERENCES

1. RTM-4898, "New Berthold Tritium Monitors"(U), February 11, 1988.
2. RTR-2628, "Leak Detection Instrumentation"(U), February, 1989.
3. WSRC-RP-90-99, "Reactor Materials Program Leak Detection System Reliability"(U), March, 1990.

B 3.4 PROCESS WATER SYSTEM

B 3.4.2 Process Water System Operational Leakage

BASES

BACKGROUND

The Process Water System Pressure Boundary (PWSPB) consists of components whose connection joints are made by welding or bolting. PWSPB piping is either of seamless construction, or rolled and welded. The PWSPB is isolated from other plant systems by valves and heat exchanger tubes. During plant life, these interfaces can produce varying amounts of process water leakage through either normal operational wear or mechanical deterioration.

A limited amount of process water is routinely lost from the Process Water System and from auxiliary systems within the reactor building. The mechanisms by which these losses occur can be divided into two broad categories: normal/expected operational leakage, and leakage due to cracks in the PWSPB.

Examples of normal/expected operational leakage include (but are not limited to):

1. Process water that evaporates from the exposed surfaces of safety and control rods as they are removed from the reactor during startup.
2. Evaporation that occurs from the surface of process water in the safety/control rod guide tubes, the vacuum breakers, and the supplementary vents. Each of these is attached to the actuator forest in the process room and is open to the confinement atmosphere.
3. Process water spills that occur in the Purification Area during routine change-out/maintenance of deionizers or other equipment.
4. Process water losses that are the result of improperly tightened flange connections, worn/damaged gaskets, or worn/damaged valve stem packing. (Note: Water from these sources is usually captured in a leak collection system.)

(continued)

BASES

**BACKGROUND
(continued)**

5. Helium blanket gas that contains process water vapor, and slowly leaks from the blanket gas system into the confinement atmosphere.
6. Seepage from the process water heat exchangers at the head flange.

Definitions have been developed that enable the various types of leakage to be categorized according to the ability of operating personnel to locate, quantify, and isolate the leak, and whether or not process water enters the confinement atmosphere.

The safety significance of PWSPB leakage can vary widely depending on the source of the leakage, leakage rate, and rate of change of the leak rate. Therefore, the detection and monitoring of process water leakage within the 105 Building is necessary. Separating the identified sources of leakage from unidentified sources is necessary to provide prompt and quantitative information to the operators to allow them to take immediate action should a leak occur that may be detrimental to the safety of the reactor.

An early warning of PRESSURE BOUNDARY LEAKAGE is provided by the Tritium Monitoring Systems that monitor the ventilation exhaust air streams in the 105 Building for tritium, which is present in the process water. PRESSURE BOUNDARY LEAKAGE would first appear as UNIDENTIFIED LEAKAGE, until its location could be positively identified by visual inspection.

**APPLICABLE
SAFETY ANALYSES**

The safety significance of leakage in the PWSPB is that this leakage could be indicative of an impending gross failure of the Process Water System. Therefore the detection, identification, quantification, and monitoring of process water leakage is essential so that appropriate action can be taken prior to a gross failure of the PWSPB.

Margins of safety against crack instability for normal plus seismic loadings have been defined by the Nuclear Regulatory Commission's Piping Review Committee in NUREG-1061 (Ref. 1). The margin on instability crack length is the most limiting margin of safety for the piping (Ref. 2). NUREG-1061 requires a margin of two between the leakage crack size a and instability crack size. This requirement implies that a crack one-half the instability length be

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

detectable by leak rate monitoring. This provides an appropriate margin of safety. The instability crack length the crack length beyond which a severe pipe failure becomes imminent.

This crack length differs for various pipe diameters, and can be correlated to leak rates. The instability crack length and leak rates have been evaluated for all sizes of the process water piping, ranging from three to twenty-four inches in diameter. A 1000 pounds/day leak rate provides a minimum safety margin of two on the instability crack length in all of the process water piping with a diameter of three inches and larger (Refs. 2 and 3). A leak rate based on previous leak-before-break analyses (Ref. 4) yielded a leak rate of 300 pounds/day corresponding to the minimum safety margin of two on the instability crack length. The rate of 300 pounds/day is chosen for conservatism. The calculated leak rate for a crack at the same location with a length that is just short of instability, is approximately 30,000 pounds/day (Ref. 2).

Three measured parameters are used in the calculation of process water leakage: the process water tritium concentration, the 105-Building ventilation air flow rates, and the ventilation air tritium content. Each of these parameters has an associated measurement/analytical uncertainty. The process water tritium concentration has an analytical uncertainty of $\pm 5\%$; the 105 Building ventilation flow rate measurement has an uncertainty of $\pm 15\%$; and the tritium content in 105 Building ventilation air has a measurement uncertainty of 30% (Refs. 5 and 6). The total uncertainty is 34%. This uncertainty was then applied such that the sum of the measured leak rate plus the uncertainty in the measurement would not exceed the 300 pound/day limit determined from the instability crack length. An instantaneous UNIDENTIFIED LEAKAGE Rate of 220 pounds/day was calculated after taking into account the uncertainty.

Two additional factors must also be considered in the determination of process water leakage; the time required for the tritium concentration in the 105 Building to reach a detectable threshold (which is dependent on the evaporation rate) and the time that it takes for the tritium in the 105 Building to reach the detector in the stack (transport time). Some calculations were performed (Ref. 10) to approximate the evaporation rate. The calculations assumed a constant leak rate of 220 pounds/day, and that the water strikes the floor of the 105 Building and spreads out to

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

form a pool of uniform thickness. The results of the calculations indicated that the time it would take for the tritium to reach a detectable threshold from a 220 pound/day leak rate is approximately 80 seconds for a Berthold Tritium Monitor and approximately 26 minutes for a Stack Tritium Monitor. A series of ventilation tests were conducted in the L Area 105 Building in 1984 (Ref. 11). During these tests, a sulfur hexafluoride tracer gas was released above the reactor and the transport time to the stack was measured. The time it took for the gas to reach the stack was approximately 3 minutes. If a leak occurred in the process water pump room or the heat exchanger bay, the transport time would be approximately the same as from the reactor room. The evaporation rate determined in Reference 10 does not consider the leak going into a sump or a trench that might impede evaporation. However, studies have shown (Ref. 12) that the time it would take for a 220 pound/day leak rate to propagate into a break at the most limiting location in the Process Water System piping would be very much longer (years) than the time it would take for the tritium monitors to detect the leakage.

The 34% uncertainty is not applied to the cumulative UNIDENTIFIED LEAKAGE of 96 pounds in a 24-hour period because this value was derived from a statistical analysis performed on actual system data, and not from calculated values.

LCO

This LCO limits reactor operation in the presence of UNIDENTIFIED LEAKAGE and PRESSURE BOUNDARY LEAKAGE from the Process Water System.

A.

The limit of no PRESSURE BOUNDARY LEAKAGE prohibits reactor operation with confirmed leaking through-wall cracks in the process water system. This limit is based on the conservative philosophy that if a known leaking through-wall crack in the PWSPB piping or components exist, the reactor should be shut down. The crack should then be repaired so as to eliminate the possibility that it may further propagate to a size that could be detrimental to the overall system integrity during reactor operation.

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INFORMATION ONLY

BASES

LCO
(continued)

B.

UNIDENTIFIED LEAKAGE is limited to a cumulative leakage of 96 pounds in a 24-hour period because analysis of daily process water loss data from past reactor operation (Refs. 7 and 8) has indicated that leakage above this amount is abnormal, and could be indicative of PRESSURE BOUNDARY LEAKAGE.

C.

An instantaneous UNIDENTIFIED LEAKAGE rate of 220 pounds/day is based on a conservative limit which has been reduced to account for measurement uncertainties and corresponds to the minimum leakage rate from a crack which is one-half the instability crack length for all process water piping greater than or equal to 3 inches in diameter.

APPLICABILITY

A limit on process water leakage is implemented for REACTOR STARTUP and REACTOR OPERATION to minimize the risk of core damage due to a large process water leak. Because the heat generation rate (fission product decay) during REACTOR SHUTDOWN is much less than that during REACTOR OPERATION, and because the process water system is subjected to lower levels of pressure, temperature, and vibration, the risk of core damage from a Process Water System leak or break is much less. Therefore no process water leakage limit is required by the Technical Specifications during REACTOR SHUTDOWN.

ACTIONS

A.1

If the UNIDENTIFIED LEAKAGE exceeds 96 pounds in a 24-hour period, but the instantaneous UNIDENTIFIED LEAKAGE rate is less than 220 pounds/day, a Completion Time of 2 hours is provided to identify or isolate the source of the leak. The 2 hour completion time is acceptable based on meeting the instantaneous unidentified leakage rate of 220 pounds/day and considering the calculated crack growth rate in the PWSPB.

(continued)

BASES

ACTIONS
(continued)

B.1

If there is any PRESSURE BOUNDARY LEAKAGE or if the instantaneous UNIDENTIFIED LEAKAGE rate is greater than 220 pounds/day, the reactor must be in REACTOR SHUTDOWN in 90 minutes. The Completion Time of 90 minutes is provided for a controlled reactor shutdown. Any leakage greater than the limits in the LCO may be indicative of significant PWSPB degradation, and continued reactor operation could cause further deterioration of the PWSPB.

C.1

If the Required Action of Condition A is not met within the required Completion Time, the reactor must be in REACTOR SHUTDOWN in 90 minutes. This Completion Time provides for a controlled reactor shutdown.

SURVEILLANCE
REQUIREMENTS

SR 3.4.2.1

Verifying that the cumulative UNIDENTIFIED LEAKAGE is less than or equal to 96 pounds in a 24-hour period once per 12 hours will ensure that the integrity of the PWSPB will be maintained. A cumulative UNIDENTIFIED LEAKAGE greater than 96 pounds in a 24-hour period may be indicative of PRESSURE BOUNDARY LEAKAGE. The 12-hour surveillance frequency is conservative based on the leak-before-break analysis results.

SR 3.4.2.2

Verifying that the instantaneous UNIDENTIFIED LEAKAGE rate is less than or equal to 220 pounds/day once every 12 hours provides assurance that there are no cracks in the PWSPB that would approach one-half of the instability crack length between surveillances.

The surveillance interval of 12 hours is conservative with respect to the calculated crack growth rates of $10E-5$ to $10E-7$ inches/hour (Ref. 9). The 12-hour interval emphasizes the relative importance of early leak detection in the prevention of gross failure of the Process Water System piping, which would lead to a Loss of Coolant Accident.

(continued)

BASES (continued)

REFERENCES

1. NUREG-1061, Vol. 3, "Evaluation of Potential for Pipe Breaks,"(U) November 1984.
 2. WSRC-RP-90-91, "Reactor Materials Program Leak - Before Break Analysis of the Reactor Process Water System Piping "(U), March 1990.
 3. WSRC-RP-90-99, "Reactor Materials Program Leak Detection Reliability"(U), March, 1990.
 4. DPST-88-426, "Reactor Materials Program Moderator Leak Detection in SRP Reactors"(U), March, 1988.
 5. RTR-2628, "Leak Detection Instrumentation"(U), February, 1989.
 6. RRD-RTS-910153, "Uncertainty in Process Water Leakage Measurement Parameters"(U), May, 1991.
 7. SCD-PGM-890007, "Extreme-Value Analysis of Historical D₂O Loss Data"(U), June 1989.
 8. RTR-2668, "Operational D₂O Leakage Limits"(U), July 1989.
 9. WSRC-RP-90-95, "Reactor Materials Program, a Derivation of a Nominal IGSCC Growth Rate for SRS Reactors"(U), March 1990.
 10. WSRC-RP-90-236, "Time Required to Detect a Leak of Tritiated Water in a Reactor Building by a Stack Monitor"(U), February 1990.
 11. DPST-85-336, "L Area Ventilation Tests-1984"(U), March 1985.
 12. SRL-EDG-900069, "Crack Growth Rate in PWS Piping"(U), February 1990.
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B 3.4 PROCESS WATER SYSTEM

INFORMATION ONLY

B 3.4.3.1 Process Water System - Operating

B 3.4.3.2 Process Water System - Zone Average Assembly Flow

BASES

The Process Water System (PWS) provides circulation of heavy water (D_2O) through the reactor for neutron moderation and heat removal. The PWS also constitutes the primary reactor coolant boundary and provides a high-integrity leak barrier against loss of moderator and release of radioactive fission and corrosion products.

Process water is circulated through the reactor by six parallel process water pumping systems (loops) at a total maximum rate of 150,000 gpm. Each loop consists of a PWS Bingham pump, two parallel heat exchangers, and a septifoil system inlet line. Each process water loop can be isolated from the reactor tank by closing the loop's process water Bingham pump suction isolation valve and both heat exchanger outlet valves (rotovalves). The PWS Bingham pumps pump the process water to the heat exchangers. Most of the process water then flows through the rotovalves and then enters the reactor. A small portion of the water exiting the heat exchanger flows through piping to the bottom of the reactor to supply reactor upflow cooling via the septifoils.

There are 61 septifoil positions in the reactor tank. These 61 septifoil positions contain control rods that are used to control the radial and axial flux distribution during normal operation. During reactor operation, neutrons are absorbed in the control rods and subsequently, the absorber neutron liberates heat that is deposited within the control rod. In addition, some of the penetrating radiation produced elsewhere in the reactor is deposited in the control rods. Therefore, the control rods must be cooled to prevent overheating.

Three loops on each side of the reactor feed a ring header beneath the reactor. The two 6-inch portions of the ring header are connected by 3-inch cross-tie lines. Each 6-inch section of the ring header supplies coolant flow to half of the septifoils through 1.25 inch lines. These lines supply flow to upflow pins that penetrate the bottom of the reactor and supply flow to each septifoil. The coolant flows up through the septifoil, cooling the control rods as it passes, and then exits the septifoil and flows

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Process Water System - Operating B 3.4.3.1
Process Water System - Zone Average Assembly Flow B 3.4.3.2

BASES (continued)

into the moderator through a set of slots and holes at the top of the septifoil housing.

Every 12 months and following any changes to the Core Operating Report (COR) specified septifoil flow limits, SR 3.4.3.1.7 requires verification that the PWS septifoil supply valves are positioned such that the septifoil ring header flow is within the limits defined in the COR, thereby ensuring that adequate control rod cooling is provided. Within 48 hours prior to locking the -20' and -40' Radiologically Controlled Areas AND prior to entry into REACTOR STARTUP after shutdown required by ACTION B.1, SR 3.4.3.1.5 requires verification that the septifoil supply valves are properly positioned (as previously established by SR 3.4.3.1.7) and deenergized, thereby ensuring the desired valve positions are established prior to REACTOR STARTUP and are maintained during REACTOR STARTUP and REACTOR OPERATION with the -20' and -40' Radiologically Controlled Areas locked.

Every 7 days, SR 3.4.3.1.6 requires verification that the septifoil supply valves are deenergized, thereby ensuring that the proper valve positions are maintained. If one or more septifoil supply valves are found energized, ACTION B.1 requires the plant to be in REACTOR SHUTDOWN within 90 minutes. It was conservatively assumed that an energized septifoil supply valve could have moved off its required position. The assumed mis-positioned valve could result in the septifoil ring header flow being outside the limits defined in the COR. Verification of valve position is not possible during REACTOR OPERATION to determine if the energized valve did actually move such that COR limits were exceeded. The plant would enter REACTOR SHUTDOWN and verification performed that the septifoil supply valves are properly positioned and deenergized.

The system of thermal and hydraulic limits used to prevent core damage and activity release is based on Zone Average Assembly Flow at a rate estimated for six Bingham PWS pump AC motors running with PWS rotovalves, PWS Bingham Pump Suction isolation valves and septifoil valves throttled. In addition, changes in process water flow could change process water temperature and introduce reactivity changes. The reactivity change for starting an idle process water pump could be on the order of 0.0001 to 0.0004 k. Changes such as this, although not large, are not under the direct control of the reactor operator, and should be avoided.

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Process Water System - Operating B 3.4.3.1
Process Water System - Zone Average Assembly Flow B 3.4.3.2

BASES (continued)

REFERENCES

1. DPSTM-110. Effluent Temperature and BOSF_N Limits for Fuel and Target Assemblies and Reactor Effluent Temperature Limits.(U)
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Process Water System - Shutdown B 3.4.3.3
PWS - Natural Circulation Bulk Moderator Temperature Limit B 3.4.3.4
PWS - Natural Circulation Reactor Level B 3.4.3.5

B 3.4 PROCESS WATER SYSTEM

B 3.4.3.3 Process Water System (PWS) - Shutdown
B 3.4.3.4 PWS - Natural Circulation Bulk Moderator Temperature Limit
B 3.4.3.5 PWS - Natural Circulation Reactor Level

BASES

The Process Water System flow must be sufficient to prevent bulk boiling in any assembly because bulk boiling could restrict flow through the assembly and cause overheating. Bulk boiling would occur with an assembly effluent temperature of about 113°C. The 80°C limit with at least three PWS Bingham pumps driven by their DC motors and at least one associated heat exchanger on at least gravity cooling and forced process water flow gives ample margin from boiling.

Experiments and calculations indicate that irradiated fuel assemblies can be adequately cooled by natural convection after the heat generation rate has decreased sufficiently, provided that requirements a and b are met.

- a) The bulk temperature of the process water must be low enough to:
 - 1) Provide an adequate temperature differential between the coolant in the assemblies and the process water.

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INFORMATION ONLY

Process Water System - Shutdown B 3.4.3.3
PWS - Natural Circulation Bulk Moderator Temperature Limit B 3.4.3.4
PWS - Natural Circulation Reactor Level B 3.4.3.5

BASES

(continued)

- 2) Provide an adequate heat sink capacity. The Mark 22 fuel maximum allowable heat generation of 6 kW is based on a bulk moderator temperature of 50°C. Assumption of higher bulk process water temperatures would result in lower acceptable heat generation values and would decrease the system heat capacity.
 - b) Natural convection cooling of assemblies is dependent upon circulation of coolant from the inner coolant annuli to the outer housing. The no forced circulation reactor level must not be less than overflow, since some circulation is blocked if the liquid level of the reactor is decreased.
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INFORMATION ONLY
Process Water System - Bulk Moderator Temperature Limits
B 3.4.4

B 3.4 PROCESS WATER SYSTEM

B 3.4.4 Process Water System - Bulk Moderator Temperature Limits

BASES

Effluent process water from fuel and target assemblies travels generally upward in the center portions of the bulk moderator space. As it travels upward, the subcooling of the process water decreases because of the reduced hydrostatic head. If the enthalpy of any portion of the process water at any elevation exceeds its enthalpy at the saturation temperature, process water will flash and form steam voids. Voids formed during flashing would reduce power because of negative process water void coefficients. This, in turn, would reduce the process water temperature. The voids would then collapse causing reactivity and power to increase. The resulting oscillations would decrease the stability of normal reactor operation.

On this basis, process water temperatures are restricted to protect against local flashing (Ref. 1). The temperatures are limited to values less than the process water saturation temperatures corresponding to the pressure at the scram setpoint of the blanket gas pressure monitors (see LCO 3.3.1) plus the hydrostatic head of process water.

The reactor is automatically shut down by either of two scram circuits if blanket gas pressure decreases below the setpoint pressure; thus, this pressure establishes the lower boundary of process water saturation temperature during normal reactor operation. Saturation temperatures corresponding to a typical scram setpoint of 4.0 psig are as follows:

<u>Elevation</u> <u>(feet from reactor midplane)</u>	<u>Saturation Temperature</u> <u>°C</u>
+6	111.2
+3	113.1
0	114.9
-3	116.6

(continued)

INFORMATION ONLY

Process Water System - Bulk Moderator Temperature Limits B 3.4.4

BASES (continued)

REFERENCES

1. DPST-63-520, Rev. 1, Heavy Water Temperature Limits, February 1964. (Secret/RD)
 2. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 15.1.2.3.3, Cavitation and Reactor Effluent Temperature (U)
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B 3.4 PROCESS WATER SYSTEM

B 3.4.5.1 Process Water Activity

BASES

A few reactor charges have been operated with fission product activity in the process water resulting in 2-hour beta-gamma activities in the range from 5×10^6 to 1×10^7 c/m/ml. In these instances high radiation levels were encountered in some working areas. Additional shielding reduced these levels so that in the more recent instances only minor accommodations were needed to continue normal work. Deionizer life was shortened and stack releases were increased, but practicable methods were available to meet these problems. Radioactivity transferred to the disassembly basins was controlled by the basin deionizer filter system with little or no increase in liquid releases to the environment.

The choice of an upper limit for fission product activity is judgmental. However, it is not apparent that operation at levels significantly in excess of those already experienced is warranted. Instead, continued effort will be made to maintain activity levels at values as low as practicable by quality control measures in the fabrication process, prompt discharge of known failures, and periodic discharge of semipermanent lattice components. A 2-hour value of 2×10^7 c/m/ml corresponds to a specific activity value slightly less than 10^5 curies, the activity release limit specified as the criterion for the BOSF_N TECHNICAL LIMITS in section 2.2. The highest value observed during regular operation was 10^7 c/m/ml (Ref. 1).

At beta-gamma activities above 2×10^6 c/m/ml there are occasional errors in determining the beta-gamma activity. It is not intended that the reactor be shut down because of a single erroneous measurement. High activity levels should be confirmed by two or more evaluations. It is recognized that up to 48 hours elapse from time of sampling until the analytical results are obtained.

(continued)

BASES (continued)

REFERENCES

1. DPSP-67-1-7, Progress Report of Works Technical Department for July, 1967. (Secret/RD)
 2. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 15.1.2.3.2, Burnout Risk (BOR) (U)
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INFORMATION ON

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Process Water Systems Activity Monitoring Systems
B 3.4.5.2

B 3.4 PROCESS WATER SYSTEM

B 3.4.5.2 Process Water Systems Activity Monitoring Systems

BASES

Release of fission products can result in high activity levels

- a) in the process water, thus interfering with early detection of subsequent failures,
- b) in the external hydraulic system of the reactor, rendering maintenance difficult,
- c) on the surfaces of the fuel, target, or housing assemblies, leading to contamination of the disassembly basin, and
- d) in the blanket gas, leading to atmospheric releases of radioactivity.

It is necessary to monitor for fission product release to permit corrective action before the levels become excessive. Two of the following instruments, or suitable equivalents, shall be OPERABLE whenever the reactor is operating above low power.

- o Fuel Failure Blanket Gas Analyzer
- o Low Energy Gamma Monitor (LEGM)
- o Blanket Gas Activity Monitor
- o 10 Minute Delay Monitor

REFERENCES

1. DPSP-67-1-7, Progress Report of Works Technical Department for July, 1967. (Secret/RD)

B 3.4 PROCESS WATER SYSTEM

B 3.4.6 Plenum Chamber Blanket Gas

BASES

The intent of the blanket gas system is to provide a means to transport and safely dispose of radioactive and flammable gases resulting from reactor operation. Helium is specified as the blanket gas because it does not become radioactive and is not converted by radiolysis into any undesirable product.

REFERENCES

1. DPSP-70-1324. F. B. Longtin, Basis for Limits on Blanket Gas Composition - 105 Reactors, RTR-1122, September 10, 1970. (U)
2. DPSP-71-1225. F. B. Longton. Basis for Limits on Blanket Gas Composition - 105 Reactors (Part II), RTR-1122, April 1, 1971. (U)
3. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 5.4.4 Blanket Gas System (U)

B 3.4 PROCESS WATER SYSTEM

B 3.4.7 Process Water System Bingham Pump DC Motors

BASES

NOTE

References to core damage in references 1 and 3 may be superceded by more recent analysis information in ref 2.

Cooling of the reactor core following a loss-of-coolant accident would be less effective than stated in References 1 and 2 with fewer than six DC motors operating. Following a loss-of-coolant accident, coolant recirculation with the process water pumps is required to supplement cooling provided by the emergency light water addition. Procedures specify shutdown of the AC motors after a period of post accident operation to lower system pressure and conserve coolant, and reliance is placed on the DC motors to drive the pumps. With only five DC motors operating the core damage for the worst case loss-of-coolant accident would be 4-7% vs. 1-3% with six DC motors operating (Refs. 1 and 3).

When the process water pumps are being driven by the DC motors only, the pump flow is a function of the power supplied to the DC motor by the diesel-driven generator. This in turn is a function of the generator power output during normal operation with the AC and DC motors operating; the generator power is adjustable and is specified in operating procedures. The 4500 gpm flow requirement is the minimum that would prevent core melting in the event of a severe loss-of-coolant accident in which only five pumps are effective because of a pipe failure in one system and the reactor is partially drained (Ref. 1).

REFERENCES

1. DPST-70-463, Analysis of the Savannah River Reactor Emergency Core Cooling System, October, 1970. (U)
 2. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 15.6.3, Loss of Coolant Accident (U)
 3. DPST-73-495, Rev. 1, Safety Implications of Reduced Diesel Oil Consumption in 100 Areas, November 15, 1973, (U).
 4. DPSP-74-1013, Power Reduction to DC Motors, January 30, 1974. (U)
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B 3.4 PROCESS WATER SYSTEM

B 3.4.8.1 Deuterium Concentration

BASES

Nuclear safety of a reactor could be compromised by combustion of a flammable blanket gas mixture either within the gas plenum or above the reactor tank. If sufficient energy is released by the combustion, the reactor structure could be damaged by heat or pressure in such a way as to prevent insertion of safety rods. This LCO has as its objectives to first prevent such damage and ultimately to assure that, in the unlikely event of actual gas combustion, the safety rods will have been inserted before such damage occurs.

A flammable gas mixture of helium, nitrogen, oxygen, and deuterium results only when the concentrations of deuterium and oxygen simultaneously exceed limits which depend on the nitrogen concentration (Ref. 1). Radiolysis of process water to produce deuterium plus oxygen can drive the blanket gas concentration into the flammable region, as can generation of deuterium by corrosion and simultaneous addition of oxygen to recombine with it, if recombination of these gases in the recombiner is inadequate. A flammable gas mixture may also result from mixing air with a blanket gas which is not flammable per se. This could occur as a result of vacuum breaker relief when the blanket gas contained less than 3 percent oxygen and a sufficiently high concentration of deuterium. The resulting flammable mixture could be found either within the gas plenum (if the vacuum breaker relief was the result of a REACTOR SHUTDOWN) or above the reactor tank.

If this LCO is met, no flammable gas mixture can exist within the blanket gas system nor can it result from mixing of blanket gas with air. There is no flammable mixture of helium, nitrogen, oxygen, and deuterium containing less than 3.8 percent deuterium, regardless of content of oxygen or nitrogen. Because a mixture of air will reduce the deuterium concentration, no flammable mixture can be produced by mixing air with any combination of these gases containing less than 3.8 percent deuterium.

(continued)

BASES (continued)

For normal reactor operation with less than 5 percent nitrogen, this requirement includes a wide margin of safety between the allowed 3.8 percent deuterium and the 7.0 percent flammability limit. The 7.0 value already contains adequate allowance for uncertainties in the experimental data on flammability limits. Consequently, very little additional margin of safety is needed to allow for uncertainties in measurement of the blanket gas composition when the nitrogen concentration is below 5 percent.

To assure that the LCO is met, adequate measurement of the concentration of deuterium in the blanket gas will be made during REACTOR OPERATION.

REFERENCES

1. DPSP-70-1324. F. B. Longtin, Basis for Limits on Blanket Gas Composition - 105 Reactors, RTR-1122, September 10, 1970. (U)
2. DPSP-71-1124. F. B. Longtin. Basis for Limits on Blanket Gas Composition - 105 Reactors (Part II), RTR-1225, April 1, 1971. (U)
3. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 5.4.4, Blanket Gas System (U)

RESTRICTED ONLY

Oxygen Concentration
3.4.8.2

B 3.4 PROCESS WATER SYSTEM

B 3.4.8.2 Oxygen Concentration

BASES

The intent is normally to operate the reactor with very low oxygen content in both the blanket gas and the process water to minimize corrosion of stainless steel. The normal concentration of oxygen in the blanket gas will be below 21 percent by volume. However, as shown in Reference 1, no harm will result from operation for brief periods with higher oxygen concentrations (such as would result when air is present in the blanket gas system) so long as Technical Specification 3.4.8.1 is met.

The 21 percent limit guards against a minor flammability hazard from blanket gas leaks with very high oxygen concentrations. Such concentrations of oxygen could be reached only by excessive addition of oxygen to the blanket gas. Oxygen is added to the gas system, when necessary, only in small amounts and under strict procedural controls.

REFERENCES

1. DPSP-70-1324. F. B. Longtin, Basis for Limits on Blanket Gas Composition - 105 Reactors, RTR-1122, September 10, 1970. (U)
 2. DPSP-71-1124. F. B. Longtin. Basis for Limits on Blanket Gas Composition - 105 Reactors (Part II), RTR-1225, April 1, 1971. (U)
 3. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 5.4.4, Blanket Gas System (U)
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B 3.4 PROCESS WATER SYSTEM

INFORMATION ONLY

B 3.4.8.3 Process Water Quality - pH and Conductivity

BASES

BACKGROUND

The Process Water System (PWS) materials have been selected and proven through service, to meet the expected service conditions. Component surfaces in contact with process water are Type 304 stainless steel. The wide use of Type 304 stainless steel was adopted to provide the best available material that featured the necessary strength, pressure-retaining, and corrosion-resistant characteristics. Type 304L stainless steel is generally used for replacement piping and components because of its increased resistance to intergranular stress corrosion cracking (IGSCC).

Water chemistry specifications are maintained by the purification system and by additions of nitric acid. The system prevents deleterious effects on stainless steel and minimizes corrosion of serviceable aluminum components by controlling pH and conductivity.

Gross degradation from outside sources that would result in exceeding limiting criteria are not considered credible, except for chlorides. Control of chlorides to prevent stress corrosion cracking is implemented. The following measures have been adopted to reduce stainless steel corrosion:

- Eliminating sources of chlorides from reactor system.
- Maintaining constant moderator pH throughout REACTOR OPERATION and during REACTOR STARTUP.
- Keeping dissolved oxygen concentration as low as possible, consistent with the requirements of pH control. If the oxygen falls to zero then the deuterium produced by radiolysis begins to react with the nitrate present. This process can result in the formation of ammonia and an increase in pH to the basic side, which swiftly attacks aluminum. By keeping oxygen present, these conditions are avoided and pH control is maintained.

(continued)

BASES (continued)

INFORMATION ONLY

APPLICABLE
SAFETY ANALYSES

The bases of this Technical Specification are operating experience and published information concerning the corrosion of aluminum and stainless steel, particularly in heavy-water reactor systems, and the decomposition of heavy water by radiolysis.

The PWS utilizes heavy-water. The measured pH of the process water is an empirical value that is related to pD by:

$$\text{pH} = \text{pD} - 0.4$$

pD represents the absolute deuterium concentration, which is used for the calculation of conductivity (Ref. 1).

1. It has been demonstrated during sustained REACTOR OPERATION at the Savannah River Site that the corrosion rate of aluminum components, and the associated problems of high radioactivity and turbidity in the process systems, is substantially reduced by control of the acidity of the process water (Ref. 2). This is confirmed by information obtained from other sites (Refs. 3, 4, and 5). The specified pH range of 4.1 to 7.2 will not impact significantly on corrosion of stainless steel. It was found that the aluminum corrosion decreased as the pH was decreased to 4.3. No tests have been made at SRS with pH less than 4.3; however, since the aluminum corrosion rate does not change rapidly with pH when operating near the minimum value of aluminum corrosion (Ref. 5), a limit on pH of 4.1 was chosen for the maximum acidity for this Technical Specification. The lower limit provides flexibility in writing operating procedures and to allow a working margin for instrument errors.

The upper limit on pH (7.2) is established to ensure gadolinium solubility if the Supplementary Safety System is used. Formation of insoluble gadolinium hydroxide would be minimal (< 1%) at process water pH < 9.1 (Ref. 9). The upper pH limit of 7.2 provides a large margin from significant gadolinium precipitation.

(continued)

INFORMATION ONLY

BASES (continued)

APPLICABLE
SAFETY ANALYSES
(continued)

2. The control of process water pH requires that conductivity be permitted to increase in accordance with the physico-chemical relationships between ionic content of process water and the ionic mobility of the ions present in the process water. For pH = 4.6 the corresponding conductivity of the process water, provided all of the conductivity is a result of the deuterium-anion pair, is 3.0 micromho/cm. Because a small amount of salt ions would sometimes be present in the process water as a result of corrosion, a slightly higher conductivity than the acid contribution is desirable for control purposes. A conductivity limit of 3.5 micromho/cm is chosen to provide sufficient excess to allow for a small amount of corrosion products (0.5 micromho/cm) when maintaining the pH at 4.6. A conductivity of 3.5 micromho/cm corresponds to pH = 4.4 (approximately) if all of the ionic material is the deuterium-anion pair; therefore, the conductivity is the limiting variable.

To permit continuous reactor operation in the event that ionic impurities get into the process water system, the reactor may be operated with conductivity greater than 3.5 micromho/cm, but not greater than 5.0 micromho/cm (pH = 4.1 minimum), while the excessive ionic material is being removed by the purification system. This excess conductivity would correspond to approximately 1 ppm of salt impurity in the reactor system when operating at pH = 4.6.

The maximum conductivity possible with the maximum amount of nitric acid added is approximately 3.5 micromho/cm. The additional 1.5 micromho/cm was added to allow for process upsets which might result in some ions circulating in the process water, it amounts to approximately 0.2 ppm of ions such as chloride and sulfate. Since this is approximately 200 times the concentrations normally present, engineering judgement has determined that 5.0 micromho/cm is an adequate margin to allow continued operation. Anything in excess of this amount is liable to result in deleterious effects to the PWS such as stress cracking and the formation of unwanted radioactivities which would result in increased personnel exposure, or any number of other unwanted events.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES
(continued)

No significant decomposition of process water has been observed in experience at the Savannah River Site. Similar results have been obtained in other water-moderated reactors; this is attributed to recombination of free radicals formed by radiolysis in the high gamma fields present in the reactors. Published data (Ref. 7), indicate that no appreciable decomposition of heavy water occurs at conductivities less than about 5.0 micromho/cm. On these bases, and the fact that the Savannah River Site has operated without significant water decomposition with conductivities as high as 3.0 micromho/cm, no significant decomposition of heavy water is expected when operating under this Technical Specification.

LCO

Requiring process water to have a pH between 4.1 and 7.2, and conductivity of less than 5.0 micromho/cm at 25°C, significantly reduces the probability of PWS piping degradation.

Both conductivity and pH are temperature dependent and the values at temperatures other than 25°C are not useful in determining the amount of materials causing the condition. The standard reference temperature for both measurements is 25°C. The monitors and readouts in the reactor have instrumentation which corrects the readings to 25°C so they are always valid, no matter the temperature of the process water.

APPLICABILITY

The PWS requirements stated are in effect throughout REACTOR OPERATION and during REACTOR STARTUP. This applicability reduces the probability that the process water will create any deleterious effects on stainless steel and will minimize corrosion of aluminum components.

(continued)

BASES (continued)

INFORMATION ONLY

ACTIONS

A.1

If the process water is not within the stated pH limits, Required Action A.1 requires the restoration of the process water to within pH limits in 24 hours. Process water not within the pH limits can potentially degrade the integrity of the PWS. The completion time period of 24 hours has been determined to be acceptable by engineering judgement based on PWS pH controls.

B.1

If the Required Action A.1 of Condition A is not met within the specified time period or the conductivity of the process water is in excess of its stated limit, the plant must be placed in the safest possible state. Therefore, Required Action B.1 requires the plant to be in REACTOR SHUTDOWN within 90 minutes.

The process water could potentially have deleterious effects to the PWS such as IGSCC or the formation of unwanted radioactivities which could result in increased personnel exposure.

SURVEILLANCE
REQUIREMENTS

SR 3.4.8.3.1

This surveillance ensures that the process water quality is within the stated conductivity limits (conductivity ≤ 5.0 micromho/cm), such that the probability of PWS piping degradation is significantly reduced. The maximum conductivity possible with the maximum amount of nitric acid added is approximately 3.5 micromho/cm. The additional 1.5 micromho/cm was added to allow for process upsets which might result in some ions circulating in the process water, it amounts to approximately 0.2 ppm of ions such as chloride and sulfate. Since this is approximately 200 times the concentrations normally present, engineering judgement has determined that 5.0 micromho/cm is an adequate margin to allow continued operation.

The process water could potentially have deleterious effects on the PWS such as IGSCC or the formation of unwanted radioactivities which could result in increased personnel exposure. The frequency of every 12 hours has been determined to be acceptable based on extensive operating experience and engineering judgement.

(continued)

INFORMATION ONLY

BASES (continued)

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.4.8.3.2

This surveillance ensures that the process water quality is within its stated pH limits (≥ 4.1 to ≤ 7.2). These limits provide protection against PWS degradation. More specifically, the lower pH limit of 4.1 provides protection from aluminum corrosion resulting from strong acidic process water. The upper pH limit of 7.2 will ensure that the SSS gadolinium solution will be soluble and thus effective in process water for reactor scram capabilities. The frequency of every 12 hours has been determined to be acceptable based on extensive operating experience and engineering judgement.

REFERENCES

1. RTM 5072, "Moderator pH Crosschecks Using Conductivity and Nitrate Measurements," A. Gibbs, October 9, 1990. (U)
2. DPSP-59-1-6-R through DPSP-61-1-6-R, "Works Technical Monthly Progress Reports," June 1959 to June 1961. (U)
3. ANL-5001, J. E. Draley and W. E. Ruther, "Aqueous Corrosion of 2S Aluminum at Elevated Temperatures," February 1, 1963. (U)
4. HW-44996, C. Groot and R. E. Wilson, "In-Reactor Corrosion of Aluminum," February 18, 1957. (U)
5. CRMet-700, M. D. Ferrier, "Minutes of a Conference on the Corrosion of Aluminum in Water at High Temperatures," June 1957. (U)
6. TC-1-1464, DPSOX 6581, W. R. Jacobsen, "Reduced Moderator pD," July 28, 1966. (U)
7. Chenouard, J., et al., "An Investigation into the Physical and Chemical Behavior of Heavy Water in Nuclear Reactors," Proc. U. N. Intern. Conf. Peaceful Uses Atomic Energy, Geneva, 7, 404 (1958). (U)
8. DPE-2140, "Stainless Steel Information Manual for the Savannah River Plant," Section 3.6, "Stress-Corrosion Cracking," Engineering Department (1960). (U)

(continued)

BASES (continued)

INFORMATION ONLY

REFERENCES
(continued)

9. DePaul, D. J., "Corrosion and Wear Handbook for Water Cooled Reactors", Chapter 10, New York: McGraw-Hill (1957). (U)
 10. DPSP-75-1027. F. B. Longtin, "Moderator pD Limit for Ink Injection", RTR-1534, January 22, 1975. (U)
 11. WSRC-SAR-RX-K, "Savannah River Site Production Reactor Safety Analysis Report", Chapter 9.3.4, "Process Water Purification System". (U)
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B 3.4 PROCESS WATER SYSTEM

B 3.4.8.4 Process Water Quality - Impurity Limits

BASES

The presence of the following materials are undesirable in the reactor system:

- a) Metals that establish large electrochemical potentials with aluminum and stainless steel.
- b) Elements such as chlorine (or chlorides) which are corrosive to aluminum and stainless steel.

Because the mechanism by which these undesirable impurities may concentrate in the reactor system is not known, the consequences of these types of corrosive action can be minimized by limiting the amount of impurities that are allowed in the reactor system to negligible quantities.

REFERENCES

- 1. DPSP-59-1-6-R through DPSP-61-1-6-R, "Works Technical Monthly Progress Reports," June 1959 to June 1961. (Secret/RD)
- 2. ANL-5001, J. E. Draley and W. E. Ruther, "Aqueous Corrosion of 2S Aluminum at Elevated Temperatures," February 1, 1963. (U)
- 3. HW-44996, C. Groot and R. E. Wilson, "In-Reactor Corrosion of Aluminum," February 18, 1957.
- 4. CRMet-700, M. D. Ferrier, "Minutes of a Conference on the Corrosion of Aluminum in Water at High Temperatures," June 1957. (U)
- 5. TC-1-1464, DPSOX 6581, W. R. Jacobsen, "Reduced Moderator pD," July 28, 1966. (U)
- 6. Chenouard, J., et al., "An Investigation into the Physical and Chemical Behavior of Heavy Water in Nuclear Reactors," Proc. U. N. Intern. Conf. Peaceful Uses Atomic Energy, Geneva, 7, 404 (1958). (U)

(continued)

BASES

INTENTION ONLY

REFERENCES
(continued)

7. DPE-2140, "Stainless Steel Information Manual for the Savannah River Plant," Section 3.6, "Stress-Corrosion Cracking," Engineering Department (1960). (U)
8. DePaul, D. J., Corrosion and Wear Handbook for Water Cooled Reactors, Chapter 10, New York: McGraw-Hill (1957). (U)
9. DPSP-75-1027. F. B. Longton, Moderator pD Limit for Ink Injection, RTR-1534, January 22, 1975. (U)
10. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 9.3.4, Process Water Purification System (U)
11. Analytical Development Section, Analytical Operating Procedures, Volume II, Procedure Manual 16.1, A.D.S. 1500 Rev. 1

"DELETED VERSION"

Emergency Cooling System - Operating
B 3.5.1.1

B 3.5 EMERGENCY COOLING SYSTEM/EMERGENCY SHUTDOWN SYSTEMS

B 3.5.1.1 Emergency Cooling System - Operating

BASES

BACKGROUND

The Emergency Cooling System (ECS) is designed to ensure removal of decay heat from the reactor core whenever the normal heat removal system is inoperable and to provide makeup water to the core following Loss of Coolant Accidents (LOCAs). The following paragraphs describe the operation of the ECS subsystems and how each of them are expected to function when the ECS is actuated either by Manual Incident Action (MIA) or automatically by Automatic Incident Action (AIA).

The ECS is comprised of three main subsystems:

1. ECS water sources.
2. ECS water conditioning and distribution.
3. ECS water addition.

ECS water sources consist of a water supply to each of the ECS pumps, including all valves and headers up to and including the series crosstie header isolation valves. Two water supplies, the 186 basin and the river water supply lines, are available as ECS water supplies. Normally all ECS water is supplied by the 186 basin to the three dedicated ECS pumps and the cooling water pumps. Should ECS water be unavailable from the 186 basin then alternate supplies of water could be provided by the river water system or from the disassembly basin.

ECS water conditioning and distribution consists of the borated water header filter screen to trap any large debris; the filter screen bypass valves which open if the screen becomes clogged; the borated water header and recirculation system which ensure adequate nuclear poisoning of approximately the first 20,000 gallons of ECS water; and the ECS crosstie header which distributes the conditioned water. Proper operation of the ECS water conditioning and distribution subsystem ensures that water supplied by the ECS is free from debris, properly poisoned and available to the ECS water addition subsystem.

"DELETED VERSION"

(continued)

"DELETED VERSION"

Emergency Cooling System - Operating
B 3.5.1.1

BASES

ONLY

BACKGROUND
(continued)

ECS water addition consists of the four lines that are connected to process water loops [REDACTED] and [REDACTED] just prior to the reactor tank inlet plenum. Each line contains one [REDACTED] series, one [REDACTED] series valve, and a magnetic flow meter. The two valve design allows for pressurization of the ECS crosstie header without concern of an inadvertent valve opening causing an ECS water addition to the reactor.

The following discussion illustrates how the above ECS subsystems are interrelated and function as an integrated ECS and also highlights important design features.

The ECS can be actuated either automatically by AIA A and B Actions or manually by MIA from the containment console. AIA and MIA result in the same response of the ECS (assuming that AIA initiates both A and B Action). AIA is controlled by reactor process water level sensors (i.e., during a LOCA). However, MIA must be initiated by operators who examine the various alarms and reactor status panels and determine if MIA is needed (i.e., during a LOPA). The ECS responds to an AIA (A and B Action) or MIA signal as follows:

-----NOTE-----
The actions stated in items 1 through 4 below are initiated by AIA A Action at 12.8 feet decreasing reactor tank level or by MIA. The actions stated in items 5 and 6 below are initiated by AIA B Action at 10.8 feet decreasing reactor tank level or by MIA.

1. The ECS Booster Pump, ECS-A Pump, and ECS-B Pump to start to ensure ECS water is available to the [REDACTED] series crosstie header isolation valves.
2. Crosstie header isolation valves [REDACTED] open to allow full flow through the borated water header and ECS crosstie header. The borated water and ECS crosstie headers are normally pressurized by one-inch bypass lines around the [REDACTED] series crosstie header isolation valves, to verify integrity of the headers and availability of ECS water.

(continued)

"DELETED VERSION"

"DELETED VERSION"

Emergency Cooling System - Operating
B 3.5.1.1

BASES

**BACKGROUND
(continued)**

3. Motorized valve [REDACTED] which connects the booster, ECS-A and ECS-B pump outputs to cooling water header [REDACTED] receives a close signal to ensure that the normally closed valve is actually closed. This valve though normally closed, may be accidentally open which could divert the booster, ECS-A and ECS-B pump flow to cooling water header [REDACTED]. The application of a close signal upon AIA ensures that the valve closes in such a case.
4. ECS addition valves [REDACTED] open, which allows access of cooling water to the respective [REDACTED] series ECS addition valves and valve [REDACTED] switches from manual to automatic operation.
5. If the reactor tank level decreases to 10.8 feet, or MIA is initiated, all of the [REDACTED] series valves open except [REDACTED]. Emergency coolant will enter the reactor when the PW plenum pressure decreases below the ECS pressure. This design feature is incorporated to ensure that ECS injection cannot occur during normal operation with the Bingham pump AC motors running, since they maintain reactor PW plenum pressure greater than the ECS design pressure. If the reactor water level decreases to the 9 foot level, then the valve [REDACTED] will automatically open, increasing the ECS flow. Valve [REDACTED] automatically closes when the reactor water level increases to 14 feet.
6. The rotovalve interlock circuit is activated.

The first approximately 20,000 gallons of ECS water injected into the reactor are poisoned with boron which will displace the D₂O from the fuel assembly flow channels and a large majority of the D₂O in the moderator prior to the admittance of unpoisoned light water into the fuel assembly flow channels. This ensures that the reactor will not return to criticality due to the positive reactivity affects which could be caused by light water in the fuel assemblies and D₂O in the moderator.

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"DELETED VERSION"

"DELETED VERSION"

Emergency Cooling System - Operating
B 3.5.1.1

BASES (continued)

MITIGATION ONLY

**APPLICABLE
SAFETY ANALYSIS**

Two design basis accidents are believed to represent the worst case accidents which require ECS action. The first accident postulated is a Loss Of Coolant Accident (LOCA) resulting from the break in one of the reactor Process Water System (PWS) recirculation lines. The second postulated accident is a Loss Of Pumping Accident (LOPA) which is caused by a rupture of a cooling water header and subsequent flooding of the Bingham pump AC and DC motors. These two accidents (LOCA and LOPA) are considered worst case accidents and envelope all other types of accidents in relative severity and required ECS response.

LCO 3.5.1.1 identifies the essential ECS equipment required to be operational for mitigating the various types of reactor incidents considered in the operation of the SRS production reactors, including worst case LOCAs and LOPAs. The analyses of the various postulated accidents are based on certain critical assumptions:

1. During a postulated large break LOCA, the AC motors on 5 of the Bingham pumps in the unaffected loops must continue to provide circulation for at least 10 minutes.
2. During a LOPA: the primary coolant piping remains intact, MIA must be initiated manually and two ECS addition systems must be OPERABLE.
3. Offsite or emergency power is immediately available to operate the ECS components when MIA or AIA is initiated, thus allowing for ECS water flow within 25 seconds of a demand signal.
4. At least two out of four ECS addition systems are capable of providing the required ECS flow.
5. Failure of the borated water header during, or prior to ECS actuation, is not a credible event and is beyond the accident analysis.
6. During a design basis LOCA, the operator must successfully control (i.e., throttle after 10 minutes) the ECS flow to preclude flooding of the DC motors while maintaining adequate ECS flow, assuming one sump pump fails.

"DELETED VERSION"

(continued)

"DELETED VERSION"

Emergency Cooling System - Operating
B 3.5.1.1

INFORMATION ONLY

BASES (continued)

LCOs

LCO 3.5.1.1 part A ensures that the number of ECS water sources required to be OPERABLE is adequate to mitigate any design basis event while considering single failure criteria.

LCO 3.5.1.1 part B ensures that the crosstie header is always pressurized. This requirement ensures that the borated water and ECS crosstie headers are continuously pressurized through the [REDACTED] series crosstie header isolation valve one-inch bypass lines. Pressurization is monitored continuously to verify the integrity of the ECS. A decrease in ECS pressure alerts operators to possible leakage in the ECS piping and valves, or inadequate pump head and allows for corrective actions to preclude unavailability of the ECS.

LCO 3.5.1.1 part C ensures that the borated water header contains a sodium polyborate solution at a concentration and temperature that is consistent with the assumptions of the safety analysis. The ECS crosstie header delivers approximately the first 20,000 gallons of boron solution and subsequent emergency coolant from the borated water header to the four sets of ECS addition valves. The safety analysis also assumes that the first approximately 20,000 gallons of the proper concentration of boron solution will be injected into the reactor prior to the admission of unpoisoned light water.

LCO 3.5.1.1 part D ensures that four ECS addition systems, each consisting of two ECS addition valves (one [REDACTED] series and one [REDACTED] series) are OPERABLE. This requirement ensures that several important assumptions are met relative to ECS addition system OPERABILITY. For a design basis LOCA it is assumed that one of the ECS addition systems will be disabled by the LOCA, one other ECS addition system will fail independently and that two ECS addition systems will function to deliver emergency coolant to the reactor. For the design basis LOPA it is known that addition system four will not function because reactor tank level does not decrease, one other addition system will fail independently and that two ECS addition systems will function to deliver emergency coolant to the reactor.

(continued)

"DELETED VERSION"

BASES

MITIGATION ONLY

LCOs
(continued)

LCO 3.5.1.1 part E ensures that the rotovalves (21-A, 21-B, 51-A and 51-B), will operate properly during a design basis LOPA (LOCA analysis assumes Bingham pumps are running, thus the rotovalve interlock must not close the rotovalves) to prevent backflow of ECS water through the Bingham pumps, and subsequent reduction in effective ECS flow. The reactor operating limits for a postulated LOPA, are based on the assumption that two ECS addition systems will function properly and that ECS flow will be directed through the proper flow paths, by ensuring automatic closure of the above rotovalves during MIA. No credit is taken in the LOPA analysis of reactor operating limits for manual closing of any of the rotovalves.

APPLICABILITY

The ECS subsystems given in LCO 3.5.1.1 are required to be OPERABLE during REACTOR STARTUP, REACTOR OPERATION, and REACTOR SHUTDOWN with shutdown time less than 3 hours, to ensure removal of decay heat from the reactor core whenever the normal heat removal system is inoperable and to provide makeup water during a LOCA. Requirements for ECS OPERABILITY after the reactor has been shutdown for greater than 3 hours are addressed in LCO 3.5.1.2.

ACTIONS

The Required Actions for each inoperable component within an ECS subsystem are supported by the reliability of the system experienced to-date during operation at SRS, and by the numbers and types of redundant components within a subsystem that could be called upon to mitigate a postulated reactor incident that could challenge the integrity of the ECS. Probabilistic and deterministic data specific to SRS failure rates of ECS subsystem components, and engineering judgement, were used to justify continued operation of the reactor while performing the Required Actions identified for each Condition outside LCO 3.5.1.1 requirements.

(continued)

BASES

INFORMATION ONLY

ACTIONS
(continued)

A.1. A.2. B.1 and B.2

Required Actions A.1 and B.1 require verification that alternate emergency-powered ECS pumping sources are OPERABLE within 1 hour, to ensure availability of at least two ECS water sources. This ensures ECS availability even in the event of loss of offsite power. This verification includes checking maintenance, surveillance and work order logs to ensure that at least two ECS water sources and associated valves and headers are not undergoing testing or maintenance that would preclude their use as an emergency ECS pumping sources. Additional checks should include verification of ECS water source OPERABILITY by checking status indication lights and gages in the control room for each associated valve, header, and pumping source as specified in Table 3.5.1.1-1. The specification as written would allow a partial loss of function if both Condition A and B were entered at the same time. This is considered acceptable since Required Actions A.1 and B.1 require that the ECS function (excluding single failure criteria) be restored within 1 hour. Required Actions A.2 and B.2 require full restoration of the system (i.e. the capability of the system to withstand a single failure) within 24 hours. If the Required Actions of Condition A or are not met within the allowed outage times then Condition G requires that the reactor be shutdown.

C.1

Condition C addresses improper boron solution concentration and/or temperature (too high or low). Although boron solution concentration, and temperature are important to proper ECS OPERATION, there is significant safety margin built-in (i.e. factor of 3 for borated water concentration) to the boron solution concentration. The temperature (i.e., 5 °C minimum, 40 °C maximum) requirements should not be exceeded since historical data shows minimum and maximum temperatures significantly above and below the allowed values used in the safety analysis. Even if the boron solution temperature and/or concentration limits were exceeded for the short 24 hour outage period, it is extremely unlikely that an accident requiring any ECS action would be required. The allowed outage time is based on engineering judgement and should allow the out-of-limit condition to be corrected within the 24 hour time period, unless significant maintenance or boron adjustments are required.

(continued)

"DELETED VERSION"

Emergency Cooling System - Operating
B 3.5.1.1

BASES

ACTIONS

C.1 (continued)

If the boron solution and/or temperature cannot be restored within allowable limits in 24 hours, then the requirements of condition G must be met.

D.1

Condition D addresses inadequate ECS crosstie header pressure. This condition can result from numerous causes such as improperly valved ECS water sources, leaking or clogged [REDACTED] series crosstie header isolation valve bypass lines, leaking ECS crosstie header, inadequate pump head, faulty pressure sensors, or others.

The cause of the low ECS crosstie header pressure must be identified and corrected within 30 minutes or the requirements of Condition G satisfied. The 30 minute allowed outage time is based on engineering judgement and should not jeopardize emergency core cooling capability of the ECS since the time of exposure to the degraded condition is extremely limited and the likelihood of requiring ECS action during the allowed outage period is small.

E.1 and E.2

Condition E addresses the inoperability of one or both ECS addition valves [REDACTED] and/or [REDACTED] series) constituting a single ECS addition system. This condition results in less ECS addition system redundancy and eliminates the single failure criteria assumed in the safety analysis but does not significantly reduce the effectiveness of the ECS for the vast majority of postulated reactor incidents. Since the ECS is degraded below requirements of LCO 3.5.1.1, a time limit of 72 hours is placed on REACTOR OPERATION.

Verification must be completed within 1 hour and every 8 hours thereafter, that the 3 remaining ECS addition systems are OPERABLE. This verification should include the checking of maintenance, repair and work order logs to ensure that the subject valves have been properly surveilled and are not out of service for testing or maintenance activities. Additional verification should be conducted to ensure that the remaining valve indications on the EEM show that all of the remaining OPERABLE ECS addition valves are properly energized and in their closed positions. The inoperable ECS addition system must be restored to OPERABLE status within 72 hours, or the requirements of Condition G met.

(continued)

"DELETED VERSION"

INFORMATION ONLY

BASES

ACTIONS

E.1 and E.2 (continued)

The 72 hour time limit is based on engineering judgement and industry standards for similar types of standby systems where availability is critical to emergency core cooling but where significant redundancy is available and the likelihood of a postulated DBA coincident with the allowed outage time is small.

F.1

Condition F addresses inoperability of the rotovalves which are required to operate during a postulated LOPA to prevent backflow of ECS water through the Bingham pumps. If the inoperable rotovalve(s) cannot be restored to OPERABLE status in 72 hours, then the requirements of Condition G must be satisfied. The allowed outage time (engineering judgement) will not significantly impact ECS ability to mitigate a LOPA since the exposure of the reactor to the degraded condition is limited, and the likelihood of requiring the ECS to mitigate a design basis LOPA coincident with the outage is small.

G.1 and G.2

Condition G addresses not meeting the Required Actions of the previous Conditions and indicates continuing inoperability of ECS subsystem components. If in REACTOR STARTUP or REACTOR OPERATION, then the reactor must be shut down within 90 minutes while actions continue to restore the required ECS system complement for the existing REACTOR STATE.

SURVEILLANCE REQUIREMENTS

The ECS is designed to permit inspection, testing, and in some cases, continuous monitoring of critical ECS subsystem components. Special provisions (e.g., 90 degree elbows to direct flow from the ECS addition systems to the cooling water effluent headers) have been made, which allow for a functional full flow test of the ECS up to the point of addition to the reactor. All of the surveillance requirements that follow, are directed toward demonstrating ECS OPERABILITY and ensuring reliability of this important engineered safety feature. The requirements demonstrate ECS OPERABILITY by verifying the characteristics of all of the ECS subsystem components are as designed, and will be OPERABLE should the ECS be needed. Periodic component tests are supplemented by extensive functional tests during scheduled outages.

(continued)

"DELETED VERSION"

Emergency Cooling System - Operating
B 3.5.1.1

BASES

1.

ONLY

**SURVEILLANCE
REQUIREMENTS
(continued)**

SURVEILLANCE FREQUENCY

In general, surveillance requirement frequencies are based on engineering judgement and industry accepted practice considering plant conditions required to perform the test, the ease of performing the test and the likelihood of a change in ECS or individual subsystem component status.

Additional consideration is given to components based on past operating history, reliability, and where applicable, on probabilistic data that indicates a component is a large contributor to potential ECS failure.

SR 3.5.1.1.1

This surveillance is performed once per 12 hours to ensure that all of the ECS valves are properly positioned and energized. Performance of the surveillance includes visual verification of each ECS valve status on the EEM for proper position, energization and status (i.e., manual or automatic mode). The 12 hour surveillance frequency ensures that a valve in the incorrect position or mode, will be detected, identified and corrected, thus ensuring maximum availability of the ECS-related valves. Fault tree analysis conducted on the ECS identifies failure of the ECS addition valves to allow ECS water flow as a major contributor to ECS inoperability. Performance of this surveillance requirement will ensure that the valves are energized and available to the ECS and thus not exacerbate the probability of valve inoperability.

SR 3.5.1.1.2

This surveillance requirement ensures that the ECS piping and valves downstream from the [REDACTED] series crosstie header isolation valves are OPERABLE and that the ECS water pumping source is providing adequate head. Low ECS crosstie header pressure alerts the reactor operator to a potential problem with the ECS valves, piping or pumping sources. ECS crosstie header pressure must be checked at least once every 12 hours and is essential to verifying ECS integrity by allowing rapid identification of inadequate ECS pressure.

(continued)

"DELETED VERSION"

"DELETED VERSION"

Emergency Cooling System - Operating
B.3.5.1.1

BASES

SURVEILLANCE

SR 3.5.1.1.3

This surveillance requirement ensures that adequate ECS water is available from each individual Cooling Water (CW) header [REDACTED] during REACTOR OPERATION. If the cooling water header is not maintained at the required header pressure when all five of the 190 building pumps are operating, then an alternate ECS source must be established until cooling water header pressure is restored. All five of the dedicated 190 building cooling water pump(s) for each header [REDACTED] must be operating (during REACTOR OPERATION) whenever a cooling water header is used as pressurized ECS source, to ensure adequate ECS water capacity.

SR 3.5.1.1.4

This surveillance requirement demonstrates that all indications of Booster Pump OPERABILITY in the control room are normal. Booster Pump availability is crucial to proper ECS function during a loss of offsite power and this verification ensures that operations personnel are frequently (once every 12 hours) checking Booster Pump availability in addition to verifying the absence of automatic alarms from the EEM associated with Booster Pump inoperability.

SR 3.5.1.1.5

This surveillance demonstrates that the borated water header concentration and temperature are within required limits. Particular attention to boron solution temperature should be provided when ambient temperatures are below 5°C since boron solubility decreases with temperature. The maximum temperature allowed for the boron solution and ECS water is essential to ECS ability to effectively remove residual heat from the core since the higher the temperature of the ECS water, the less heat the ECS can remove from the reactor core for a given ECS volume. The maximum borated water solution temperature and ECS water temperature is assumed when calculating maximum reactor operating power limits and should not be exceeded.

(continued)

"DELETED VERSION"

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.5.1.1.6

This surveillance verifies that the manual block addition valves are maintained in the fully open position. Personnel must physically verify that the valve stem is in the fully open position prior to locking the -20' and -40' radiologically controlled areas.

SR 3.5.1.1.7

This surveillance demonstrates the OPERABILITY of the Booster Pump, ECS-A and ECS-B pumps. Each pump should be started at least once per 14 days and continue to run for at least 10 minutes to demonstrate OPERABILITY of the pumps. The diesel-powered Booster Pump auxiliaries should be checked after the surveillance to ensure that the standby status (e.g., fuel supply, lube oil etc.) is restored to pretest levels.

SR 3.5.1.1.8

This surveillance demonstrates that the boron₁₀ concentration and boron weight % in the boron solution is maintained within required limits. The results of the chemical sampling and analysis should verify that the conductivity measurements of the boron solution are an accurate representation of boron weight % in the boron solution. If significant contamination of the boron solution were to occur, then the conductivity measurements might not provide accurate data relative to actual boron weight % in the boron solution. Although contamination of the borated water header is not anticipated, the boron solution must be sampled and analyzed every 24 hours if conductivity is out of specification, and within 24 hours following addition of water to the storage header, and at least every month.

SR 3.5.1.1.9

This surveillance demonstrates that the counter-weighted borated water header filter screen bypass valves are free, and properly balance, to allow flow around the filter screen only when the filter screen pressure drop exceeds 7 to 7.5 psid. An applied torque to the external valve actuator and subsequent valve opening, demonstrates that the bypass valves are not binding or frozen. Care must be used to verify that the valve returns to its normal, completely closed, position (i.e., external actuator return to its original position).

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.5.1.1.10

This surveillance demonstrates that the ECS-A and ECS-B pumps can be started and run from the emergency buses and containment substation if required. The electric-powered ECS-A and ECS-B pumps should be run for at least 30 minutes from the emergency buses and then switched to the containment substation buses and run an additional 30 minutes. The electric-powered ECS-A and ECS-B pumps must be powered from the emergency buses to demonstrate capability of the pumps to operate as ECS emergency pumping sources during a loss of offsite power.

SR 3.5.1.1.11

This surveillance demonstrates functional full flow OPERABILITY of the ECS as an integrated system. The ECS is placed in a standby condition similar to the condition that exists during REACTOR OPERATION. The system is then activated by MIA and the system valves, pumps, and instrumentation are monitored for proper operation. The primary success criteria is that total flow to the reactor (through the test lines) is monitored and verified as $\geq 11,945$ gpm in ≤ 25 seconds. Additional checks should verify that filter screen measured pressure drop is ≤ 7 psid and that valve properly cycles on reactor tank levels of 9 feet and 14 feet.

SR 3.5.1.1.12 through SR 3.5.1.1.16

These surveillances verify that performance of each individual ECS source has not degraded to an unacceptable level (i.e. the flow is $\geq$ to the value stated for that source).

SR 3.5.1.1.17

This surveillance verifies that the Booster Pump when started can achieve and maintain the proper engine speed for at least one hour, and can be successfully restarted within 60 seconds of shutdown while the diesel is hot. This surveillance is representative of the way the Booster Pump is anticipated to be operated during emergency conditions. Procedures require that the Booster Pump run for at least one hour after reactor shutdown and then be shutdown as soon as practical after MIA to conserve diesel fuel and maintain the Booster Pump in a standby configuration. The Booster Pump must be capable of

(continued)

BASES

INFORMATION ONLY

**SURVEILLANCE
REQUIREMENTS**

SR 3.5.1.1.17 (continued)

restarting immediately following shutdown to demonstrate OPERABILITY of the Booster Pump diesel when starting from a hot condition.

SR 3.5.1.1.18

This surveillance ensures that the rotovalves will function as required during a LOPA to prevent backflow through the PWS Bingham pumps.

SR 3.5.1.1.19

This surveillance ensures that the ECS sources and the valves associated with those sources will function as required to provide the minimum flow in the required time to mitigate the LOPA event.

SR 3.5.1.1.20

This surveillance verifies that the manual suction valve lineup to the ECS pumps is correct for normal operation.

REFERENCES

1. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 15, "Accident Analyses." (U)
2. DPST-87-100, "An Action Plan to Demonstrate The Adequacy of The SRS Reactor System Response to a Hypothetical Severe Loss-Of-Coolant Accident", Appendix B, July 22, 1987. (U)
3. DPST-87-679, "Revision 1 - Probabilistic Analysis of The Initial Response of The L Reactor ECCS And Water Disposal System to a Guillotine Break in a Below Grade Reactor Inlet Pipe", January 11, 1988, page 5. (U)
4. DPST-87-332, "Response of The Emergency Core Cooling System to a Design Basis Loss of Pumping Accident", June 19, 1987. (U)
5. DPST-77-240, "Reliability of The Reactor Emergency Cooling System", March 16, 1977. (UCNI)

(continued)

BASES

EXPLANATION ONLY

REFERENCES
(continued)

6. 10CFR50, General Design Criteria 17 and 35, "Electric Power Systems" and "Emergency Core Cooling", January 1, 1986.
7. DPST-87-506, "P-10 Mark 22 Emergency Cooling System Power Limits (LOPA) Using the Double Pressure Plate Bottom Fitting Insert", June 25, 1987. (U)
8. I.O.M., "Net Positive Suction Head for Booster Pump", J.H. Hinton, November 11, 1987. (U)
9. RTM-4884, "Emergency Cooling System Polybor Limits", December 8, 1987. (U)
10. DPST-88-418, "Flow Instability Analysis of a Design Basis LOCA For The L-3.3 Subcycle, March, 1988. (U)
11. NUREG - 0452, 3/4.5 "Westinghouse Standard Technical Specification, Emergency Core Cooling System" (U)

"DELETED VERSION"

Emergency Cooling System - Shutdown
B 3.5.1.2

3.5 EMERGENCY COOLING SYSTEMS/EMERGENCY SHUTDOWN SYSTEMS

B 3.5.1.2 Emergency Cooling System - Shutdown

BASES

BACKGROUND See the 3.5.1.1 Bases for the Background discussion of Emergency Cooling System (ECS).

APPLICABLE SAFETY ANALYSIS See the 3.5.1.1 Bases for the discussion of the ECS.

LCOs The 3.5.1.1 Bases discussion of LCOs is applicable except that only two (versus three) ECS water sources specified in Table 3.5.1.2-1 is required to be OPERABLE and only two (versus four) sets (one of which must be system ~~or~~ of ECS addition systems are required to be OPERABLE, due to the reduced cooling requirements in REACTOR SHUTDOWN with shutdown time greater than 3 hours.

APPLICABILITY The emergency cooling system subsystems given in LCO 3.5.1.2 are required to be OPERABLE during REACTOR SHUTDOWN with shutdown time greater than 3 hours, to ensure removal of residual heat from the reactor core, whenever the normal heat removal system is inoperable. The ECS must remain OPERABLE until all HGAs have been removed from the reactor.

ACTIONS

A.1 and A.2

With less than two required ECS water sources OPERABLE, Action A.1 specifies that CHARGE AND DISCHARGE OPERATIONS be suspended IMMEDIATELY since these operations could lead to a LOCA from a dropped assembly. Required Action A.2 specifies that the required number of sources be restored within 72 hours. If the required number of sources cannot be restored then the LCO has been violated and the results must be reported in accordance with the Administrative Procedures.

(continued)

BASES

ONLY

ACTIONS
(continued)

B.1

With inadequate crosstie header pressure, Action B.1 requires the header pressure to be restored to within limits within 30 minutes. This condition can result from numerous causes such as improperly valved ECS water sources, leaking or clogged series crosstie header isolation valve bypass lines, leaking ECS crosstie header, inadequate pump head, faulty pressure sensors, or others.

The cause of the low ECS crosstie header pressure must be identified and corrected within 30 minutes or the LCO has been violated and the results must be reported in accordance with the Administrative Procedures. The 30 minute allowed outage time is based on engineering judgement and should not jeopardize emergency core cooling capability of the ECS since the time of exposure to the degraded condition is extremely limited and the likelihood of requiring ECS action during the allowed outage period is small.

C.1 and C.2

With one or more of the rotovalves inoperable, Required Actions C.1 and C.2 require that the rotovalve(s) be restored within 72 hours or that the affected rotovalves are closed and deenergized. Proper function of the rotovalves assures that the ECS flow will be directed through the proper flow paths, providing full effectiveness of the cooling flow. The 72 hour Completion Time is considered to be acceptable, since the system is still functional, but in a degraded condition. Considering that the specification Applicability is REACTOR SHUTDOWN, with shutdown time greater than 3 hours, the heat removal requirements are reduced from that required during REACTOR OPERATION; the likelihood of an accident requiring the full capability of the system during the allowed outage time is small.

D.1 and D.2

With less than two of the required addition systems OPERABLE, Action D.1 specifies that CHARGE AND DISCHARGE OPERATIONS be suspended IMMEDIATELY since these operations could lead to a LOCA from a dropped assembly. Required Action D.2 specifies that the required number of addition systems be restored within 72 hours. If the required number of addition systems cannot be restored then the LCO has been violated and the results must be reported in accordance with the Administrative Procedures.

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

The 72 hour Completion Time is considered to be acceptable based on the capability of the remaining addition system to supply adequate cooling flow for the shutdown condition. However, since only one addition system is OPERABLE and a single failure could result in the loss of the associated ECS water source, the inoperable addition system must be restored within 72 hours.

E.1

With the boron solution concentration and/or temperature outside the required limits, Action E.1 specifies restoration of the concentration and/or temperature within 96 hours. Although boron solution concentration, and temperature are important to proper ECS operation, there is significant safety margin included in the boron solution concentration requirement (i.e., factor of 3 for the worst case accident during REACTOR OPERATION) (Ref. 11). A low solution temperature ($< 5^{\circ}\text{C}$) is only significant if it reduces the solution concentration. A high solution temperature ($> 40^{\circ}\text{C}$) reduces the heat removal capability of the solution. These limits are based on the requirements for a worst case accident during REACTOR OPERATION. The 72 hour Completion Time is considered to be acceptable, since the system is still functional, but in a degraded condition. Considering that the specification Applicability is REACTOR SHUTDOWN, with shutdown time greater than 3 hours, the heat removal requirements are reduced from that required during REACTOR OPERATION; the likelihood of an accident requiring the full capability of the system during the allowed outage time is small.

SURVEILLANCE
REQUIREMENTS

SR 3.5.1.2.1

This surveillance is performed once per 12 hours to ensure that the required ECS valves are properly positioned and energized. Performance of the surveillance includes visual verification of each ECS valve status on the EEM for proper position, energization and status (i.e., manual or automatic mode). The 12 hour surveillance frequency ensures that a valve in the incorrect position or mode, will be detected, identified and corrected, thus ensuring maximum availability of the ECS-related valves.

(continued)

BASES

**SURVEILLANCE
REQUIREMENTS**

SR 3.5.1.2.1 (continued)

Fault tree analysis conducted on the ECS identifies failure of the ECS addition valves to allow ECS water flow as a major contributor to ECS inoperability. Performance of this surveillance requirement will ensure that the valves are energized and available to the ECS and thus not exacerbate the probability of valve inoperability.

SR 3.5.1.2.2

This surveillance requirement demonstrates that all indications of Booster Pump OPERABILITY in the control room are normal. Booster Pump availability is crucial to proper ECS function during a loss of offsite power and this verification ensures that operations personnel are frequently (once every 12 hours) checking Booster Pump availability in addition to verifying the absence of automatic alarms from the EEM associated with Booster Pump inoperability.

SR 3.5.1.2.3

This surveillance demonstrates that the borated water header concentration and temperature is within required limits. Particular attention to boron solution temperature should be provided when ambient temperatures are below 5°C (Ref. 14) since boron solubility decreases with temperature. The maximum temperature allowed for the boron solution and ECS water is essential to ECS ability to effectively remove residual heat from the core since the higher the temperature of the ECS water, the less heat the ECS can remove from the reactor core for a given ECS volume. The maximum borated water solution temperature and ECS water temperature is assumed when calculating maximum reactor operating power limits and should not be exceeded.

SR 3.5.1.2.4

This surveillance verifies that the manual block addition valves are maintained in the fully open position. Personnel must physically verify that the valve stem is in the full open position once every 12 hours.

(continued)

"DELETED VERSION"

Emergency Cooling System - Shutdown
B 3.5.1.2

ONLY

BASES

**SURVEILLANCE
REQUIREMENTS
(continued)**

SR 3.5.1.2.5

This surveillance requirement ensures that the ECS piping and valves downstream from the [REDACTED] series crosstie header isolation valves full of water to ensure that the ECS water pumping sources can provide flow within the required time. This surveillance is only required once every 12 hours when [REDACTED] are being used as an ECS source.

SR 3.5.1.2.6

See Bases B 3.5.1.1 discussion of surveillance SR 3.5.1.1.2.

SR 3.5.1.2.7 through SR 3.5.1.2.10

See Bases B 3.5.1.1 discussion of surveillance SR 3.5.1.1.7 through SR 3.5.1.1.10.

SR 3.5.1.2.11 through SR 3.5.1.2.17

See Bases B 3.5.1.1 discussion of surveillance SR 3.5.1.1.12 through SR 3.5.1.1.18.

SR 3.5.1.2.18

See Bases B 3.5.1.1 discussion of surveillance SR 3.5.1.1.20.

REFERENCES

See B 3.5.1.1 for list of References.

"DELETED VERSION"

B 3.5 EMERGENCY COOLING SYSTEMS/EMERGENCY SHUTDOWN SYSTEMS

B 3.5.2.1 Supplementary Safety System - Operating

BASES

BACKGROUND

The Supplementary Safety System (SSS) is designed to shut down the reactor, if the safety rods fail to insert upon receipt of a scram signal. The reactor is shut down by injecting a potent neutron absorber (poison) consisting of a solution of gadolinium and process water (D_2O). A portion of the injection is via spargers located in the center of the reactor core with the remaining injection going directly into the suction lines of the Bingham pumps. The rate of injection and the duration of the injection are controlled to both the spargers and pump suction lines to ensure that sufficient gadolinium is injected and evenly dispersed in the reactor to maintain the reactor shutdown even following temperature decrease and xenon decay.

In addition to shutting down the reactor automatically in the event of a transient accompanied by scram failure, the SSS can be manually actuated to shut down the reactor should the automatic actuation via the explosive valves or seismic triggers become disabled.

The following paragraphs describe the SSS components and how each contributes to SSS OPERABILITY. Additional discussion is included relative to modifications which have been performed on SSS components and the supporting documentation which ensures that SSS OPERABILITY as benchmarked against the 1975 C-area test has been preserved.

Each side of the SSS contains the necessary piping, valves, instrumentation and controls to perform its design basis function. The major components of the system and their functions are described below (Ref. 6, 7, 8, 9 and 21):

GADOLINIUM STORAGE TANKS

There are two gadolinium storage tanks per side of the SSS. One of the tanks supplies three sparger injection lines and the other tank supplies three pump suction injection lines. Each tank is filled with a given volume of liquid and is pressurized internally with nitrogen.

(continued)

BASES

INFORMATION ONLY

BACKGROUND

GADOLINIUM STORAGE TANKS (continued)

The liquid is a solution of gadolinium and process water (D_2O). Limits are placed on both the liquid and gas volumes to ensure proper operation of the system.

EXPLOSIVE VALVES AND ASSOCIATED FIRING CIRCUITS

Each ABS/SC explosive valve has a dedicated firing circuit which can permit firing by allowing current to flow from the nicad batteries (see description of 24 volt batteries below) through the explosive valve primer, which detonates and opens the valve. The ABS/SC explosive valve firing circuit can be actuated automatically via the safety computer, or manually from the External Fission Counter (EFC) or Nuclear Console (NC).

Although the ABS/SC explosive valve in each side can be actuated manually from the EFC or NC, manual action is not relied on for mitigation of any accidents identified in the safety analysis.

EXPLOSIVE VALVE 24 BATTERIES AND CHARGER

Each explosive valve has two 24 volt batteries connected such that either battery can detonate the primer in each explosive valve. A current of at least five amperes applied to the primer for approximately eight msec is required to ensure detonation of the valve primer. The batteries have a capacity of one ampere-hour each. Reference 8 indicates that the condition of batteries is difficult to assess in the unloaded standby condition germane to the SSS. A battery charger is connected across the explosive valve batteries to maintain the necessary charge. The voltage measured across a 24 volt battery bank of ≥ 24 volts with the charger disconnected indicates sufficient battery capacity to detonate a primer.

(continued)

BASES

INFORMATION ONLY

BACKGROUND
(continued)

SPARGER INJECTION LINES

The sparger injection lines act as a header to distribute the gadolinium solution released from one of two storage tanks to the three spargers associated with each SSS side. The injection lines are connected from the low pressure end of the SSS injection valves (1 explosive and 1 pneumatic) to the block valves located upstream of the three spargers.

The injection lines are kept full of the gadolinium solution so that the SSS will begin injecting into the bulk moderator reactor in ≤ 0.88 second and continue injecting for ≥ 35 seconds. Calculations (Refs. 12, 13 and 14) indicate that the quantity of trapped air in the injection lines must be controlled to ensure that the SSS will maintain its capability to inject the gadolinium solution within the time necessary to reduce reactor power as established in the 1975 C-area test. Prior to modifying the injection line slope in 1986, significant quantities (i.e., > 1050 cc) of air could be trapped in the injection lines. The SSS injection lines were resloped to provide positive or zero slopes and thus reduce the potential for trapping unwanted air, particularly when refilling the lines.

Concurrent with the resloping of injection lines, a modification was made which allows for remote filling of injection line probe pots (Ref. 9). The remote fill capability enables the injection lines to be filled without shutting down the reactor if an abnormal (low) probe pot level is detected. The probe pots are used to indicate if the injection lines are full by measuring the presence of SSS gadolinium contained in the probe pots and providing a "Probe Pot Line Level" light at the SSS panel in the -40 Pump Observation Room (POR) when the pots are properly filled.

Each injection line and probe pot contains a leak collection and vent line respectively. The leak collection and vent lines contain orifices to prevent bypass of SSS gadolinium solution flow beyond normal leakage should the SSS be actuated, while allowing for rapid movement of the gadolinium solution through the SSS injection lines.

(continued)

BASES

INFORMATION ONLY

BACKGROUND

SPARGER INJECTION LINES (continued)

An inadvertent bypass of the injection line level control system orifice could be caused by failure to ensure that all injection line fill/drain/vent connections are double isolated.

PNEUMATIC VALVE AND NITROGEN SURGE TANK

A pneumatic valve and associated surge tank are provided to permit manual actuation of the SSS from the pull ring on the SSS panel in the Central Control Room (CCR) or automatic actuation by the Seismic Trigger Subsystem described below. The pneumatic valve is a "Gas-To-Close" valve. The surge tank provides a reservoir of nitrogen available to maintain the pneumatic valve at approximately 25 psig. The pull ring actuates the SSS by engaging a disc cutter which punctures a rupture disc. Once punctured, the hole in the disc vents the nitrogen to atmosphere, opening the pneumatic valve.

Even though the pneumatic valve provides for manual actuation of the SSS, there is no accident analyzed in the SRS safety analysis which relies on manual actions for mitigation during REACTOR OPERATION. However, control room fires and other events can be postulated which would disable the explosive valves and associated firing circuits. The manual actuation capability using the pneumatic valves is included in this Technical Specification for prudence. Automatic actuation for seismic events is provided by the Seismic Switch Trigger Subsystem described below.

SEISMIC TRIGGER SUBSYSTEM

Automatic actuation of the SSS for seismic events is accomplished by the use of seismic triggers, relays, solenoid valves, piping, manual valves, wiring, and alarms (Ref. 24). Methodology developed by the Seismic Qualification Utility Group (Ref. 28) addressed seismic concerns in older nuclear plants (A-46) (Ref. 29). This approach of using real earthquake and generic test data is an NRC approved alternate to the formal seismic

(continued)

BASES

BACKGROUND

SEISMIC TRIGGER SUBSYSTEM (continued)

qualification of equipment by tests or analysis (Ref. 28). The solenoid and manual valves were selected and installed using this methodology to ensure seismic adequacy. Either generic test data or shake table tests were used to seismologically qualify the bypass and system actuation alarm panels. The seismic triggers were qualified by shake table tests (Ref. 27) and the piping and supports were qualified by analysis (Ref. 27). The system is designed to actuate during an earthquake that generates a horizontal or vertical acceleration greater than .02g in the -40 foot area of the reactor building. Each relay is designed to independently open the normally held closed solenoid valves and vent the pressurized nitrogen system that holds the pneumatic valves closed. Piping that connects the solenoid valves to the nitrogen closing system are attached to the reactor building so that this piping will behave in a rigid manner during earthquakes and not crimp closed.

Two seismic trigger actuation systems (one each for both the near side and far side SSS) are provided for redundancy. The two redundant seismic triggers are mounted on a common concrete pad and instrument rack. The actuation of the near side SSS is physically and electrically separated from the far side SSS to maintain the present redundant feature of the overall system.

The system design is fail-safe for loss of power source to the system. The solenoid valves are held closed with power supplied from the "IR" Panel Uninterruptable Power Supply (UPS). Loss of that power source, during either a seismic event or normal operation, will actuate the SSS.

The system is equipped with a bypass feature and safe-to-unbypass indication to allow maintenance and testing of the seismic triggers without actuating the SSS. Manual block valves upstream of the solenoid valves provide isolation that will prevent any spurious actuation of the SSS when the system is not required to be OPERABLE.

A CCR miscellaneous annunciator alarm is provided to alert operating personnel if either SSS seismic trigger actuation system is bypassed or if the manual block valves are not fully open. Also, two redundant seismically qualified alarms are provided in the CCR to indicate that the associated SSS seismic trigger system has actuated.

(continued)

INFORMATION ONLY

BASES

BACKGROUND (continued)

MANUAL BLOCK VALVES, BALL CHECK VALVES AND SPARGERS

Two manual block valves are located between each injection line and associated sparger, and are used to facilitate maintenance activities on the check valves, or for isolation of a sparger. The block valves are monitored by indicating lights in the CCR and by limit switches which alarm in the CCR if the valves are closed greater than one turn from the fully opened position.

A ball check valve is located below each sparger to prevent the reverse flow of moderator from the bottom of the reactor into the SSS injection lines while allowing a readily accessible path for injection of SSS gadolinium solution into the spargers. Leaky check valves can be detected by monitoring sight glasses that lead to a leak collection tank, and by monitoring the level in the leakage collection tank.

The spargers are long tubes which protrude through the bottom of the reactor upward into the center of the reactor core (spargers are located 30 inches from the center of the core with each side having 120 degree symmetry) and are designed to disperse the gadolinium solution into the moderator. The gadolinium solution exits through holes at seven elevations along the length of the sparger, with the first set of sparger holes located approximately two feet above the tank bottom.

PUMP SUCTION INJECTION

Each side of the SSS contains a storage tank that is connected through a pneumatic valve to three of the PWS loops just prior to the suction of the Bingham pumps. One side connects to loops 2, 4, and 6 and the other side connects to loops 1, 3, and 5. The pneumatic valve is normally held closed by nitrogen and opens when the nitrogen is vented through a solenoid operated vent valve. The solenoid valve is actuated when the pressure in 2 out of 3 sparger injection lines on the same side is ≥ 75 to 90 psig. The liquid and gas volumes in the tank used for pump suction injection are controlled such that the rate of injection and duration of injection of the gadolinium solution are within the limits assumed in the safety analysis.

(continued)

INFORMATION

BASES (continued)

APPLICABLE SAFETY ANALYSES

The SRS safety analysis assumes that the SSS will mitigate selected accidents, without safety rod scram, by initiating SSS action by Automatic Backup Shutdown logic which resides in the Safety Computer (ABS/SC) or via the Seismic Trigger Subsystem. The design basis for the SSS is to maintain the integrity of the Airborne Activity Confinement System (AACS) during transients that are not terminated by safety rod scram. The AACS integrity will be preserved provided that the SSS terminates the transient before steam-generated forces within the reactor tank generate sufficient force to lift the plenum.

The design basis for the Seismic Trigger Subsystem is that the SSS be actuated by the Seismic Trigger Subsystem at least as rapidly as the ABS/SC actuation of the SSS following the loss of AC power to the Bingham pumps. If the postulated seismic event causes a loss of offsite power which causes the Bingham pump AC motors to trip and prevents the safety rods from entering the core following the scram signal, the SSS activation by the Seismic Trigger Subsystem will provide the required protection.

A test conducted in C-area in 1975 in which the SSS was actuated with the reactor at 60% power showed that reactor power began to decrease 0.70 second after the SSS was actuated. During the test, the decrease in reactor power versus time was measured using special instrumentation to detect and measure neutron and poison levels and distributions (Refs. 1 and 4). The 1975 SSS test is of extreme importance since the results of the test are used to benchmark SSS performance and anticipated reactor response to the gadolinium solution during an accident or transient (Ref. 5).

Proper operation of the SSS ensures that Confinement Protection Limits are met to protect the Airborne Activity Confinement System (AACS), if the safety rods should fail to insert to stop certain postulated reactor transients (Ref. 2). The Confinement Protection Limits are derived assuming that the ABS/SC logic, which resides in the Safety Computers (SC), will actuate the SSS ABS/SC explosive valves to inject the gadolinium solution if the safety rods fail to shut down the reactor. Calculations used to establish Confinement Protection Limits are based on k_{eff} versus time data for the reactor which were generated during the 1975 C-area test of the SSS. Operation within the calculated Confinement Protection Limits ensures that

(continued)

BASES

INFORMATION ONLY

**APPLICABLE
SAFETY ANALYSES
(continued)**

steam formation during a postulated reactor accident without expected safety rod scram will be limited such that the plenum will not lift, and thus not cause an overheating or over pressurization of the AACS.

Surveillance requirements on SSS process variables and component OPERABILITY ensure that the SSS is in a condition to permit mitigation of any design basis event considered in the safety analysis in conjunction with an active single failure.

LCO

The safety analysis assumes that both sides of the SSS are OPERABLE to ensure that even in the event of a single active failure on one side, that the other side of the SSS will mitigate any design basis event. Satisfactory performance of the surveillance requirements within their specified frequencies ensures the OPERABILITY of the SSS.

APPLICABILITY

Both sides of the SSS are required to be OPERABLE prior to REACTOR STARTUP and during REACTOR OPERATION. These requirements ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the reactor shutdown during the resulting temperature decrease, and subsequent xenon decay during the range of postulated transients or accidents (except LOPA or large-break LOCA).

ACTIONS

A.1

With one side of the SSS inoperable, the Required Action specifies restoration of the inoperable system within 72 hours. Under this degraded condition the SSS cannot withstand a single failure. The Completion Time is considered to be acceptable based on the OPERABILITY of the safety rod system, which is the primary means of emergency shutdown.

(continued)

BASES

CONDITION ONLY

ACTIONS
(continued)

B.1

With the SSS panel Pull Ring System inoperable, the Required Action specifies restoration within 72 hours. When the Pull Ring System is inoperable, automatic actuation of the SSS can still be achieved by the ABS/SC in the event of a scram failure. The 72 hour Completion Time is considered to be acceptable based on the OPERABILITY of the ABS/SC which is the primary means of actuation.

C.1

With both sides of the SSS inoperable, or if the Required Actions of Conditions A, or B are not met within the required Completion Times, Condition C is applicable. Under this Condition the plant must be placed in the safest possible status. Therefore, the reactor must be shutdown within 90 minutes.

SURVEILLANCE
REQUIREMENTS

SR 3.5.2.1.1

This surveillance demonstrates that the gadolinium solution storage tanks contain the necessary pressure to provide for injection of the gadolinium solution into the reactor tank and into the process water pump suction piping within the time assumed in the safety analysis. The pressure must be checked at least once per 12 hours to ensure that the SSS tank pressure remains within the required range of 280-320 psig (i.e., pressure has not bled off or increased beyond allowable levels). The 12 hour surveillance interval is adequate since fluctuations in the SSS pressure are infrequent and monitored continuously by alarms in the CCR.

SR 3.5.2.1.2

This surveillance verifies that the gadolinium solution storage tanks contain the required volume of gadolinium solution consistent with the assumptions made in the safety analysis. The pressure of nitrogen specified in SR 3.5.2.1.1 above in conjunction with the gadolinium solution volume specified in the surveillance ensure that the sparger injection duration will be at least 35 seconds and the gadolinium pump suction injection will be at least 65 seconds and that the rate of gadolinium solution injection

(continued)

BASES

INJECTION ONLY

SURVEILLANCE
REQUIREMENTS

SR 3.5.2.1.2 (continued)

will be at or above that assumed in the safety analysis for the entire duration of injection. Verification of the contained gadolinium solution volumes can only be conducted remotely by ensuring that the gadolinium solution storage tank High ($23 \pm 1/16$ " probes) and Very Low ($26 \frac{13}{16} \pm 1/16$ ") level lights on the SSS panel in the -40 Pump Observation Room (POR) are not illuminated. Conductivity probes are measured from the top of the probe threads that engage the probe nozzle, located on the top flange of the tanks, to the probe tip. Other level lights on the SSS panel can be used as an aid in this verification. A gadolinium solution volume within the high probe limit will ensure that the solution volume is within the 44.7 gallon limit. (Ref 32).

The contained gadolinium solution volume in each storage tank must be verified at least every 12 hours. The 12 hour surveillance interval is adequate since fluctuations in the SSS volume are small, and monitored continuously by alarms in the CCR.

SR 3.5.2.1.3

This surveillance demonstrates continuity in the portion of the ABS/SC SSS explosive valve firing circuit and primer that is in common with the manual actuation circuits, to permit valve actuation. Additional surveillances include verification that at least one of the explosive valve 24 volt battery per side indicates a voltage of ≥ 24 volts with the associated charger off (i.e., indicating adequate charge). Additionally, surveillance of bypass switch positions verifies that at least the minimum complement of ABS/SC explosive valve firing circuits, required to actuate both sides of the SSS are not open (bypassed). This surveillance must be performed at least once every 12 hours to ensure that the explosive valves will function as required. The 12 hour surveillance frequency is sufficient since the valve firing circuits are normally in a standby condition, continuously monitored and alarmed.

(continued)

BASES

**SURVEILLANCE
REQUIREMENTS
(continued)**

SR 3.5.2.1.4

This surveillance ensures that the injection lines are kept full of gadolinium solution. The only indication of SSS injection line level available during REACTOR OPERATION is the probe pot line level lights at the -40 Pump Observation Room. Conductivity probes are measured from the top of the probe threads that engage the probe nozzle, located on the top of the probe pots, to the probe tip. The three probe pots (per side) are connected such that if a low level is present in one probe pot, the remaining 2 probe pots will achieve the same level. This feature is useful for determining if a probe pot is truly low or if the indicating light circuit is inoperable. As long as at least 2 of 3 probe pot line level lights are illuminated on each side of the SSS it is assumed that the injection lines are full of gadolinium.

Verification that the probe pot line levels are normal (at least 2 of 3 lights illuminated at the -40 Pump Observation Room) must be performed at least once every 12 hours. The 12 hour surveillance interval is adequate since fluctuations in probe line levels are small, and monitored continuously by alarms in the CCR.

SR 3.5.2.1.5

This surveillance is a status check of the Seismic Trigger Subsystem of the SSS. Lights on the Seismic Trigger Subsystem bypass panels, alarm panels including the "ON-OFF" switch status, "DC VOLTMETER" voltage, alarm test, and checks of the status lights on the alarm panel battery pack provide assurance that the Seismic Trigger Subsystem of the SSS is OPERABLE. The 12 hour surveillance Frequency is based on extensive operating experience.

SR 3.5.2.1.6

This surveillance ensures that the pump suction injection lines are kept full of gadolinium solution.

Verification that the gadolinium solution level is greater than 16" within the standpipe sight glass (Ref 33) must be performed at least once every 12 hours. The 12 hour surveillance interval is adequate since fluctuations in the sight glass level are small.

(continued)

INFORMATION ONLY

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.5.2.1.7

This surveillance verifies that the pump suction injection instrumentation has power available and that the system is in a ready state for operation. The 12 hour surveillance Frequency is based on extensive operating experience.

SR 3.5.2.1.8

This surveillance demonstrates that the sparger injection and pump suction injection in-line block valves are open, thus allowing for SSS injection into the reactor. The sparger injection valves are continuously monitored by alarm, and status indicated by lights in the CCR. The surveillance must be performed prior to locking the -20' and -40' radiologically controlled areas prior to entering REACTOR STARTUP. The surveillance interval ensures that immediately prior to beginning the reactor startup procedure that the valves are verified open. There is no need for repeated surveillance during REACTOR OPERATION since the valves are located in the radiologically controlled areas which are high radiation areas and not subject to entry during REACTOR OPERATION and there is no mechanism (i.e., automatic valve actuators) for inadvertent misalignment of these valves.

SR 3.5.2.1.9

This surveillance is a status check of the batteries that supply power to the seismic triggers that are part of the Seismic Trigger Subsystem of the SSS. Verifying that the battery voltage is ≥ 11.5 VDC provides reasonable assurance that the seismic triggers will function as designed during a seismic event. The surveillance frequency of once every 12 hours is considered adequate since the system is continuously monitored and alarmed for abnormal conditions.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.5.2.1.10

This surveillance demonstrates for sparger injection and pump suction injection that the pneumatic valve nitrogen pressure is within the optimum range to preclude valve operator diaphragm damage or spurious actuation of the SSS pneumatic valves while allowing for manual SSS actuation or automatic actuation of the SSS for a seismic event via the Seismic Trigger Subsystem. The 24 hour surveillance frequency is adequate since failure of the valve operator diaphragm will result in fail-safe actuation of the SSS and safe shutdown of the reactor while allowing for detection of the condition prior to spurious actuation in most cases.

SR 3.5.2.1.11

This surveillance demonstrates that the gadolinium concentration is within limits prior to charging the gadolinium solution into the SSS. The SSS must contain gadolinium with a concentration of at least 0.14 grams of gadolinium per milliliter of solution, to ensure that the SSS will mitigate postulated transients or accidents within the assumptions made in the safety analysis. The 0.50 g/ml limit is significantly less than the 0.56 g/ml solubility limit of gadolinium at 25 °C. There is no known mechanism for degradation of the SSS gadolinium solution once charged to the gadolinium solution storage tank, therefore, the concentration of gadolinium is only required to be verified prior to charging.

SR 3.5.2.1.12

This surveillance demonstrates that the SSS injection lines contain less than 500 ml of trapped air. The volume of trapped air in the SSS is critical to the anticipated response time of the SSS. Analysis of the SSS response to trapped air (Ref. 12) indicates that air in excess of 1050 ml could cause the SSS to begin decreasing reactor power in a time greater than assumed in the safety analysis. Ultrasonic testing of the SSS injection lines is used to measure the amount of void space (trapped air) present in the lines subsequent to filling the system. Studies indicate that the evolution of air bubbles in the SSS gadolinium is insufficient to increase the system void space from the 500 ml limit to greater than 1050 ml (Ref. 14). This surveillance must be performed prior to REACTOR STARTUP if the SSS injection lines have been drained and refilled.

(continued)

INFORMATION

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.1.13

This surveillance ensures that the gadolinium solution in the SSS injection lines will remain in the injection lines and be available for injection into the reactor tank in the event of a SSS actuation. This surveillance must be performed prior to REACTOR STARTUP if the SSS injection lines have been filled or gadolinium has been charged to a gadolinium storage tank.

SR 3.5.2.1.14

This surveillance demonstrates that the sparger injection and pump suction injection orifices are in place. If the orifices were not in place, the SSS injection duration would be less than the times assumed in the safety analysis. The sparger injection and pump suction injection line orifices ensure that gadolinium sparger injection is metered such that the SSS will continue to inject the gadolinium solution for at least 35 seconds and that pump suction injection is metered to continue injection for at least 65 seconds. Functional testing conducted on the SSS includes the sparger injection and pump suction injection orifices being in place. Verification that all of the correct system orifices are installed must be performed prior to REACTOR STARTUP.

SR 3.5.2.1.15

This surveillance demonstrates that each seismic trigger battery is $\leq$ 18 months old. Operating experience shows that these batteries may be unreliable after extended service. Failure to replace the battery could result in battery failure (and SSS failure). Any battery that is older than 18 months should be replaced. This surveillance must be performed at least once every 6 months. The six month surveillance interval allows the battery to remain in service for a minimum of 18 months while ensuring that a battery nearing its end of life (24 months) is replaced.

SR 3.5.2.1.16

This surveillance ensures that the seismic triggers will initiate SSS actuation in the event that the 105-K building is subjected to a seismic disturbance that exceeds the seismic trip setpoint. This surveillance must be performed at least once every C&D outage.

(continued)

INFORMATION ONLY

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.1.17

This surveillance demonstrates that each explosive valve battery is ≤ 40 months old. Operating experience shows that the batteries are unreliable when in service longer than 48 months (Ref. 8). Failure to replace the battery could result in battery failure (and SSS failure). Any battery that is older than 40 months should be replaced. This surveillance must be performed at least once every 6 months. The six month surveillance interval allows the battery to remain in service for a minimum of 40 months while ensuring that any battery nearing its end of life (48 months) is replaced.

SR 3.5.2.1.18

This surveillance demonstrates that each explosive valve primer is ≤ 52 months old. The manufacturer recommends that the primers not be used for more than 60 months since age causes the primer reliability to decrease. The surveillance must be performed at least once every 6 months. Any primer that is older than 52 months should be replaced. The six month surveillance interval allows the primer to remain in service for a minimum of 52 months while ensuring that any primer nearing its end of life (60 months) is replaced.

SR 3.5.2.1.19

This surveillance demonstrates that the 24 volt batteries and associated firing circuitry on each side of the SSS is capable of providing at least 5 amperes for 8 msec. This load test ensures that the batteries are fit for continued service up to the maximum of 48 months and also demonstrates that the firing circuitry is capable of supplying the necessary power to ensure explosive valve primer detonation. The 18 month surveillance interval ensures that the batteries and firing circuits are checked at least twice during the maximum life of a battery bank and thus ensures that a high degree of reliability is maintained.

(continued)

BASES

INFORMATION ONLY

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.5.2.1.20

This surveillance demonstrates that the pneumatic valves will open completely and allow for manual actuation of the SSS. The required 5 second opening time is based on experience which shows that a properly functioning SSS pneumatic valve will facilitate full SSS injection within this time interval. The 24 month surveillance interval (Ref. 20) is designed to ensure that the valve is exercised regularly, since the valve does not normally cycle during plant operation. Note that this surveillance does not require actual injection of the gadolinium solution (or test fluid) into the reactor or into the pump suction.

SR 3.5.2.1.21

This surveillance demonstrates that no significant leakage exists between the process water space and the gadolinium solution space of each sparger assembly. Excessive leakage between the process water space and the gadolinium solution space of a sparger could adversely affect the gadolinium injection transient due to the effects of the back pressure which would have to be overcome. This surveillance must be performed at least once every 48 months and (for affected spargers) prior to REACTOR STARTUP after repair, replacement or modification of a sparger(s). The 48 month surveillance frequency is adequate since leakage of the spargers is uncommon unless spargers have been repaired, replaced or modified.

SR 3.5.2.1.22

During the C-area 1975 test of the SSS (Ref. 4), three explosive valves failed to open when fired. The valve failures were attributed to improper installation of the valve primer assemblies. This surveillance requires that the primer assembly-to-trigger body clearance be verified as ≤ 0.004 inches during primer assembly installation in an explosive valve trigger body, to prevent future valve failures. The surveillance is only required after replacing a primer assembly since there is no mechanism for the valve clearance to change once the assembly has been completed.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.5.2.1.22 (continued)

Subsequent to the 1975 test, new explosive valves were installed; not because of primer assembly problems but because an SSS explosive valve test program conducted in 1976 showed that the design safety factor for the valves could be as low as 1.1. The new valves were improved such that the design safety factor is at least 1.4 (Ref. 11).

No change to the SSS performance was expected due to installation of the new valve design since the valve body and primer characteristics were not changed. Subsequent SSS injection tests have shown (Ref. 6) that the new explosive valves did not cause any SSS performance degradation.

SR 3.5.2.1.23

This surveillance demonstrates that the sparger injection portion of the SSS is capable of performing the safety functions assumed in the safety analyses.

The surveillance response time limit of 4.8 seconds from loss of Process Water AC pump power to SSS sparger injection is conservative relative to the response time limit assumed in the bounding safety analysis of 5.0 seconds. (Ref. 33)

The 96 month surveillance frequency is based on engineering judgement and has been shown to be acceptable through extensive operating experience.

SR 3.5.2.1.24

This surveillance verifies that the SSS will function within the limits assumed in the safety analysis by demonstrating that: the explosive valves will fire; the gadolinium solution will begin reducing reactor power within 0.70 second; sparger injection from the bottom sparger hole will start within 0.88 seconds of explosive valve actuation; pump suction injection will start within 3.0 seconds of explosive valve actuation; and that the gadolinium solution injection rates and the duration of the injection for both sparger injection and pump suction injection are at or above those assumed in the safety analysis.

(continued)

INFORMATION ONLY

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.5.2.1.24 (continued)

The surveillance response time limit of 7.8 seconds from loss of Process Water AC pump power to SSS pump suction injection is conservative relative to the response time limit assumed in the bounding safety analysis of 8.0 seconds. (Ref. 33)

Verification that the pressure transducer response downstream of the ball check valve shows an initial pressure rise in ≤ 0.24 seconds and the pressure peaks in ≤ 0.29 seconds when using process water as the test fluid has been shown (Ref. 17) to indicate that under actual gadolinium solution injection, the reactor power would begin to decrease in ≤ 0.70 second after SSS actuation.

Verification that the test fluid injection continues for at least 35 seconds for sparger injection and 65 seconds for pump suction injection ensures that gadolinium will remain in the reactor tank throughout postulated transient or accident. This surveillance must be performed on both sides at least once every 96 months. The 96 month surveillance frequency is adequate based on historical data which demonstrates high reliability of the SSS. Test firing of the explosive valve with the oldest primer verifies that the valve firing circuits, batteries and primers are OPERABLE and the limit on primer life is adequate.

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(continued)

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BASES

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(continued)

BASES

INFORMATION ONLY

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(continued)

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Supplementary Safety System - Shutdown
B 3.5.2.2

ACTION ONLY

B 3.5 EMERGENCY COOLING SYSTEMS/EMERGENCY SHUTDOWN SYSTEMS

B 3.5.2.2 Supplementary Safety System - Shutdown

BASES

BACKGROUND	See the 3.5.2.1 Bases for the Background discussion of Supplementary Safety System (SSS).
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APPLICABLE SAFETY ANALYSES	See the 3.5.2.1 Bases for the discussion of the SSS Applicable Safety Analyses.
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LCO	Both sides of the SSS fluid system actuated by the SSS panel Pull Ring and the SSS explosive valve manual actuation switches must be OPERABLE. The remaining portions of the SSS fluid system and the other SSS actuating instrumentation are not required to be OPERABLE. This required capability is adequate to respond to any postulated criticality accident during REACTOR SHUTDOWN.
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APPLICABILITY	The SSS fluid systems actuated by the SSS panel Pull Ring or via the explosive valve manual actuation switches must be OPERABLE during REACTOR SHUTDOWN with fuel in the reactor and rods capable of being withdrawn or with POSITIVE REACTIVITY CHANGES in progress. This ensures that gadolinium solution can be injected in the event of a criticality accident. The SSS must remain OPERABLE until all rods are inserted and their power supplies are de-energized and no POSITIVE REACTIVITY CHANGES are being made.
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BASES (continued)

ACTIONS

A.1

With one side of the SSS inoperable, the Required Action specifies restoration of the inoperable system within 72 hours. Under this degraded condition the SSS is vulnerable to a single failure which could lead to a complete loss of system function. This Completion Time is considered to be acceptable based on past reliability of the SSS and the remote possibility that the SSS would be required to mitigate a reactivity excursion when the reactor is shutdown.

B.1 and B.2

With the SSS panel Pull Ring System inoperable, both sides of the SSS inoperable, or the Required Action and associated Completion Time of Condition A not met, the Required Action specifies IMMEDIATELY suspending operations involving POSITIVE REACTIVITY CHANGES, insertion of any withdrawn rods and de-energization of the rod drive power supplies. These Actions place the reactor in the safest possible status to minimize the potential for a criticality accident.

SURVEILLANCE
REQUIREMENTS

SR 3.5.2.2.1

This surveillance demonstrates that the gadolinium solution storage tanks contain the necessary pressure to provide for injection of the gadolinium solution into the reactor tank within the time assumed in the safety analysis. The pressure must be checked at least once per 12 hours to ensure that the SSS tank pressure remains within the required range of 280-320 psig (i.e., pressure has not bled off or increased beyond allowable levels). The 12 hour surveillance interval is adequate since fluctuations in the SSS pressure are infrequent and monitored continuously by alarms in the CCR.

(continued)

BASES

INFORMATION ONLY

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.5.2.2.2

This surveillance verifies that the SSS gadolinium solution storage tanks contain the required volume of gadolinium consistent with the assumptions made in the safety analysis. The minimum volume of gadolinium specified ensures that the gadolinium sparger injection duration will be at least 35 seconds. The maximum volume of gadolinium specified in conjunction with the nitrogen pressure specified above ensure that the required volume of nitrogen will be available to ensure that the rate of gadolinium solution injection will be at or above that assumed in the safety analysis for the entire duration of injection. Verification of the contained gadolinium volumes can only be conducted remotely by ensuring that the gadolinium storage tank High ($23 \pm 1/16"$ probes) and Very Low ($26 \pm 1/16"$) level lights on the SSS panel in the -40 Pump Observation Room (POR) are not illuminated. Conductivity probes are measured from the top of the probe threads that engage the probe nozzle, located on the top flange of the tanks, to the probe tip. Other level lights on the SSS panel can be used as an aid in this verification. The contained gadolinium volume in each storage tank must be verified at least every 12 hours. The 12 hour surveillance interval is adequate since fluctuations in the SSS volume are small, and monitored continuously by alarms in the CCR.

SR 3.5.2.2.3

This surveillance ensures that the injection lines are kept full of gadolinium solution. The only indication of SSS injection line level available is the probe pot line level lights at the -40 Pump Observation Room. Conductivity probes are measured from the top of the probe threads that engage the probe nozzle, located on the top of the probe pots, to the probe tip. The three probe pots (per side) are connected such that if a low level is present in one probe pot, the remaining 2 probe pots will achieve the same level. This feature is useful for determining if a probe pot is truly low or if the indicating light circuit is inoperable. As long as at least 2 of 3 probe pot line level lights are illuminated on each side of the SSS it is assumed that the injection lines are full of gadolinium.

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BASES

INFORMATION ONLY

SURVEILLANCE
REQUIREMENTS

SR 3.5.2.2.3 (continued)

Verification that the probe pot line levels are normal (at least 2 of 3 lights illuminated at the -40 Pump Observation Room) must be performed at least once every 12 hours. The 12 hour surveillance interval is adequate since fluctuations in probe line levels are small, and monitored continuously by alarms in the CCR.

SR 3.5.2.2.4

This surveillance verifies that the injection line block valves are open, thus allowing for SSS injection into the reactor. The valves are continuously monitored by alarm, and status indicated by lights in the CCR. The surveillance must be performed each 12 hours, since these valves are accessible for manual operation during REACTOR SHUTDOWN.

SR 3.5.2.2.5

This surveillance demonstrates continuity in the portion of the ABS/SC SSS explosive valve firing circuit and primer that is in common with the manual actuation circuits, to permit valve actuation. Additional surveillances include verification that at least one of the explosive valve 24 volt nicad battery per side indicates a voltage of ≥ 24 volts with the associated charger off (i.e., indicating adequate charge). Additionally, surveillance of bypass switch positions verifies that at least the minimum complement of ABS/SC explosive valve firing circuits, required to actuate both sides of the SSS are not open (bypassed). This surveillance must be performed at least once every 12 hours to ensure that the explosive valves will function as required. The 12 hour surveillance frequency is sufficient since the valve firing circuits are normally in a standby condition, continuously monitored and alarmed.

(continued)

1. INFORMATION ONLY

BASES

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(continued)

SR 3.5.2.2.6

This surveillance demonstrates that the pneumatic valve surge tank pressure is within the optimum range to preclude valve operator diaphragm damage or spurious actuation of the SSS pneumatic valves while allowing for manual SSS actuation. The 24 hour surveillance frequency is adequate, since failure of the valve operator diaphragm will result in fail-safe actuation of the SSS while allowing for detection of the condition prior to spurious actuation in most cases.

SR 3.5.2.2.7

This surveillance demonstrates that each explosive valve nicad battery is ≤ 40 months old. Operating experience shows that the nicad batteries are unreliable when in service longer than 48 months (Ref. 8). Failure to replace the battery could result in battery failure (and SSS failure). Any nicad battery that is older than 40 months should be replaced. This surveillance must be performed at least once every 6 months. The six month surveillance interval allows the battery to remain in service for a minimum of 40 months while ensuring that any battery nearing its end of life (48 months) is replaced.

SR 3.5.2.2.8

This surveillance demonstrates that each explosive valve primer is ≤ 52 months old. The manufacturer recommends that the primers not be used for more than 60 months since age causes the primer reliability to decrease. The surveillance must be performed at least once every 6 months. Any primer that is older than 52 months should be replaced. The six month surveillance interval allows the primer to remain in service for a minimum of 52 months while ensuring that any primer nearing its end of life (60 months) is replaced.

SR 3.5.2.2.9

This surveillance demonstrates that the 24 volt nicad batteries and associated firing circuitry on each side of the SSS is capable of providing at least 5 amperes for 8 msec. This load test ensures that the batteries are fit for continued service up to the maximum of 48 months and also demonstrates that the firing circuitry is capable of supplying the necessary power to ensure explosive valve primer detonation. The 18 month surveillance interval

(continued)

DISCONTINUATION ONLY

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.5.2.2.9 (continued)

ensures that the batteries and firing circuits are checked at least twice during the maximum life of a battery bank and thus ensures that a high degree of reliability is maintained.

SR 3.5.2.2.10

During the C-area 1975 test of the SSS (Ref. 4), three explosive valves failed to open when fired. The valve failures were attributed to improper installation of the valve primer assemblies. This surveillance requires that the primer assembly-to-trigger body clearance be verified as ≤ 0.004 inches during primer assembly installation in an explosive valve trigger body, to prevent future valve failures. The surveillance is only required after replacing a primer assembly since there is no mechanism for the valve clearance to change once the assembly has been completed.

Subsequent to the 1975 test, new explosive valves were installed; not because of primer assembly problems but because an SSS explosive valve test program conducted in 1976 showed that the design safety factor for the valves could be as low as 1.1. The new valves were improved such that the design safety factor is at least 1.4 (Ref. 11).

No change to the SSS performance was expected due to installation of the new valve design since the valve body and primer characteristics were not changed. Subsequent SSS injection tests have shown (Ref. 6) that the new explosive valves did not cause any SSS performance degradation.

SR 3.5.2.2.11

This surveillance demonstrates that the gadolinium concentration is within limits prior to charging the gadolinium solution into the SSS. The SSS must contain gadolinium with a concentration of at least 0.14 grams of gadolinium per milliliter of solution, to ensure that the SSS will mitigate postulated transients or accidents within the assumptions made in the safety analysis. The 0.50 g/ml limit is significantly less than the 0.56 g/ml solubility limit of gadolinium at 25 °C. There is no known mechanism for degradation of the SSS gadolinium solution once charged to the gadolinium solution storage tank, therefore, the concentration of gadolinium is only required to be verified prior to charging.

(continued)

INFORMATION ONLY

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.2.12

This surveillance demonstrates that the SSS injection lines contain less than 500 ml of trapped air. Analysis of the SSS response to trapped air (Ref. 12) indicates that air in excess of 1050 ml could cause the SSS to begin decreasing reactor power in a time greater than assumed in the safety analysis. Ultrasonic testing of the SSS injection lines is used to measure the amount of void space (trapped air) present in the lines subsequent to filling the system. Studies indicate that the evolution of air bubbles in the SSS gadolinium is insufficient to increase the system void space from the 500 ml limit to greater than 1050 ml (Ref. 14). This surveillance must be performed whenever the SSS injection lines have been drained and refilled.

SR 3.5.2.2.13

This surveillance ensures that the gadolinium solution in the SSS injection lines will remain in the injection lines and be available for injection into the reactor tank in the event of a SSS actuation. This surveillance must be performed if the SSS injection lines have been filled or gadolinium has been charged to a gadolinium storage tank.

SR 3.5.2.2.14

This surveillance demonstrates that the sparger injection orifices are in place. If the orifices were not in place, the SSS injection duration would be less than the times assumed in the safety analysis. The sparger injection line orifices ensure that gadolinium sparger injection is metered such that the SSS will continue to inject the gadolinium solution for at least 35 seconds. Functional testing conducted on the SSS includes the sparger injection orifices being in place. Verification that the correct system orifices are installed must be performed if the SSS injection lines have been filled or gadolinium has been charged to a gadolinium storage tank.

SR 3.5.2.2.15

This surveillance demonstrates that the pneumatic valves will open completely and allow for manual actuation of the SSS. The required 5 second opening time is based on experience which shows that a properly functioning SSS pneumatic valve will facilitate full SSS injection within

(continued)

BASES

SHUTDOWN UNIT

SURVEILLANCE
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SR 3.5.2.2.15 (continued)

this time interval. The 24 month surveillance interval (Ref. 20) is designed to ensure that the valve is exercised regularly, since the valve does not normally cycle during plant operation. Note that this surveillance does not require actual injection of gadolinium solution (or test fluid) into the reactor.

SR 3.5.2.2.16

This surveillance verifies that the SSS will function within the limits assumed in the safety analysis by demonstrating that: the explosive valves will fire; the gadolinium solution will begin reducing reactor power within 0.70 second; sparger injection from the bottom sparger hole will start within 0.88 seconds of explosive valve actuation; and that the gadolinium solution injection rates and the duration of the injection for sparger injection is at or above those assumed in the safety analysis.

The surveillance response time limit of 4.8 seconds from loss of Process Water AC pump power to SSS sparger injection is conservative relative to the response time limit assumed in the bounding safety analysis of 5.0 seconds. (Ref. 33)

Verification that the pressure transducer response downstream of the ball check valve shows an initial pressure rise in ≤ 0.24 seconds and the pressure peaks in ≤ 0.29 seconds when using process water as the test fluid has been shown (Ref. 17) to indicate that under actual gadolinium solution injection, the reactor power would begin to decrease in ≤ 0.70 second after SSS actuation.

Verification that the test fluid injection continues for at least 35 seconds for sparger injection ensures that gadolinium will remain in the reactor tank throughout a postulated transient or accident. This surveillance must be performed on both sides at least once every 96 months. The 96 month surveillance frequency is adequate based on historical data which demonstrates high reliability of the SSS. Test firing of the explosive valve with the oldest primer verifies that the valve firing circuits, batteries and primers are OPERABLE and the limit on primer life is adequate.

REFERENCES

See B 3.5.2.1 for list of References.

B 3.6 CONFINEMENT SYSTEMS

- B 3.6.1 Airborne Activity Confinement System - HGAs
B 3.6.2 Airborne Activity Confinement System - No HGAs

BASES

The requirements for the airborne activity confinement system are intended to reduce the potential offsite thyroid dose for credible accidents and typical conditions to as low as reasonably achievable. Comparison of doses from various postulated accident conditions with federal regulations are discussed in Reference 1.

The process areas of the reactor building are sufficiently sealed that they can be maintained at a negative pressure with respect to adjoining areas (and the atmosphere) by exhausting air from the process areas through filtration-adsorption compartments which are provided for each reactor building, at least three of which are always online when the reactor contains heat generating assemblies. Each compartment contains: moisture separators for removal of entrained water particles; followed by filters for removal of particulates; followed by carbon beds for adsorption of halogens. No provision is made for retention of noble gases or tritium.

Charge-discharge operations that affect reactivity are those involving components containing fissile material, or significant neutron absorbing material, such as lithium or cadmium. For purposes of this Specification, moderator temperature changes are not considered to produce reactivity changes.

1. The bases for the required carbon bed performance are:

The carbon beds in the filter compartments are the primary defense of the airborne activity confinement system against release of radioiodine to the atmosphere in the unlikely event of a melting accident.

(continued)

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BASES

(continued)

The mechanical integrity of the assembled carbon beds is measured following installation and periodically thereafter using in place testing methods (Refs. 2 and 3). The performance of the carbon in the beds is measured by periodic removal of beds from the system and subjecting a representative carbon sample removed from a bed to a standard methyl iodide penetration test. The original radiolytic desorption test (Ref. 4) is no longer performed. A 5-hour methyl iodide penetration test was developed that was directly correlated to the radiolytic desorption test (Ref. 5). The methyl iodide test is used in conjunction with the developed correlation to predict radiolytic and thermal desorption performance. The data for acceptable carbon beds show that the radiolytic and thermal desorption at 80°C will be less than 0.70% during a 5-hour period. This is for carbon aged 30 months, the maximum service age of carbon. Thermal desorption of iodine at 80°C for carbon aged 30 months is very low (0.01%, Ref. 6). The results of calculated thyroid and whole body doses for potential accident sequences are given in Reference 1. For these calculations it is assumed that the instantaneous filter bypass is 0.05% of the iodine reaching the confinement system and the desorption during the first 5 hours after the assumed accident is 0.70% of the equilibrium iodine inventory (reaching the carbon) at 3000 MW.

In the event of an uncontrolled release in the 105 Building of a significant quantity of fumes, chemicals, or foreign materials (e.g., fire, paint spill, etc.) having an unknown effect on the carbon filters, and evaluation and/or special tests are conducted to determine if the carbon filters are adversely affected.

(continued)

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BASES

(continued)

2. The bases for specifying a minimum carbon ignition temperature are:

Radioactive decay of halogens absorbed on the carbon in the confinement system generates heat which is normally removed by air flowing through the system. Abnormal filter plugging (from dust or smoke particle loading or failure of filter media water repellency) could cause a reduced air flow through the filter compartments and an additional temperature increase in the carbon beds. Also, service aging reduces the ignition temperature of carbon.

If the temperature rise is sufficient to ignite the carbon, the performance of the airborne activity confinement system would be severely degraded. To ensure that the system will not fail because of carbon ignition caused by radiiodine decay heat, a minimum carbon ignition temperature of 250°C is needed. This temperature is above that at which most of the iodine would be thermally desorbed (Ref. 6). Specification of the minimum ignition temperature of new carbon at 330°C (Ref. 7) ensures that the ignition temperature of service aged carbon does not fall to 250°C. The carbon ignition temperature is measured according to the procedure described in ASTM D3466 except at a face velocity of 55 + 1.6 fpm and a heating rate of 5°C/min., and is reported to the nearest 5°C.

3. The bases for the required particulate filter efficiency (99.97%) are:

- a) Efficiencies approaching 99.97% are required to retain iodine-laden particulates that could otherwise pass through the carbon bed and result in excessive thyroid doses (Ref. 2). In addition, the highest practical efficiency is desirable to minimize the effects of any activity release. In this case, the highest practical efficiency requirement is judged to be 99.97% (Ref. 17).

(continued)

INFORMATION ONLY

BASES

(continued)

- b) The efficiency of the filter media is measured with 0.3-micron particulates prior to assembly, and the mechanical integrity of the installed filters is measured with 0.6-micron particles following installation. Tests have shown that 99.5 weight percent of particles released from burning or melting uranium would be 2-micron diameter (Refs. 2 and 8).
4. The bases for the minimum required exhaust capacity of the online airborne activity confinement system are:
- a) Reliance is placed on negative pressure to prevent leakage of airborne activity from the process areas by any path other than through the filtration-adsorption units. Negative pressure is normally maintained by the continuous operation of at least two of the three exhaust fans, each of which can be operated by an electrical power source that is independent of the normal reactor building power distribution system. The normal building exhaust air flow with two fans on line is about 100,000 cfm.

Negative pressure is maintained even while the reactor is shut down because the remote possibility of an accident still exists. However, when the reactor is not operating, discharge of air from below grade directly to the stack (no filtration) is permitted as necessary to facilitate routine maintenance of the reactor systems. The risk of an unfiltered airborne release from a reactor accident is considered small, because normal filtration could be re-established well before the accident progressed far enough to admit radioactive releases into the below grade process areas. Also, the extremely small probability of a reactor accident is judged to be even less during REACTOR SHUTDOWN (except during charge and discharge operations) than during operation.

(continued)

BASES

(continued)

- INFORMATION ONLY**
- b) An exhaust flow of 40,000 cfm maintains, under normal ventilation conditions, the process areas at negative pressure with respect to the personnel areas and the atmosphere (Ref. 9). This air flow is also sufficient to maintain carbon temperature below that associated with desorption and well below 250°C (minimum required ignition temperature for carbon) even if 50% of the equilibrium iodine inventory is deposited on the carbon beds of only two confinement compartments. This assumes equal air distribution through the filter compartments and that the Emergency Cooling System (ECS) or the Confinement Heat Removal System (CHRS), is actuated to prevent overheating of the building air (Ref. 10).
- c) A minimum total exhaust capacity of 21,000 cfm per inch of H₂O average delta P across online particulate filters is specified to prevent damage to the particulate filters from exposure to increased air and/or steam flow. The increased flow might occur during a severe reactor power excursion or during delayed ECS addition following a loss of process water (Refs. 2 and 11).

This exhaust capacity will discharge air and steam from the reactor room at a maximum peak rate of 800,000 cfm (well in excess of any credible accidents) with a delta P of less than 15 inches of H₂O across the particulate filters. Tests indicate that the particulate filters will not be significantly damaged at a delta P less than 15 inches of water (Refs. 12 and 13).

(continued)

BASES

(continued)

- d) Three online filter compartments are required when heat generating assemblies are in the reactor, so that the minimum exhaust flow required to maintain process areas at negative pressure (40,000 cfm) can be maintained even if adverse conditions following a severe reactor accident cause an increase in filter flow resistance (Ref. 11). If no heat generating assemblies are in the reactor, the airborne releases from a potential accident are reduced sufficiently to be contained by one filter compartment.
- e) Two online exhaust fans are also required to ensure confinement system pressure relief capacity (Ref. 2), and the ability to maintain the carbon beds at low enough temperatures to prevent thermal desorption if they were heavily loaded with iodine. To allow time for startup of an additional fan after an online fan has failed, three hours are permitted with only one fan online.

In the unlikely event that an activity release is initiated during operations affecting reactivity, at least one [REDACTED] fan must be operating to ensure that the release is drawn through the confinement filters. Redundant protection is provided by maintaining two fans online during operations affecting reactivity when heat generating assemblies are in the reactor. If no heat generating assemblies are in the reactor, the pressure surge from a potential accident is greatly reduced, so that only one [REDACTED] fan online is needed.

REFERENCES

1. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 6.2.2, "Airborne Activity Confinement System." (U)
2. DPST-66-300. W. S. Durant, et al., Activity Confinement System of the SRP Reactors, February 1966. (UNCI)

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BASES

REFERENCES
(continued)

3. DPSOL 105-1874B-PLKC, Rev. 2, DPSOP 330-44, "Freon Test of Carbon Filters Using B-C Analyzer", Approval date 7/8/82. (UNCI)
4. DP-1329. A. G. Evans. Confinement of Airborne Radioactivity - Progress Report: July-December 1972, June 1973. (U)
5. DPST-78-630. A. G. Evans. A New Test Method for Evaluating the Performance of SRP Confinement System Carbon, December 24, 1978. (U)
6. DP-1213. R. C. Milham, L. R. Jones. Iodine Retention Studies - Progress Report, January-June 1969, November 1969. (U)
7. DPSOP 116. Essential Material Specifications, EM Specification 190, Activated Carbon Filters (100 Areas), Revision 1, 8/80. (U)
8. DP-812. A. H. Peters. Application of Demisters and Particulate Filters in Reactor Containment, December 1962. (U)
9. RTR-1382. J. B. Guffey. Minimum Reactor Building Flows Required for Subatmospheric Process Area Pressures, April 9, 1973. (U)
10. RTW-222. J. H. Hinton and S. P. Tinnes. FY 1977 Budget Proposals - Improved Confinement of Radioactive Releases, January 1976 and subsequent revision. (U)
11. DPST-70-433. W. S. Durant and R. J. Brown. Analysis of Postulated Core Meltdown of an SRP Reactor. October 1970. (UNCI)
12. DPST-73-482, L. R. Jones, Effects of HEPA Filter Blinding on Confinement System Performance. November 2, 1973. (U)
13. DP-1316. A. G. Evans and L. R. Jones. Confinement of Airborne Radioactivity - Progress Report. January-June 1972, January 1973. (U)

(continued)

BASES

INFORMATION ONLY

REFERENCES
(continued)

14. DPST-79-555. S. P. Tinnes. Airborne Activity Confinement System Performance - First Five Hours after Reactor Accident. (U)
 15. Reactor Site Criteria, Title 10, Code of Federal Regulations, Part 100, February 1961 and subsequent amendments. (U)
 16. DPST-83-844. J. B. Price. Calculational Method Used for Accidental Airborne Release and Resultant from SRP Reactors, August 23, 1983. (U)
 17. DPSOP 40-2. "Reactor Confinement HEPA Filter", Rev. 5, December 4, 1990. (U)
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B 3.7 PLANT SYSTEMS

INFORMATION ONLY

B 3.7.1 Cooling Water System Flow

BASES

Full cooling water flow is established prior to REACTOR STARTUP and maintained during REACTOR OPERATION. This requirement ensures that reactivity increases will not occur due to starting an idle cooling water pump or opening a closed cooling water valve.

REFERENCES - None.

B 3.7 PLANT SYSTEMS

INFORMATION ONLY

B 3.7.2 Ultimate Heat Sink

BASES

BACKGROUND

The Ultimate Heat Sink for the K Production Reactor is the 186 basin. The 186 basin is the source of cooling water provided to dissipate reactor decay heat and essential cooling system heat loads. It is also the water source for the Emergency Cooling System (ECS) and the Process Room Spray System (PRSS). The minimum level and maximum temperature of the 186 basin ensure adequate heat load removal capacity after a normal reactor shutdown or a shutdown following an accident, including a Loss of Coolant Accident (LOCA) or a Loss of Pumping Accident (LOPA). During the course of a normal reactor shutdown or the mitigation of a LOCA or LOPA the 186 basin water inventory will be depleted and must be replenished. The water inventory can be replenished by the River Water System or the [REDACTED] building effluent sump can be recirculated back to the 186 basin.

The water source which normally supplies the 186 basin is the River Water System which draws water from the Savannah River. The 186 basin is designed to operate at a minimum level of ≥ 13.8 feet. The minimum level ensures an adequate Net Positive Suction Head (NPSH) for the Cooling Water System (CWS) pumps, the ECS pumps and the process room spray pumps. The ultimate heat sink function of the 186 basin is described in the Final Safety Analysis Report (FSAR) Section 9.2.5 (Ref. 1).

**APPLICABLE
SAFETY ANALYSES**

Since decay heat removal must be provided for during a normal shutdown as well as part of long term accident mitigation, the ultimate heat sink is a necessary support system whenever there are heat generating assemblies in the reactor tank.

The parameters of the 186 basin are modeled in the Final Safety Analysis Report (FSAR) Section 15.2.2.4 (Ref. 2). The 186 basin serves as a support system to various systems which are assumed to function to mitigate various Design Basis Accidents (DBA). The River Water System (RWS) which supplies water to the 186 basin also is a support system which is required to be OPERABLE whenever the 186 basin is OPERABLE.

(continued)

BASES

**APPLICABLE
SAFETY ANALYSES
(continued)**

The RWS is vulnerable to certain design basis accidents such as loss of off-site power or flooding of the river water pumphouse. The consequences of the loss of the RWS can be minimized by recirculating cooling water back to the 186 basin from the building effluent sump. With recirculation, the 186 basin can perform the ultimate heat sink function for approximately 9 days (FSAR 9.2.2.1) with the reactor shutdown. The 186 basin is a system that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

LCOs

Without an OPERABLE 186 basin, the heat generated from fission product decay following shutdown may not be removed and could result in the eventual melting of the fuel elements and release of fission product activity to the reactor coolant. Should this occur in conjunction with a LOCA, fission product activity could be released to the 105 building atmosphere.

The 186 basin average water temperature limits are based on precluding process water freezing (FSAR 9.2.2), preserving the GM Diesel cooling water temperature limits, and preserving the safety analysis LOCA assumption for the highest temperature of ECS water.

The 186 basin minimum allowable water level is based on the required water volume for core cooling and mitigating temperature increases after loss of make-up water to the 186 basin.

The interconnection of all three sections of the 186 basin and the two 190 pump suction wells ensure a correct basin level indication since any basin section that is isolated will yield an incorrect reading of the total basin level.

APPLICABILITY

The 186 basin ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal operations and during and after a DBA. The 186 basin must be OPERABLE whenever the reactor or the disassembly basin contains heat generating assemblies so that decay heat can be safely dissipated.

186 BASIN ONLY

BASES

ACTIONS

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

If the 186 basin minimum water level requirement is not met the reactor must be shutdown, so that the demand for decay heat rejection is minimized. This is done by placing the unit in REACTOR SHUTDOWN immediately.

In addition, action must be initiated immediately to restore the 186 basin water level and the cooling water effluent must be aligned for recirculation back to the 186 basin to conserve the basin water inventory. The completion time of 1 hour for aligning the cooling water effluent to the recirculation alignment is based on the relative ease of the operation (opening the pump discharge valve and starting the recirculation pump). All water conservation actions must be taken, particularly shutting down the cooling water pumps and valving off 10 of 12 heat exchangers.

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

Action B.1 requires that the cooling water be recirculated back to the 186 basin if the minimum temperature of 5°C is not met. Recirculation will return warmer water to the 186 basin and eventually increase the temperature sufficiently to restore the minimum temperature to at least 5°C. The completion time of 1 hour for aligning the cooling water effluent to the recirculation alignment is based on the relative ease of the operation (opening the pump discharge valve and starting the recirculation pump). If the completion time of 1 hour can not be met, REACTOR SHUTDOWN must occur in the time listed. While putting the reactor in the shutdown condition, actions which restore the cooling water temperature to 5°C should be initiated immediately.

C.1

Action C.1 requires that the reactor be shutdown if the maximum temperature of 30°C is exceeded. This action will reduce the probability of a loss of coolant accident mitigation failure because of ECS water being warmer than the temperature assumed in the safety analysis. No action to restore the temperature is required because the 186 basin temperature is determined by the temperature of the Savannah River and there is no practical means available to the plant to change the river temperature.

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BASIS

ACTIONS
(continued)

D.1

Action D.1 requires that all 186 basin sections and two 190 pump suction wells be interconnected immediately if they fail to be interconnected. The action will ensure the 186 basin level reading of the 186 basin water level will be a correct representation of the basin level. An incorrect reading results if one of the 186 basin level sections or 190 pump suction wells are not interconnected.

SURVEILLANCE
REQUIREMENTS

SR 3.7.2.1

The temperature specified ensures that the ECS and the CWS can provide emergency core cooling and cool the safety related heat loads to their maximum design temperature with the maximum accident design heat loads. This temperature also ensures that the process water system will be protected from damage due to freezing. The frequency of 12 hours is based on engineering judgment. This frequency has been shown to be acceptable through operating experience.

SR 3.7.2.2

The level specified ensures that there is at least a minimum volume of water and that basin temperature increases will not result in exceeding GM Diesel cooling water temperatures. The frequency of 12 hours is based on engineering judgment. This frequency has been shown to be acceptable through operating experience.

SR 3.7.2.3

The interconnection specified ensures the basin level indication is a correct reading of the total 186 basin water level. The frequency of 7 days is based on engineering judgement. This frequency has been shown to be acceptable through operating experience.

REFERENCES

1. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 9.2.5, "Ultimate Heat Sink." (U)
 2. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 15.2.2., "Loss of Cooling Water Supply to The 186 Basin." (U)
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INFORMATION ONLY

B 3.8 ELECTRICAL SYSTEMS

BASES

Power shall be provided to permit the following functions in the event of loss of offsite power:

1. Operation of reactor shutdown systems including control rods, safety rods, and supplementary safety system.
2. Circulation of process water with adequate distribution to cool irradiated components in the reactor.
3. Circulation of cooling water through the main heat exchangers.
4. Monitoring of critical nuclear and hydraulic parameters.
5. Operation of the airborne activity confinement system.
6. Operation of the emergency cooling system.
7. Discharge of irradiated components from the reactor.

Each SRS production reactor is provided with a variety of redundant and diverse power sources as shown in Table 3.8-1. These sources (in addition to use of gravity, batteries, and compressed gas for some applications) are intended to permit normal operation of the reactors as well as to provide for contingencies. The vital functions required by these LCOs will permit: shutdown of the reactor to reduce heat generation, removing decay heat from the shutdown reactor under normal and emergency conditions, monitoring the nuclear and hydraulic status of the reactor, confinement of radioactivity released from the core, and removal of the core from the reactor. Each of these operations is necessary for coping with the various types of incidents considered in the operation of the reactors.

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ELECTRICAL SYSTEMS
B 3.8

BASES (continued)

TABLE 3.8-1

Typical Power Supplies

<u>Description</u>	<u>Total Available</u>	<u>Minimum Provided During REACTOR OPERATION</u>
Rod Equipment Room (RER) 125 volt floating emergency battery	1	1
Rod Equipment Room (RER) motor-generator/ motor-alternator set	2	1
Emergency AC Diesel Generator (GM)	2	2
████████ Fan Emergency Diesel Generator	2	2
Normal AC Power Supplies	2 or 3	1

REFERENCES

1. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 8.0, "Electric Power." (U)
2. DPST-70-4197. G. F. Merz, et. al., Analysis of the Savannah River Protective Instrument System, October 1970. (UCNI)

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B 3.9 COMPONENTS HANDLING

B 3.9.1 Irradiated Assemblies

BASES

For each type of assembly, discharge is permitted only after the heat generation rate has decayed to a specified limit. For fuel and target assemblies, the specified limit is the lower of the following (Ref. 1):

Maximum heat generation for adequate cooling by the discharge machine cooling system

Maximum heat generation rate for adequate cooling if the assembly is dropped accidentally to a horizontal position on the disassembly floor

Maximum heat generation rate to allow initiation of primary and secondary discharge machine cooling systems and provide full coolant flow to the base of the assembly without exceeding a surface temperature of 250°C.

The 250°C maximum temperature criteria is based on tests which showed that blisters formed during HGA manufacture (extrusion) would probably reappear at temperatures above 350°C; however, no growth, rupture, or formation of new blisters would be expected (Ref. 2).

The heat generation rates are a strong function of assembly geometry and are calculated for each assembly type based on heat transfer and flow distribution tests (Ref. 1). The LCO time limit assures that the heat generation rate has decayed to a level which meets the above criteria. The time after shutdown which must be met prior to discharge of HGAs is a cycle specific parameter and is specified in the CORE SHUTDOWN REPORT.

(continued)

Information

BASES (continued)

REFERENCES

1. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 9.1.4, "Light Load Handling Systems (Related to Refueling)." (U)
 2. DPSTM-18 (H). Technical Manual - Hydraulics and Heat Transfer of Mark 18 and Mark 18A Fuel Assemblies. (U)
 3. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 15.7.1, "Fuel Handling Accident." (U)
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B 3.9 COMPONENTS HANDLING

B 3.9.2 Reactor Level

INFORMATION ONLY

BASES

The presence of process water in the lattice is necessary to obtain meaningful neutron flux levels at existing monitoring positions. Changes in the process water level can affect monitor readings in a manner unrelated to multiplication of the lattice. Also, if a reactor loading were such that it would be critical at less than full process water height, reactivity could be added rapidly from filling the tank with process water.

REFERENCES

1. DPST-71-342, P. L. Roggenkamp and H. E. Hootman to G. Dessauer, Safety of Charge and Discharge Operations - Results of the Ad Hoc Committee, September 7, 1971. (U)
 2. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Table 15.1-4, "Safety Rod Scram System Actuation Points and Times Assumed in Analysis." (U)
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B 3.9 COMPONENTS HANDLING

B 3.9.3 Fissile Components - Handling and Storage

INFORMATION ONLY

BASES

Nuclear criticality safety is maintained by exercising control over amounts and configurations of fissile materials and of associated material that may affect reactivity. The principal fissile materials are ^{233}U , ^{235}U , ^{239}Pu , and ^{241}Pu . Associated materials can affect reactivity by neutron moderation, reflection, or capture.

Criticality is prevented by controlling a specified set of variables within sufficiently restrictive limits. The limits imposed upon each controlled variable are sufficient to prevent criticality when all other controlled variables have their limiting values and when all uncontrolled variables have their most reactive values.

A controlled variable may be held within a limit by procedure, by equipment design, or by nature. Examples are, respectively, the maximum number of identical fuel units allowed in a given operation, the spacing among units maintained by a storage rack, and the absence of snow. (Being a low-density hydrogenous material, snow could introduce optimum moderation.)

The limit imposed upon one controlled variable generally depends on the limit imposed on another. For example, the limit on the number of fuel tubes that may be in an isolated group without regard for the spacing among them depends on the fissile content of the fuel tubes.

Some variables may assume only two values, yes or no. For example, the control of integrity requires that a fuel element should not be cut. A number of enriched fuel tubes, safe in any configuration, may no longer be safe after being cut up into shorter pieces.

(continued)

ONLY

BASES

(continued)

The technical bases for derivation of the limits are given in the Technical Manual (Ref. 1), in which additional references are listed, including references describing all special boxes, buckets, storage racks, etc.. Experimental data and calculations normalized to available data for similar configurations are the bases for determining whether or not a configuration is subcritical. Neither data nor calculations may be considered exact, and margins are required that are judged to be adequate to compensate for inaccuracies in both the data and the calculations. In choosing the margins, two conditions must be considered: one is the accuracy with which the computational technique calculates the reactivity of configurations measured experimentally; the other is the accuracy with which the computation represents the actual configurations of fissile materials. For example, the former margin may be the normalization of the computational technique to a measurement in an exponential facility; the latter may be the margin allowed when the computation is applied to the disassembly basin. Margins allowed in the limits derivation are given in the Technical Manual.

The physical restraints that are built into each facility and the Operating Procedures are designed to ensure that the limits will not be exceeded in the event of operating contingencies that are deemed to be credible. Such credible contingencies depend upon the particular facility for handling fissile components.

REFERENCES

1. DPSTM-100 CRIT-85. Technical Manual for 100-Area Criticality Safety. (U)
 2. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 9.1.2, "Irradiated Fuel, Irradiated Target, and Component Storage." (U)
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B 3.9 COMPONENTS HANDLING

INFORMATION ONLY

B 3.9.4.1 Disassembly Basin Operation - Water Temperature

BASES

Water temperature higher than 45°C would result in less effective cooling of irradiated fuel and could cause accelerated cladding corrosion and cladding melting. The limits on heat generation of assemblies discharged from the reactor are based on basin water temperature $\leq 45^{\circ}\text{C}$ (Ref. 1). Increased water temperature can also increase tritium release from the basin water to the air.

REFERENCES

1. DPST-73-259, H. E. Wingo and M. W. Lewis, Heat Removal from Reactor Assemblies in the Basin, March 19, 1973. (U)
 2. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 9.1.3, "Disassembly Basin Cooling and Cleanup System." (U)
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B 3.9 COMPONENTS HANDLING

B 3.9.4.2 Disassembly Basin Operation - Water Level

BASES

Basin water level outside of the specified range indicates an abnormal condition or equipment failure that requires corrective action to avert the following undesirable consequences:

- a) Activity release due to basin overflow (overflow to the process sewers occurs with water level at -11.5 inches).
- b) Inadequate water supply for the Confinement Heat Removal (CHR) system.

The CHR system is designed to operate with the disassembly basin water level $\geq$ -18 inches.

Establishing the upper limit of the disassembly basin water level at -11.5 inches provides some margin to prevent overflow to the process sewers.

REFERENCES

- 1. DPST-73-259, H. E. Wingo and M. W. Lewis, Heat Removal from Reactor Assemblies in the Basin, March 19, 1973. (U)
- 2. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 9.1.3, "Disassembly Basin Cooling and Cleanup System." (U)

INFORMATION ONLY

B 3.9 COMPONENTS HANDLING

B 3.9.5 Charge and Discharge Operations - Process Water to Cooling Water AT

BASES

The flow must be sufficient to prevent bulk boiling in any assembly because bulk boiling could stop flow through the assembly and cause overheating. Bulk boiling would occur with an assembly effluent temperature of about 113°C. The 80°C limit gives ample margin from boiling to enable an overtemperature to be detected and corrected by normal procedures such as opening rotovalves or starting another pump. Incident action probably would not be necessary to prevent bulk boiling in an assembly. The effluent temperature must also be low enough to accommodate the flow reduction resulting from loss of one pump without bulk boiling in any assembly. Again, this would provide some margin to avoid bulk boiling by opening rotovalves or starting other pumps, and would not necessarily require incident action.

Flow through a reactor assembly can be reduced momentarily when an auxiliary orifice is removed from the assembly by the discharge machine prior to discharge of the assembly. Flow is decreased when the largest diameter section of the orifice assembly passes the coolant inlet holes in the universal sleeve housing (USH). This momentary flow reduction has no adverse effect. However, should the discharge machine stall at the worst possible elevation, flow would equilibrate at about 6 gpm for an assembly with the worst combination of USH/orifice dimensional tolerances, and located at the minimum flow position in the reactor (Ref. 6). Six gpm is adequate for decay heat removal but is marginally lower than the nominal minimum of 9 gpm for good temperature monitoring. This is a very low probability occurrence.

Experiments and calculations indicate that irradiated fuel assemblies can be adequately cooled by natural convection after the heat generation rate has decreased sufficiently, provided that requirements a and b are met.

(continued)

Charge and Discharge Operations - Process Water to Cooling Water ΔT
B 3.9.5

BASES

(continued)

- a) The bulk temperature of the process water must be low enough to:
 - 1) Provide an adequate temperature differential between the coolant in the assemblies and the process water.
 - 2) Provide an adequate heat sink capacity. The Mark 22 fuel maximum allowable heat generation of 6 kW is based on a bulk moderator temperature of 50°C. Assumption of higher bulk process water temperatures would result in lower acceptable heat generation values and would decrease the system heat capacity.
- b) Natural convection cooling of assemblies is dependent upon circulation of coolant from the inner coolant annuli to the outer housing. This circulation is blocked if the liquid level in the reactor tank falls sufficiently.

Cessation of charge-discharge operations, when the bulk moderator temperature is more than 5°C above the cooling water temperature, is required to limit potential reactivity addition should process water flow be restarted. Reactivity added by charge-discharge operations could be masked by reactivity held in negative process water temperature coefficient, and an unexpected reactivity increase could occur when flow is restarted. When the bulk moderator temperature is within 5°C of the temperature of the cooling water supply to the heat exchangers, however, the potential reactivity increase through temperature coefficient is small, <0.002 k, and charge-discharge operations are permitted. This restriction applies whether or not the reactor contains irradiated assemblies.

An increase in cooling water flow could decrease the process water temperature and add reactivity because of the negative temperature coefficient. LCO 3.9.5 limits the potential temperature decrease to <5°C, corresponding to <0.002 k, during charge-discharge operations for the same reason outlined in a)1 above.

(continued)

INFORMATION ONLY

BASES (continued)

REFERENCES

1. DPSTM-16 (H). Technical Manual - Hydraulics and Heat Transfer of Mark 16 Fuel Assemblies. (UCNI)
 2. DPSTM-18 (H). Technical Manual - Hydraulics and Heat Transfer of Mark 18 Fuel Assemblies. (U)
 3. DPSTM-22 (H). Technical Manual - Hydraulics and Heat Transfer of Mark 22 Fuel Assemblies. (U)
 4. DPSTM-31 (30) (H). Technical Manual - Hydraulics and Heat Transfer of Mark 30 Target Assemblies. (UCNI)
 5. DPSTM-Targets (H). Technical Manual - Hydraulics and Heat Transfer of Target Assemblies. (SECRET/RD)
 6. DPST-74-324. J. P. Faraci, letter to L. W. Fox, Auxiliary Orifice Assembly Design, April 10, 1974. (U)
 7. DPST-TS-105-2.7. Normal Cooling System. (U)
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B 3.9 COMPONENTS HANDLING

INFORMATION ONLY

B 3.9.6 Discharge Machine Water Sources

BASES

During discharge from the reactor, the maximum temperature of irradiated fuel and target assemblies is limited to 2/3 of the centigrade temperature at which abrupt changes in physical properties could occur. This temperature limit, usually 385°C, can be exceeded should transport time be longer than normal; hence operability of the cooling system is required.

A source of process water is required so that assembly cooling can be initiated while the discharge machine is over an open reactor position without diluting the process water in the reactor with cooling water. A normal and a backup source of cooling water are required for possible long-term cooling.

Minimum flow rates of 8 gpm for the primary source and 13 gpm for the secondary source are specified because these values represent current system capabilities. Assembly geometry and limits on maximum decay power are designed so that adequate cooling is provided at the specified flow rate.

REFERENCES

1. DPSTM-22 (H), Technical Manual - Hydraulics and Heat Transfer of Mark 22 Assemblies. (U)
2. DPST-86-269, "Discharge Machine Coolant Distribution in Current Production Assemblies," March 19, 1986. (U)
3. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 9.1.4, "Light Load Handling Systems (Related to Refueling)." (U)

B 3.9 COMPONENTS HANDLING

B 3.9.8 Charge and Discharge Equipment Power

BASES

Power shall be provided to permit discharge of irradiated components from the reactor in the event of loss of offsite power. This capability is necessary for coping with the various types of incidents considered in the operation of the reactors. Additionally, the capability to complete component transfer in progress in the event of loss of offsite power is provided by the required power source.

Backup power required to permit charge or discharge of components is provided by the Rod Equipment Room (RER) 125 volt floating emergency battery. This battery is charged by two parallel Rod Equipment Room (RER) motor-generator sets. In the event of loss of offsite power the battery provides power the required equipment via either of two parallel motor-alternator sets. LCO 3.9.8 requires that the RER battery, one RER motor-generator set, and one RER motor-alternator set must be OPERABLE.

REFERENCES

1. WSRC-SAR-RX-K, Savannah River Site Production Reactor Safety Analysis Report, Section 8.0, "Electric Power." (U)
 2. DPST-70-4197. G. F. Merz, et. al., Analysis of the Savannah River Protective Instrument System, October 1970. (UCNI)
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B 3.9 COMPONENTS HANDLING

INFORMATION ONLY

B 3.9.9 Emergency Cooling Water Pump Well

BASES

BACKGROUND

The minimum Confinement Heat Removal System (CHRS) Disassembly Basin Water requirements are based on water availability from all sections of the Disassembly Basin (excluding the Vertical Tube Storage (VTS) area and the D&E canal).

The emergency pump suction well sluice gate and the stop logs connecting the Disassembly Basin dry cave area, machine area, emergency disassembly, monitor basin, bucket storage, and the transfer pit are required to be open whenever the reactor contains HGAs.

REFERENCES

1. DPST-73-259, H. E. Wingo and M. W. Lewis, Heat Removal from Reactor Assemblies in the Basin, March 19, 1973.
(U)
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B 3.10 SPECIAL TEST EXCEPTIONS

B.3.10.1 Safety Rod Testing

INFORMATION ONLY

BASES

BACKGROUND

The Safety Rod System provides negative reactivity addition to the reactor core capable of rapid REACTOR SHUTDOWN within safety analysis limits to prevent fuel damage during all analyzed transients requiring termination by a scram. In order to perform design basis functions, the safety rods must provide a subcritical margin of at least 1% Δk (shutdown margin prior to post shutdown Xenon decay) under all operational conditions ranging from cold conditions to hot FULL POWER operation at all stages of fuel charge life.

SR 3.1.2.3 demonstrates that during a reactor scram under simulated operating conditions, the safety rods will be inserted into the reactor to the extent and within the time assumed in the safety analysis. SR 3.1.2.4 demonstrates that the safety rods will travel to the bottom limit position within 30 seconds after scram relay de-energization.

The current Technical Specification reactor states (i.e. REACTOR STARTUP, REACTOR OPERATION) are defined in such a way (based on safety rod position) as to require multiple reactor state changes during performance of safety rod testing per SR 3.1.2.3 or SR 3.1.2.4. Each reactor state change requires numerous OPERABILITY determinations and/or surveillance test performances.

APPLICABLE
SAFETY ANALYSES

Safety rod OPERABILITY ensures that sufficient negative reactivity is available to shut down the reactor following a reactor scram from FULL POWER and maintain the reactor subcritical during the resulting temperature decrease. The requirement to have all full length and partial length control rods FULLY INSERTED ensures that subcriticality will be maintained as Xenon decays.

(continued)

BASES (continued)

LCOs

LCO 3.10.1

During performance and upon successful completion of safety rod testing per SR 3.1.2.3 or SR 3.1.2.4, the reactor shall be considered in REACTOR STARTUP for all APPLICABILITY determinations provided the control rods remain FULLY INSERTED and the testing is completed within 7 days. Requiring the control rods to be inserted with the control rod drive power supplies de-energized ensures the reactor stays subcritical and precludes the need to do testing associated with only REACTOR OPERATION. With the reactor subcritical during performance of the safety rod testing, no heat generation nor fission product accumulation within the fuel will result. Since the reactor would still be considered in REACTOR STARTUP, all the surveillances associated with REACTOR STARTUP would continue to be performed thereby bounding any surveillances that would have been required by entry into REACTOR SHUTDOWN during performance of safety rod testing per SR 3.1.2.3 or SR 3.1.2.4. The test completion time limit ensures the test is completed in a timely manner and prevents extended time spent in REACTOR STARTUP.

APPLICABILITY

This Special Test Exception is for the Technical Specification required safety rod testing and is only applicable during the performance of safety rod testing per SR 3.1.2.3 or SR 3.1.2.4.

ACTIONS

A.1

Condition A is entered when any full length or partial length control rod is not FULLY INSERTED. Required Action A.1 is to IMMEDIATELY insert all safety rods, with the purpose of adding negative reactivity.

B.1

Condition B is entered when all control rod drive power supplies are not de-energized. Required Action B.1 is to verify that all control rods are FULLY INSERTED and that the control rod drive power supplies are de-energized, with the purpose of assuring the control rods remain inserted and the reactor stays subcritical.

(continued)

NEC ACTION ONLY

BASES (continued)

**ACTIONS
(continued)**

C.1

Condition C is entered when the total test completion time is greater than 7 days. Safety rod testing should be completed in a timely manner so that excessive time is not spent in REACTOR STARTUP. Required Action C.1 is to be in REACTOR SHUTDOWN within 90 minutes, precluding a longer than desired stay in REACTOR STARTUP. The 90 minute completion time is required to facilitate a CONTROLLED REACTOR SHUTDOWN and is not provided to attempt to complete the safety rod tests.

**SURVEILLANCE
REQUIREMENTS**

SR 3.10.1.1

This Surveillance Requirement verifies that all full length and all partial length control rods are FULLY INSERTED, ensuring that the control rods' contribution to the shutdown margin is available and maintained.

SR 3.10.1.2

This Surveillance Requirement verifies that the control rod drive power supplies are de-energized, ensuring that the control rods' contribution to the shutdown margin is maintained.

SR 3.10.1.3

This Surveillance Requirement verifies that safety rod testing has been completed per the requirements of SR 3.1.2.3 or SR 3.1.2.4 within the specified 7 day period.

REFERENCES

1. WSRC-SAR-K, Savannah River Site K Production Reactor Safety Analysis Report, Section 7.2, "Reactor Trip System."
 2. WSRC-SAR-K, Savannah River Site K Production Reactor Safety Analysis Report, Section 7.7, "Safety and Control Rod Drive Systems and Control Computer."
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