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1.0 USE AND APPLICATION

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 Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)	The APLHGR shall be applicable to a specific planar height and is equal to the sum of the LHGRs for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle at the height.
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, display, and trip functions, and shall include the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an in-place qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is calibrated.
CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

(continued)

1.1 Definitions (continued)

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm, interlock, display, and trip functions, and channel failure trips. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.

CORE ALTERATION

CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS:

- a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and
- b. Control rod movement, provided there are no fuel assemblies in the associated core cell.

Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT (COLR)

The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for

(continued)

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)

Power and Test Reactor Sites;" Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977; or ICRP 30, Supplement to Part 1, page 192-212, Table titled "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

END OF CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) SYSTEM RESPONSE TIME

The EOC-RPT SYSTEM RESPONSE TIME shall be that time interval from initial signal generation by the associated turbine throttle valve limit switch or from when the turbine governor valve hydraulic control oil pressure drops below the pressure switch setpoint to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

ISOLATION SYSTEM RESPONSE TIME

The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves travel to their required positions. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

(continued)

1.1 Definitions (continued)

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE into the drywell such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or
2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE;

b. Unidentified LEAKAGE

All LEAKAGE into the drywell that is not identified LEAKAGE;

c. Total LEAKAGE

Sum of the identified and unidentified LEAKAGE; and

d. Pressure Boundary LEAKAGE

LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.

LINEAR HEAT GENERATION
RATE (LHGR)

The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL
TEST

A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all required logic components (i.e., all required relays and contacts, trip units, solid state logic elements, etc.) of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.

(continued)



1.1 Definitions (continued)

MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)	The MFLPD shall be the largest value of the fraction of limiting power density (FLPD) in the core. The FLPD shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.
MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core for each class of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE—OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	<p>PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:</p> <ul style="list-style-type: none">a. Described in Chapter 14, Initial Test Program of the FSAR;b. Authorized under the provisions of 10 CFR 50.59; orc. Otherwise approved by the Nuclear Regulatory Commission.

(continued)

1.1 Definitions (continued)

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3486 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	<p>SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that:</p> <ol style="list-style-type: none">The reactor is xenon free;The moderator temperature is 68°F; andAll control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during n Surveillance Frequency intervals, where n is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TURBINE BYPASS SYSTEM RESPONSE TIME	The TURBINE BYPASS SYSTEM RESPONSE TIME shall be the time from when the turbine bypass control unit generates a turbine bypass valve flow signal until 80% of the turbine bypass capacity is established.

(continued)

1.1 Definitions

TURBINE BYPASS SYSTEM
RESPONSE TIME
(continued)

The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTOR MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	Run	NA
2	Startup	Refuel ^(a) or Startup/Hot Standby	NA
3	Hot Shutdown ^(a)	Shutdown	> 200
4	Cold Shutdown ^(a)	Shutdown	≤ 200
5	Refueling ^(b)	Shutdown or Refuel	NA

(a) All reactor vessel head closure bolts fully tensioned.

(b) One or more reactor vessel head closure bolts less than fully tensioned.



1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE	<p>The purpose of this section is to explain the meaning of logical connectors.</p> <p>Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are <u>AND</u> and <u>OR</u>. The physical arrangement of these connectors constitutes logical conventions with specific meanings.</p>
BACKGROUND	<p>Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors.</p> <p>When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.</p>
EXAMPLES	<p>The following examples illustrate the use of logical connectors.</p>

(continued)

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . <u>AND</u> A.2 Restore . . .	

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

(continued)

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip . . . <u>OR</u> A.2.1 Verify . . . <u>AND</u> A.2.2.1 Reduce . . . <u>OR</u> A.2.2.2 Perform . . . <u>OR</u> A.3 Align . . .	

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

1.0 USE AND APPLICATION

1.3 Completion Times

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

BACKGROUND	Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).
------------	--

DESCRIPTION	<p>The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability.</p> <p>If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition.</p> <p>Once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will <u>not</u> result in separate entry into the Condition unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition. However, when a <u>subsequent</u> division, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:</p>
-------------	--

(continued)

1.3 Completion Times

DESCRIPTION
(continued)

- a. Must exist concurrent with the first inoperability; and
- b. Must remain inoperable or not within limits after the first inoperability is resolved.

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

- a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or
- b. The stated Completion Time as measured from discovery of the subsequent inoperability.

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

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ." Example 1.3-3 illustrates one use of this type of Completion Time. The 10 day Completion Time specified for Conditions A and B in Example 1.3-3 may not be extended.

(continued)

1.3 Completion Times (continued)

EXAMPLES

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

EXAMPLE 1.3-1

ACTIONS

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

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

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

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

(continued)

1000

1000



1.3 Completion Times

EXAMPLES (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 MODE 3.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	36 hours

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

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

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

(continued)

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1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-2 (continued)

While in LCO 3.0.3, if 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 accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

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

(continued)

1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-3

ACTIONS

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

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-3 (continued)

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

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

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

(continued)

2 5 2 4



2 5 2 4

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-4

ACTIONS

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

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

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

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

(continued)

1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-5

ACTIONS

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

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

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

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

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-5 (continued)

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

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

EXAMPLE 1.3-6

ACTIONS

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

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-6 (continued)

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

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

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-7

ACTIONS

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

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

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

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-7 (continued)

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

IMMEDIATE COMPLETION TIME

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

1.0 USE AND APPLICATION

1.4 Frequency

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

DESCRIPTION	<p>Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated Limiting Conditions for Operation (LCO). An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.</p> <p>The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, "Surveillance Requirement (SR) Applicability." The "specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements.</p> <p>Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillance, or both. Example 1.4-4 discusses these special situations.</p> <p>Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.</p> <p>The use of "met" or "performed" in these instances conveys specified meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not "met." "Performance" refers only to the requirement to specifically determine the ability to meet the acceptance criteria. SR 3.0.4 restrictions would not apply if both the following conditions are satisfied:</p>
-------------	---

(continued)



1.4 Frequency

DESCRIPTION (continued)

- a. The Surveillance is not required to be performed; and
- b. The Surveillance is not required to be met or, even if required to be met, is not known to be failed.

EXAMPLES

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

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the interval specified in the Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the unit is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Examples 1.4-3 and 1.4-4), then SR 3.0.3 becomes applicable.

(continued)



1.4 Frequency

EXAMPLES

EXAMPLE 1.4-1 (continued)

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

EXAMPLE 1.4-2SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after $\geq 25\%$ RTP <u>AND</u> 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level $< 25\%$ RTP to $\geq 25\%$ RTP, the Surveillance must be performed within 12 hours.

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

(continued)

1. 2. 3.

4. 5. 6.

7. 8. 9.

10. 11. 12.



13. 14. 15.

16. 17. 18.

19. 20. 21.

22. 23. 24.

25. 26. 27.



28. 29. 30.

31. 32. 33.

34. 35. 36.

37. 38. 39.

40. 41. 42.

43. 44. 45.

46. 47. 48.

49. 50. 51.



1.4 Frequency

EXAMPLES

EXAMPLE 1.4-2 (continued)

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

EXAMPLE 1.4-3SURVEILLANCE REQUIREMENTS

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

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

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

(continued)

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-3 (continued)

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

EXAMPLE 1.4-4SURVEILLANCE REQUIREMENTS

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

Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the unit was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow $< 10\%$ rated core flow:

THERMAL POWER shall be $\leq 25\%$ RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow $\geq 10\%$ rated core flow:

MCPR shall be ≥ 1.07 for two recirculation loop operation or ≥ 1.08 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1 LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2 and LCO 3.0.7.

LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

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

LCO 3.0.3 When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

- a. MODE 2 within 7 hours;
- b. MODE 3 within 13 hours; and
- c. MODE 4 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

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

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

LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

(continued)

3.0 LCO APPLICABILITY

LCO 3.0.4
(continued) Exceptions to this Specification are stated in the individual Specifications. These exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered allow unit operation in the MODE or other specified condition in the Applicability only for a limited period of time.

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

LCO 3.0.5 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

LCO 3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, additional evaluations and limitations may be required in accordance with Specification 5.5.11, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

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

(continued)

3.0 LCO APPLICABILITY (continued)

LCO 3.0.7 Special Operations LCOs in Section 3.10 allow specified Technical Specifications (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Special Operations LCOs is optional. When a Special Operations LCO is desired to be met but is not met, the ACTIONS of the Special Operations LCO shall be met. When a Special Operations LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with the other applicable Specifications.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1 SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 3.0.2 The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

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

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

Exceptions to this Specification are stated in the individual Specifications.

SR 3.0.3 If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance.

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

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

(continued)

3.0 SR APPLICABILITY (continued)

SR 3.0.4 Entry into a MODE or other specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 SDM shall be:

- a. $\geq 0.38\% \Delta k/k$, with the highest worth control rod analytically determined; or
- b. $\geq 0.28\% \Delta k/k$, with the highest worth control rod determined by test.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limits in MODE 1 or 2.	A.1 Restore SDM to within limits.	6 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	12 hours
C. SDM not within limits in MODE 3.	C.1 Initiate action to fully insert all insertable control rods.	Immediately
D. SDM not within limits in MODE 4.	D.1 Initiate action to fully insert all insertable control rods. <u>AND</u>	Immediately (continued)

D ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. (continued)	D.2 Initiate action to restore secondary containment to OPERABLE status.	1 hour
	<u>AND</u>	
	D.3 Initiate action to restore one standby gas treatment (SGT) subsystem to OPERABLE status.	1 hour
	<u>AND</u>	
	D.4 Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated.	1 hour
E. SDM not within limits in MODE 5.	E.1 Suspend CORE ALTERATIONS except for control rod insertion and fuel assembly removal.	Immediately
	<u>AND</u>	
	E.2 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
	<u>AND</u>	
		(continued)



ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. (continued)	E.3 Initiate action to restore secondary containment to OPERABLE status.	1 hour
	<u>AND</u>	
	E.4 Initiate action to restore one SGT subsystem to OPERABLE status.	1 hour
	<u>AND</u>	
	E.5 Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.1.1.1 Verify SDM is:</p> <p>a. $\geq 0.38\% \Delta k/k$ with the highest worth control rod analytically determined; or</p> <p>b. $\geq 0.28\% \Delta k/k$ with the highest worth control rod determined by test.</p>	<p>Prior to each in vessel fuel movement during fuel loading sequence</p> <p><u>AND</u></p> <p>Once within 4 hours after criticality following fuel movement within the reactor pressure vessel or control rod replacement</p>



3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Reactivity Anomalies

LCO 3.1.2 The reactivity difference between the monitored core k_{eff} and the predicted core k_{eff} shall be within $\pm 1\% \Delta k/k$.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Core reactivity difference not within limit.	A.1 Restore core reactivity difference to within limit.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.1.2.1 Verify core reactivity difference between the monitored core k_{eff} and the predicted core k_{eff} is within $\pm 1\% \Delta k/k$.</p>	<p>Once within 24 hours after reaching equilibrium conditions following startup after fuel movement within the reactor pressure vessel or control rod replacement</p> <p><u>AND</u></p> <p>1000 MWD/T thereafter during operations in MODE 1</p>

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Control Rod OPERABILITY

LCO 3.1.3 Each control rod shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each control rod.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One withdrawn control rod stuck.	-----NOTE----- Rod Worth Minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation," if required, to allow continued operation. -----	
	A.1 Verify stuck control rod separation criteria are met.	Immediately
	<u>AND</u>	
	A.2 Disarm the associated control rod drive (CRD).	2 hours
	<u>AND</u>	
		(continued)



ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 Perform SR 3.1.3.2 and SR 3.1.3.3 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM
	<u>AND</u> A.4 Perform SR 3.1.1.1.	72 hours
B. Two or more withdrawn control rods stuck.	B.1 Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	C.1 Verify the total number of "slow" and inoperable control rods is \leq eight.	Immediately
	<u>AND</u>	(continued)

1. *Chlorophyll a* (Chl *a*)

5 6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	<p>C.2 -----NOTE----- RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of inoperable control rod and continued operation. -----</p> <p>Fully insert inoperable control rod.</p>	3 hours
	<p><u>AND</u></p> <p>C.3 Disarm the associated CRD.</p>	4 hours
D. -----NOTE----- Not applicable when THERMAL POWER > 10% RTP. ----- Two or more inoperable control rods not in compliance with banked position withdrawal sequence (BPWS) and not separated by two or more OPERABLE control rods.	<p>D.1 Restore compliance with BPWS.</p>	4 hours
	<p><u>OR</u></p> <p>D.2 Restore control rod to OPERABLE status.</p>	4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>E. -----NOTE----- Not applicable when THERMAL POWER > 10% RTP. -----</p> <p>One or more groups with four or more inoperable control rods.</p>	<p>E.1 Restore the control rod to OPERABLE status.</p>	<p>4 hours</p>
<p>F. Required Action and associated Completion Time of Condition A, C, D, or E not met.</p> <p><u>OR</u></p> <p>Nine or more control rods inoperable.</p>	<p>F.1 Be in MODE 3.</p>	<p>12 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.3.1 Determine the position of each control rod.	24 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.1.3.2 -----NOTE----- Not required to be performed until 7 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM. ----- Insert each fully withdrawn control rod at least one notch.</p>	<p>7 days</p>
<p>SR 3.1.3.3 -----NOTE----- Not required to be performed until 31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM. ----- Insert each partially withdrawn control rod at least one notch.</p>	<p>31 days</p>
<p>SR 3.1.3.4 Verify each control rod scram time from fully withdrawn to notch position 5 is ≤ 7 seconds.</p>	<p>In accordance with SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.1.3.5 Verify each control rod does not go to the withdrawn overtravel position.	Each time the control rod is withdrawn to "full out" position <u>AND</u> Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect coupling

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Control Rod Scram Times

LCO 3.1.4 The average scram time of all OPERABLE control rods in all two-by-two arrays shall not exceed the limits of Table 3.1.4-1.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each two-by-two array.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more two-by-two arrays with average scram time not within the limits of Table 3.1.4-1.	A.1 Declare each control rod in the two-by-two array with a scram time slower than the average scram time limits "slow."	Immediately
	<u>AND</u>	
	A.2 Verify the total number of "slow" and inoperable control rods is \leq eight.	Immediately
	<u>AND</u>	
	A.3 Verify each "slow" control rod meets the "slow" control rod separation criteria.	Immediately
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----
During single control rod scram time Surveillances, the control rod drive (CRD) pumps shall be isolated from the associated scram accumulator.

SURVEILLANCE	FREQUENCY
SR 3.1.4.1 Verify each control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure \geq 800 psig.	Prior to exceeding 40% RTP after each refueling <u>AND</u> Prior to exceeding 40% RTP after each reactor shutdown \geq 120 days
SR 3.1.4.2 Verify, for a representative sample, each tested control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure \geq 800 psig.	120 days cumulative operation in MODE 1
SR 3.1.4.3 Verify each affected control rod scram time is within the limits of Table 3.1.4-1 with any reactor steam dome pressure.	Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect scram time

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.1.4.4 Verify each affected control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure \geq 800 psig.	Prior to exceeding 40% RTP after work on control rod or CRD System that could affect scram time <u>AND</u> Prior to exceeding 40% RTP after fuel movement within the reactor pressure vessel

Table 3.1.4-1
Control Rod Scram Times

-----NOTE-----
Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position 5. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

NOTCH POSITION	SCRAM TIMES(a)(b) (seconds) WHEN REACTOR STEAM DOME PRESSURE ≥ 800 psig
45	0.430
39	0.868
25	1.936
5	3.497

- (a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids as time zero.
- (b) Scram times as a function of reactor steam dome pressure, when < 800 psig, are within established limits.

11 12 13 14 15 16 17 18 19 20 21 22 23 24 25 26 27 28 29 30 31 32 33 34 35 36 37 38 39 40 41 42 43 44 45 46 47 48 49 50 51 52 53 54 55 56 57 58 59 60 61 62 63 64 65 66 67 68 69 70 71 72 73 74 75 76 77 78 79 80 81 82 83 84 85 86 87 88 89 90 91 92 93 94 95 96 97 98 99 100 101 102 103 104 105 106 107 108 109 110 111 112 113 114 115 116 117 118 119 120 121 122 123 124 125 126 127 128 129 130 131 132 133 134 135 136 137 138 139 140 141 142 143 144 145 146 147 148 149 150 151 152 153 154 155 156 157 158 159 160 161 162 163 164 165 166 167 168 169 170 171 172 173 174 175 176 177 178 179 180 181 182 183 184 185 186 187 188 189 190 191 192 193 194 195 196 197 198 199 200 201 202 203 204 205 206 207 208 209 210 211 212 213 214 215 216 217 218 219 220 221 222 223 224 225 226 227 228 229 230 231 232 233 234 235 236 237 238 239 240 241 242 243 244 245 246 247 248 249 250 251 252 253 254 255 256 257 258 259 260 261 262 263 264 265 266 267 268 269 270 271 272 273 274 275 276 277 278 279 280 281 282 283 284 285 286 287 288 289 290 291 292 293 294 295 296 297 298 299 300 301 302 303 304 305 306 307 308 309 310 311 312 313 314 315 316 317 318 319 320 321 322 323 324 325 326 327 328 329 330 331 332 333 334 335 336 337 338 339 340 341 342 343 344 345 346 347 348 349 350 351 352 353 354 355 356 357 358 359 360 361 362 363 364 365 366 367 368 369 370 371 372 373 374 375 376 377 378 379 380 381 382 383 384 385 386 387 388 389 390 391 392 393 394 395 396 397 398 399 400 401 402 403 404 405 406 407 408 409 410 411 412 413 414 415 416 417 418 419 420 421 422 423 424 425 426 427 428 429 430 431 432 433 434 435 436 437 438 439 440 441 442 443 444 445 446 447 448 449 450 451 452 453 454 455 456 457 458 459 460 461 462 463 464 465 466 467 468 469 470 471 472 473 474 475 476 477 478 479 480 481 482 483 484 485 486 487 488 489 490 491 492 493 494 495 496 497 498 499 500 501 502 503 504 505 506 507 508 509 510 511 512 513 514 515 516 517 518 519 520 521 522 523 524 525 526 527 528 529 530 531 532 533 534 535 536 537 538 539 540 541 542 543 544 545 546 547 548 549 550 551 552 553 554 555 556 557 558 559 560 561 562 563 564 565 566 567 568 569 570 571 572 573 574 575 576 577 578 579 580 581 582 583 584 585 586 587 588 589 590 591 592 593 594 595 596 597 598 599 600 601 602 603 604 605 606 607 608 609 610 611 612 613 614 615 616 617 618 619 620 621 622 623 624 625 626 627 628 629 630 631 632 633 634 635 636 637 638 639 640 641 642 643 644 645 646 647 648 649 650 651 652 653 654 655 656 657 658 659 660 661 662 663 664 665 666 667 668 669 670 671 672 673 674 675 676 677 678 679 680 681 682 683 684 685 686 687 688 689 690 691 692 693 694 695 696 697 698 699 700 701 702 703 704 705 706 707 708 709 710 711 712 713 714 715 716 717 718 719 720 721 722 723 724 725 726 727 728 729 730 731 732 733 734 735 736 737 738 739 740 741 742 743 744 745 746 747 748 749 750 751 752 753 754 755 756 757 758 759 760 761 762 763 764 765 766 767 768 769 770 771 772 773 774 775 776 777 778 779 780 781 782 783 784 785 786 787 788 789 790 791 792 793 794 795 796 797 798 799 800 801 802 803 804 805 806 807 808 809 810 811 812 813 814 815 816 817 818 819 820 821 822 823 824 825 826 827 828 829 830 831 832 833 834 835 836 837 838 839 840 841 842 843 844 845 846 847 848 849 850 851 852 853 854 855 856 857 858 859 860 861 862 863 864 865 866 867 868 869 870 871 872 873 874 875 876 877 878 879 880 881 882 883 884 885 886 887 888 889 890 891 892 893 894 895 896 897 898 899 900 901 902 903 904 905 906 907 908 909 910 911 912 913 914 915 916 917 918 919 920 921 922 923 924 925 926 927 928 929 930 931 932 933 934 935 936 937 938 939 940 941 942 943 944 945 946 947 948 949 950 951 952 953 954 955 956 957 958 959 960 961 962 963 964 965 966 967 968 969 970 971 972 973 974 975 976 977 978 979 980 981 982 983 984 985 986 987 988 989 990 991 992 993 994 995 996 997 998 999 1000 1001 1002 1003 1004 1005 1006 1007 1008 1009 1010 1011 1012 1013 1014 1015 1016 1017 1018 1019 1020 1021 1022 1023 1024 1025 1026 1027 1028 1029 1030 1031 1032 1033 1034 1035 1036 1037 1038 1039 1040 1041 1042 1043 1044 10

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Control Rod Scram Accumulators

LCO 3.1.5 Each control rod scram accumulator shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each control rod scram accumulator.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One control rod scram accumulator inoperable with reactor steam dome pressure ≥ 900 psig.	<p>A.1 -----NOTE----- Only applicable if the average scram times of the two-by-two arrays associated with the control rod with the inoperable accumulator are within the limits of Table 3.1.4-1 during the last scram time Surveillance. -----</p> <p>Declare the average scram time in all two-by-two arrays associated with the control rod with the inoperable accumulator not within the limits of Table 3.1.4-1 and declare the associated control rod "slow."</p> <p><u>OR</u></p>	<p>8 hours</p> <p>(continued)</p>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Declare the associated control rod inoperable.	8 hours
B. Two or more control rod scram accumulators inoperable with reactor steam dome pressure ≥ 900 psig.	<p>B.1 Restore charging water header pressure to ≥ 940 psig.</p> <p><u>AND</u></p> <p>B.2.1 -----NOTE----- Only applicable if the average scram times of the two-by-two arrays associated with the control rod with the inoperable accumulator are within the limits of Table 3.1.4-1 during the last scram time Surveillance. -----</p> <p>Declare the average scram time in all two-by-two arrays associated with the control rod with the inoperable accumulator not within the limits of Table 3.1.4-1 and declare the associated control rod "slow."</p> <p><u>OR</u></p>	<p>20 minutes from discovery of Condition B concurrent with charging water header pressure < 940 psig</p> <p>1 hour</p> <p>(continued)</p>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2.2 Declare the associated control rod inoperable.	1 hour
C. One or more control rod scram accumulators inoperable with reactor steam dome pressure < 900 psig.	C.1 Verify the associated control rod is fully inserted.	Immediately upon discovery of charging water header pressure < 940 psig
	<u>AND</u> C.2 Declare the associated control rod inoperable.	1 hour
D. Required Action B.1 or C.1 and associated Completion Time not met. ✓	D.1 -----NOTE----- Not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods. ----- Place the reactor mode switch in the shutdown position.	Immediately

$\frac{d}{dt} \left(\frac{\partial L}{\partial \dot{x}} \right) = \frac{\partial L}{\partial x}$

1. 1848 1849 1850 1851 1852 1853 1854 1855 1856 1857 1858 1859 1860 1861 1862 1863 1864 1865 1866 1867 1868 1869 1870 1871 1872 1873 1874 1875 1876 1877 1878 1879 1880 1881 1882 1883 1884 1885 1886 1887 1888 1889 1890 1891 1892 1893 1894 1895 1896 1897 1898 1899 1900 1901 1902 1903 1904 1905 1906 1907 1908 1909 1910 1911 1912 1913 1914 1915 1916 1917 1918 1919 1920 1921 1922 1923 1924 1925 1926 1927 1928 1929 1930 1931 1932 1933 1934 1935 1936 1937 1938 1939 1940 1941 1942 1943 1944 1945 1946 1947 1948 1949 1950 1951 1952 1953 1954 1955 1956 1957 1958 1959 1960 1961 1962 1963 1964 1965 1966 1967 1968 1969 1970 1971 1972 1973 1974 1975 1976 1977 1978 1979 1980 1981 1982 1983 1984 1985 1986 1987 1988 1989 1990 1991 1992 1993 1994 1995 1996 1997 1998 1999 2000 2001 2002 2003 2004 2005 2006 2007 2008 2009 2010 2011 2012 2013 2014 2015 2016 2017 2018 2019 2020 2021 2022 2023 2024 2025 2026 2027 2028 2029 2030 2031 2032 2033 2034 2035 2036 2037 2038 2039 2040 2041 2042 2043 2044 2045 2046 2047 2048 2049 2050 2051 2052 2053 2054 2055 2056 2057 2058 2059 2060 2061 2062 2063 2064 2065 2066 2067 2068 2069 2070 2071 2072 2073 2074 2075 2076 2077 2078 2079 2080 2081 2082 2083 2084 2085 2086 2087 2088 2089 2090 2091 2092 2093 2094 2095 2096 2097 2098 2099 2100 2101 2102 2103 2104 2105 2106 2107 2108 2109 2110 2111 2112 2113 2114 2115 2116 2117 2118 2119 2120 2121 2122 2123 2124 2125 2126 2127 2128 2129 2130 2131 2132 2133 2134 2135 2136 2137 2138 2139 2140 2141 2142 2143 2144 2145 2146 2147 2148 2149 2150 2151 2152 2153 2154 2155 2156 2157 2158 2159 2160 2161 2162 2163 2164 2165 2166 2167 2168 2169 2170 2171 2172 2173 2174 2175 2176 2177 2178 2179 2180 2181 2182 2183 2184 2185 2186 2187 2188 2189 2190 2191 2192 2193 2194 2195 2196 2197 2198 2199 2200 2201 2202 2203 2204 2205 2206 2207 2208 2209 2210 2211 2212 2213 2214 2215 2216 2217 2218 2219 2220 2221 2222 2223 2224 2225 2226 2227 2228 2229 2230 2231 2232 2233 2234 2235 2236 2237 2238 2239 2240 2241 2242 2243 2244 2245 2246 2247 2248 2249 2250 2251 2252 2253 2254 2255 2256 2257 2258 2259 2260 2261 2262 2263 2264 2265 2266 2267 2268 2269 2270 2271 2272 2273 2274 2275 2276 2277 2278 2279 2280 2281 2282 2283 2284 2285 2286 2287 2288 2289 2290 2291 2292 2293 2294 2295 2296 2297 2298 2299 2300 2301 2302 2303 2304 2305 2306 2307 2308 2309 2310 2311 2312 2313 2314 2315 2316 2317 2318 2319 2320 2321 2322 2323 2324 2325 2326 2327 2328 2329 2330 2331 2332 2333 2334 2335 2336 2337 2338 2339 2340 2341 2342 2343 2344 2345 2346 2347 2348 2349 2350 2351 2352 2353 2354 2355 2356 2357 2358 2359 2360 2361 2362 2363 2364 2365 2366 2367 2368 2369 2370 2371 2372 2373 2374 2375 2376 2377 2378 2379 2380 2381 2382 2383 2384 2385 2386 2387 2388 2389 2390 2391 2392 2393 2394 2395 2396 2397 2398 2399 2400 2401 2402 2403 2404 2405 2406 2407 2408 2409 2410 2411 2412 2413 2414 2415 2416 2417 2418 2419 2420 2421 2422 2423 2424 2425 2426 2427 2428 2429 2430 2431 2432 2433 2434 2435 2436 2437 2438 2439 2440 2441 2442 2443 2444 2445 2446 2447 2448 2449 2450 2451 2452 2453 2454 2455 2456 2457 2458 2459 2460 2461 2462 2463 2464 2465 2466 2467 2468 2469 2470 2471 2472 2473 2474 2475 2476 2477 2478 2479 2480 2481 2482 2483 2484 2485 2486 2487 2488 2489 2490 2491 2492 2493 2494 2495 2496 2497 2498 2499 2500 2501 2502 2503 2504 2505 2506 2507 2508 2509 2510 2511 2512 2513 2514 2515 2516 2517 2518 2519 2520 2521 2522 2523 2524 2525 2526 2527 2528 2529 2530 2531 2532 2533 2534 2535 2536 2537 2538 2539 2540 2541 2542 2543 2544 2545 2546 2547 2548 2549 2550 2551 2552 2553 2554 2555 2556 2557 2558 2559 2560 2561 2562 2563 2564 2565 2566 2567 2568 2569 2570 2571 2572 2573 2574 2575 2576 2577 2578 2579 2580 2581 2582 2583 2584 2585 2586 2587 2588 2589 2590 2591 2592 2593 2594 2595 2596 2597 2598 2599 2600 2601 2602 2603 2604 2605 2606 2607 2608 2609 2610 2611 2612 2613 2614 2615 2616 2617 2618 2619 2620 2621 2622 2623 2624 2625 2626 2627 2628 2629 2630 2631 2632 2633 2634 2635 2636 2637 2638 2639 2640 2641 2642 2643 2644 2645 2646 2647 2648 2649 2650 2651 2652 2653 2654 2655 2656 2657 2658 2659 2660 2661 2662 2663 2664 2665 2



$\frac{1}{\sqrt{\pi}} \int_{-\infty}^{\infty} f(x) e^{-x^2} dx = \frac{1}{\sqrt{\pi}} \int_{-\infty}^{\infty} f(x) e^{-x^2} dx$

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.5.1 Verify each control rod scram accumulator pressure is \geq 940 psig.	7 days



3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Rod Pattern Control

LCO 3.1.6 OPERABLE control rods shall comply with the requirements of the banked position withdrawal sequence (BPWS).

APPLICABILITY: MODES 1 and 2 with THERMAL POWER \leq 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more OPERABLE control rods not in compliance with BPWS.	A.1 -----NOTE----- Rod Worth Minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation." -----	8 hours
	Move associated control rod(s) to correct position.	
	<u>OR</u> A.2 Declare associated control rod(s) inoperable.	8 hours

(continued)



ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Nine or more OPERABLE control rods not in compliance with BPWS.	B.1 -----NOTE----- RWM may be bypassed as allowed by LCO 3.3.2.1. -----	Immediately
	Suspend withdrawal of control rods.	
	<u>AND</u> B.2 Place the reactor mode switch in the shutdown position.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.6.1 Verify all OPERABLE control rods comply with BPWS.	24 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SLC subsystem inoperable.	A.1 Restore SLC subsystem to OPERABLE status.	7 days
B. Two SLC subsystems inoperable.	B.1 Restore one SLC subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.7.1 Verify available volume of sodium pentaborate solution is ≥ 4587 gallons.	24 hours

(continued)



SURVEILLANCE REQUIREMENTS (continued)

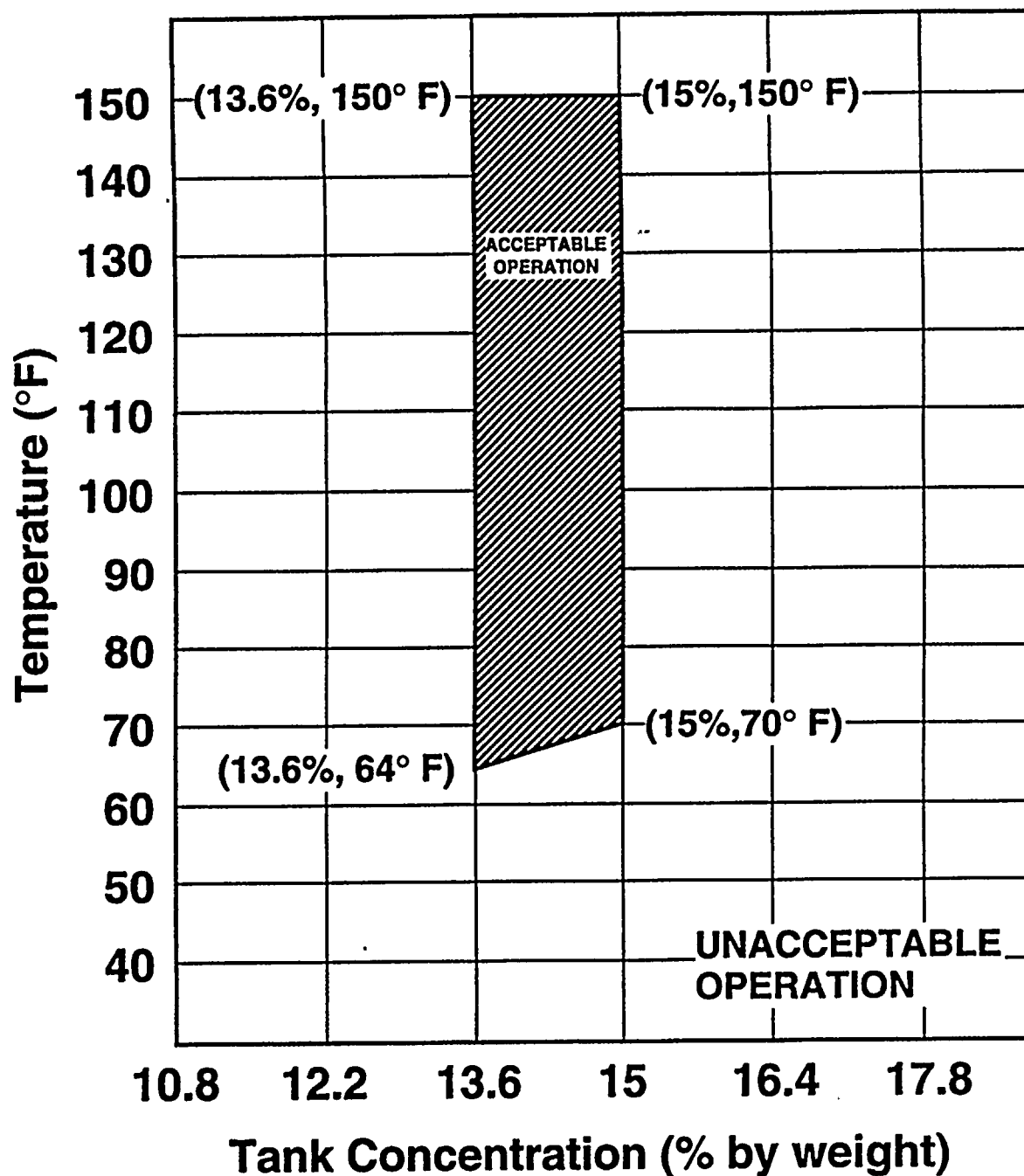
SURVEILLANCE	FREQUENCY
SR 3.1.7.2 Verify temperature of sodium pentaborate solution is within the limits of Figure 3.1.7-1.	24 hours
SR 3.1.7.3 Verify continuity of explosive charge.	31 days
SR 3.1.7.4 Verify the concentration of boron in solution is within the limits of Figure 3.1.7-1.	31 days <u>AND</u> Once within 24 hours after water or boron is added to solution <u>AND</u> Once within 24 hours after solution temperature is restored within the limits of Figure 3.1.7-1
SR 3.1.7.5 Verify each SLC subsystem manual and power operated valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.1.7.6 Verify each pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ 1220 psig.	In accordance with the Inservice Testing Program
SR 3.1.7.7 Verify flow through one SLC subsystem from pump into reactor pressure vessel.	24 months on a STAGGERED TEST BASIS
SR 3.1.7.8 Verify all heat traced piping between storage tank and pump suction valve is unblocked.	24 months <u>AND</u> Once within 24 hours after solution temperature is restored within the limits of Figure 3.1.7-1





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Figure 3.1.7-1 (Page 1 of 1)
Sodium Pentaborate Solution Temperature/Concentration Requirements



3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

LCO 3.1.8 Each SDV vent and drain valve shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

NOTES

1. Separate Condition entry is allowed for each SDV vent and drain line.
2. An isolated line may be unisolated under administrative control to allow draining and venting of the SDV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SDV vent or drain lines with one valve inoperable.	A.1 Isolate the associated line.	7 days
B. One or more SDV vent or drain lines with both valves inoperable.	B.1 Isolate the associated line.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours



SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.1	<p>-----NOTE----- Not required to be met on vent and drain valves closed during performance of SR 3.1.8.2. -----</p> <p>Verify each SDV vent and drain valve is open.</p>	31 days
SR 3.1.8.2	Cycle each SDV vent and drain valve to the fully closed and fully open position.	92 days
SR 3.1.8.3	<p>Verify each SDV vent and drain valve:</p> <ul style="list-style-type: none"> a. Closes in ≤ 30 seconds after receipt of an actual or simulated scram signal; and b. Opens when the actual or simulated scram signal is reset. 	24 months

3.2 POWER DISTRIBUTION LIMITS

3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

LCO 3.2.1 All APLHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER \geq 25% RTP.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.1 Verify all APLHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after \geq 25% RTP <u>AND</u> 24 hours thereafter



3.2 POWER DISTRIBUTION LIMITS

3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

LCO 3.2.2 All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR.

APPLICABILITY: THERMAL POWER \geq 25% RTP.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.2.1 Verify all MCPRs are greater than or equal to the limits specified in the COLR.	Once within 12 hours after \geq 25% RTP <u>AND</u> 24 hours thereafter



3.2 POWER DISTRIBUTION LIMITS

3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

LC0 3.2.3 All LHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER \geq 25% RTP:

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.3.1 Verify all LHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after \geq 25% RTP <u>AND</u> 24 hours thereafter



3.2 POWER DISTRIBUTION LIMITS

3.2.4 Average Power Range Monitor (APRM) Gain and Setpoint

- LCO 3.2.4
- a. MFLPD shall be less than or equal to Fraction of RTP (F RTP); or
 - b. Each required APRM Flow Biased Simulated Thermal Power—High Function Allowable Value shall be modified by greater than or equal to the ratio of F RTP and the MFLPD; or
 - c. Each required APRM gain shall be adjusted such that the APRM readings are $\geq 100\%$ times MFLPD.

APPLICABILITY: THERMAL POWER $\geq 25\%$ RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	6 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to $< 25\%$ RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.2.4.1 -----NOTE----- Not required to be met if SR 3.2.4.2 is satisfied for LCO 3.2.4.b or LCO 3.2.4.c requirements. -----</p> <p>Verify MFLPD is within limits.</p>	<p>Once within 12 hours after $\geq 25\%$ RTP</p> <p><u>AND</u></p> <p>24 hours thereafter</p>
<p>SR 3.2.4.2 -----NOTE----- Not required to be met if SR 3.2.4.1 is satisfied for LCO 3.2.4.a requirements. -----</p> <p>Verify each required:</p> <p>a. APRM Flow Biased Simulated Thermal Power—High Function Allowable Value is modified by greater than or equal to the ratio of FRTP and the MFLPD; or</p> <p>b. APRM gain is adjusted such that the APRM reading is $\geq 100\%$ times MFLPD.</p>	<p>12 hours</p>

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	12 hours
	<u>OR</u> A.2 Place associated trip system in trip.	12 hours
B. One or more Functions with one or more required channels inoperable in both trip systems.	B.1 Place channel in one trip system in trip.	6 hours
	<u>OR</u> B.2 Place one trip system in trip.	6 hours
C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour

(continued)



ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1 Reduce THERMAL POWER to < 30% RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1 Be in MODE 2.	6 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1 Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTES-----

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.
2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.1.2	<p>-----NOTE----- Not required to be performed until 12 hours after THERMAL POWER \geq 25% RTP. -----</p> <p>Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power \leq 2% RTP plus any gain adjustment required by LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoint," while operating at \geq 25% RTP.</p>	7 days
SR 3.3.1.1.3	<p>-----NOTE----- Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -----</p> <p>Perform CHANNEL FUNCTIONAL TEST.</p>	7 days
SR 3.3.1.1.4	Perform CHANNEL FUNCTIONAL TEST.	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.5	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to withdrawing SRMs from the fully inserted position
SR 3.3.1.1.6	<p>-----NOTE----- Only required to be met during entry into MODE 2 from MODE 1. -----</p> <p>Verify the IRM and APRM channels overlap.</p>	7 days
SR 3.3.1.1.7	Calibrate the local power range monitors.	1130 MWD/T average core exposure
SR 3.3.1.1.8	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.9	<p>-----NOTES----- 1. Neutron detectors are excluded. 2. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -----</p> <p>Perform CHANNEL CALIBRATION.</p>	184 days

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.10	<p>-----NOTES-----</p> <ol style="list-style-type: none"> 1. Neutron detectors are excluded. 2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. <p>-----</p> <p>Perform CHANNEL CALIBRATION.</p>	18 months
SR 3.3.1.1.11	Verify the APRM Flow Biased Simulated Thermal Power—High Function time constant is ≤ 7 seconds.	18 months
SR 3.3.1.1.12	Verify Turbine Throttle Valve—Closure, and Turbine Governor Valve Fast Closure Trip Oil Pressure—Low Functions are not bypassed when THERMAL POWER is $\geq 30\%$ RTP.	18 months
SR 3.3.1.1.13	Perform CHANNEL FUNCTIONAL TEST.	24 months
SR 3.3.1.1.14	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

(continued)

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.3.1.1.15 -----NOTES-----</p> <ol style="list-style-type: none"> 1. Neutron detectors are excluded. 2. For Function 5, "n" equals 4 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. <p>-----</p> <p>Verify the RPS RESPONSE TIME is within limits.</p>	<p>24 months on a STAGGERED TEST BASIS</p>

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux - High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 122/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 122/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.14	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux - High, Setdown	2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 20% RTP
b. Flow Biased Simulated Thermal Power - High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.14	≤ 0.58W + 62% RTP and ≤ 114.9% RTP
c. Fixed Neutron Flux - High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 120% RTP
d. Inop	1,2	2	G	SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.14	NA
3. Reactor Vessel Steam Dome Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 1079 psig

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.



Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. Reactor Vessel Water Level - Low, Level 3	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.15	≥ 9.5 inches
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 12.5% closed
6. Primary Containment Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1.88 psig
7. Scram Discharge Volume Water Level - High					
a. Transmitter/Trip Unit	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 529 ft 9 inches elevation
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 529 ft 9 inches elevation
b. Float Switch	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 529 ft 9 inches elevation
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 529 ft 9 inches elevation
8. Turbine Throttle Valve - Closure	≥ 30% RTP	4	E	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.12 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 7% closed
9. Turbine Governor Valve Fast Closure, Trip Oil Pressure - Low	≥ 30% RTP	2	E	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.12 SR 3.3.1.1.14 SR 3.3.1.1.15	≥ 1000 psig
10. Reactor Mode Switch - Shutdown Position	1,2	2	G	SR 3.3.1.1.13 SR 3.3.1.1.14	NA
	5(a)	2	H	SR 3.3.1.1.13 SR 3.3.1.1.14	NA

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

Table 3.3.1.1-1 (page 3 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
11. Manual Scram	1,2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
	5(a)	2	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

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3.3 INSTRUMENTATION

3.3.1.2 Source Range Monitor (SRM) Instrumentation

LCO 3.3.1.2 The SRM instrumentation in Table 3.3.1.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.2-1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required SRMs inoperable in MODE 2 with intermediate range monitors (IRMs) on Range 2 or below.	A.1 Restore required SRMs to OPERABLE status.	4 hours
B. Three required SRMs inoperable in MODE 2 with IRMs on Range 2 or below.	B.1 Suspend control rod withdrawal.	Immediately
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3.	12 hours
D. One or more required SRMs inoperable in MODE 3 or 4.	D.1 Fully insert all insertable control rods. <u>AND</u>	1 hour (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. (continued)	D.2 Place reactor mode switch in the shutdown position.	1 hour
E. One or more required SRMs inoperable in MODE 5.	E.1 Suspend CORE ALTERATIONS except for control rod insertion.	Immediately
	<u>AND</u> E.2 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.1.2-1 to determine which SRs apply for each applicable MODE or other specified condition.

SURVEILLANCE	FREQUENCY
SR 3.3.1.2.1 Perform CHANNEL CHECK.	12 hours

(continued)

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.3.1.2.2 -----NOTES-----</p> <ol style="list-style-type: none"> 1. Only required to be met during CORE ALTERATIONS. 2. One SRM may be used to satisfy more than one of the following. <p>-----</p> <p>Verify an OPERABLE SRM detector is located in:</p> <ol style="list-style-type: none"> a. The fueled region; b. The core quadrant where CORE ALTERATIONS are being performed when the associated SRM is included in the fueled region; and c. A core quadrant adjacent to where CORE ALTERATIONS are being performed, when the associated SRM is included in the fueled region. 	<p>12 hours</p>
<p>SR 3.3.1.2.3 Perform CHANNEL CHECK.</p>	<p>24 hours</p>
<p>SR 3.3.1.2.4 -----NOTE-----</p> <p>Not required to be met with less than or equal to four fuel assemblies adjacent to the SRM and no other fuel assemblies in the associated core quadrant.</p> <p>-----</p> <p>Verify count rate is:</p> <ol style="list-style-type: none"> a. ≥ 3.0 cps with a signal to noise ratio $\geq 2:1$, or b. ≥ 0.7 cps with a signal to noise ratio $\geq 20:1$. 	<p>12 hours during CORE ALTERATIONS</p> <p><u>AND</u></p> <p>24 hours</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.3.1.2.5 -----NOTE----- The determination of signal to noise ratio is not required to be met with less than or equal to four fuel assemblies adjacent to the SRM and no other fuel assemblies in the associated core quadrant. ----- Perform CHANNEL FUNCTIONAL TEST and determination of signal to noise ratio.</p>	<p>7 days</p>
<p>SR 3.3.1.2.6 -----NOTE----- Not required to be performed until 12 hours after IRMs on Range 2 or below. ----- Perform CHANNEL FUNCTIONAL TEST and determination of signal to noise ratio.</p>	<p>31 days</p>
<p>SR 3.3.1.2.7 -----NOTES----- 1. Neutron detectors are excluded. 2. Not required to be performed until 12 hours after IRMs on Range 2 or below. ----- Perform CHANNEL CALIBRATION.</p>	<p>18 months</p>

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Table 3.3.1.2-1 (page 1 of 1)
Source Range Monitor Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS
1. Source Range Monitor	2(a)	3	SR 3.3.1.2.1
			SR 3.3.1.2.4
			SR 3.3.1.2.6
			SR 3.3.1.2.7
	3,4	2	SR 3.3.1.2.3
			SR 3.3.1.2.4
			SR 3.3.1.2.6
			SR 3.3.1.2.7
	5	2(b),(c)	SR 3.3.1.2.1
			SR 3.3.1.2.2
			SR 3.3.1.2.4
			SR 3.3.1.2.5
			SR 3.3.1.2.7

(a) With IRMs on Range 2 or below.

(b) Only one SRM channel is required to be OPERABLE during spiral offload or reload when the fueled region includes only that SRM detector.

(c) Special movable detectors may be used in place of SRMs if connected to normal SRM circuits.

3.3 INSTRUMENTATION

3.3.2.1 Control Rod Block Instrumentation

LCO 3.3.2.1 The control rod block instrumentation for each Function in Table 3.3.2.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.2.1-1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One rod block monitor (RBM) channel inoperable.	A.1 Restore RBM channel to OPERABLE status.	24 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Two RBM channels inoperable.	B.1 Place one RBM channel in trip.	1 hour
C. Rod worth minimizer (RWM) inoperable during reactor startup.	C.1 Suspend control rod movement except by scram. <u>OR</u>	Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2.1.1 Verify ≥ 12 rods withdrawn.	Immediately
	<u>OR</u>	
	C.2.1.2 Verify by administrative methods that startup with RWM inoperable has not been performed in the last calendar year.	Immediately
	<u>AND</u>	
	C.2.2 Verify movement of control rods is in compliance with banked position withdrawal sequence (BPWS) by a second licensed operator or other qualified member of the technical staff.	During control rod movement
D. RWM inoperable during reactor shutdown.	D.1 Verify movement of control rods is in compliance with BPWs by a second licensed operator or other qualified member of the technical staff.	During control rod movement

(continued)



ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One or more Reactor Mode Switch—Shutdown Position channels inoperable.	E.1 Suspend control rod withdrawal.	Immediately
	AND E.2 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

- NOTES-----
1. Refer to Table 3.3.2.1-1 to determine which SRs apply for each Control Rod Block Function.
 2. When an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability.
-

SURVEILLANCE	FREQUENCY
SR 3.3.2.1.1 Perform CHANNEL FUNCTIONAL TEST.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.3.2.1.2 -----NOTE----- Not required to be performed until 1 hour after any control rod is withdrawn at $\leq 10\%$ RTP in MODE 2. -----</p> <p>Perform CHANNEL FUNCTIONAL TEST.</p>	92 days
<p>SR 3.3.2.1.3 -----NOTE----- Not required to be performed until 1 hour after THERMAL POWER is $\leq 10\%$ RTP in MODE 1. -----</p> <p>Perform CHANNEL FUNCTIONAL TEST.</p>	92 days
<p>SR 3.3.2.1.4 -----NOTE----- Neutron detectors are excluded. -----</p> <p>Verify the RBM is not bypassed:</p> <p>a. When THERMAL POWER is $\geq 30\%$ RTP; and</p> <p>b. When a peripheral control rod is not selected.</p>	92 days
<p>SR 3.3.2.1.5 -----NOTE----- Neutron detectors are excluded. -----</p> <p>Perform CHANNEL CALIBRATION.</p>	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.6	Verify the RWM is not bypassed when THERMAL POWER is \leq 10% RTP.	18 months
SR 3.3.2.1.7	<p>-----NOTE----- Not required to be performed until 1 hour after reactor mode switch is in the shutdown position. -----</p> <p>Perform CHANNEL FUNCTIONAL TEST.</p>	24 months
SR 3.3.2.1.8	Verify control rod sequences input to the RWM are in conformance with BPWS.	Prior to declaring RWM OPERABLE following loading of sequence into RWM

Control Rod Block Instrumentation
3.3.2.1

Table 3.3.2.1-1 (page 1 of 1)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
a. Upscale	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.5	$\leq 0.58W + 51\% \text{ RTP}$
b. Inop	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4	NA
c. Downscale	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.5	$\geq 3\% \text{ RTP}$
2. Rod Worth Minimizer	1 ^(b) , 2 ^(b)	1	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.6 SR 3.3.2.1.8	NA
3. Reactor Mode Switch - Shutdown Position	(c)	2	SR 3.3.2.1.7	NA

(a) THERMAL POWER $\geq 30\% \text{ RTP}$ and no peripheral control rod selected.

(b) With THERMAL POWER $\leq 10\% \text{ RTP}$.

(c) Reactor mode switch in the shutdown position.

3.3 INSTRUMENTATION

3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation

LCO 3.3.2.2 Three channels of feedwater and main turbine high water level trip instrumentation shall be OPERABLE.

APPLICABILITY: THERMAL POWER \geq 25% RTP.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One feedwater and main turbine high water level trip channel inoperable.	A.1 Place channel in trip.	7 days
B. Two or more feedwater and main turbine high water level trip channels inoperable.	B.1 Restore feedwater and main turbine high water level trip capability.	2 hours
C. Required Action and associated Completion Time not met.	C.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided feedwater and main turbine high water level trip capability is maintained.

SURVEILLANCE	FREQUENCY
SR 3.3.2.2.1 Perform CHANNEL CHECK.	24 hours
SR 3.3.2.2.2 Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.2.3 Perform CHANNEL CALIBRATION. The Allowable Value shall be ≤ 56.0 inches.	24 months
SR 3.3.2.2.4 Perform LOGIC SYSTEM FUNCTIONAL TEST, including valve actuation.	24 months

3.3 INSTRUMENTATION

3.3.3.1 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.3.1 The PAM instrumentation for each Function in Table 3.3.3.1-1 shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

- NOTES-----
1. LCO 3.0.4 is not applicable.
 2. Separate Condition entry is allowed for each Function.
-

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1 Restore required channel to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action in accordance with Specification 5.6.6.	Immediately
C. One or more Functions with two or more required channels inoperable.	C.1 Restore all but one required channel to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met.	D.1 Enter the Condition referenced in Table 3.3.3.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.3.1-1.	E.1 Be in MODE 3.	12 hours
F. As required by Required Action D.1 and referenced in Table 3.3.3.1-1.	F.1 Initiate action in accordance with Specification 5.6.6.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTES-----

1. These SRs apply to each Function in Table 3.3.3.1-1.
 2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the other required channel(s) in the associated Function is OPERABLE.
-

SURVEILLANCE	FREQUENCY
SR 3.3.3.1.1 Perform CHANNEL CHECK.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.3.1.2	Perform CHANNEL CALIBRATION for Function 8.	92 days
SR 3.3.3.1.3	Perform CHANNEL CALIBRATION for Functions 1, 2, 4, 5, 7, 9, and 10.	18 months
SR 3.3.3.1.4	Perform CHANNEL CALIBRATION for Functions 3 and 6.	24 months

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Table 3.3.3.1-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1
1. Reactor Vessel Pressure	2	E
2. Reactor Vessel Water Level		
a. -150 inches to +60 inches	2	E
b. -310 inches to -110 inches	2	E
3. Suppression Pool Water Level		
a. -25 inches to +25 inches	2	E
b. 2 ft to 52 ft	2	E
4. Suppression Chamber Pressure	2	E
5. Drywell Pressure		
a. -5 psig to +3 psig	2	E
b. 0 psig to 25 psig	2	E
c. 0 psig to 180 psig	2	E
6. Primary Containment Area Radiation	2	F
7. PCIV Position	2 per penetration flow path (a)(b)	E
8. Drywell H ₂ Analyzer	2	E
9. Drywell O ₂ Analyzer	2	E
10. ECCS Pump Room Flood Level	5	E

(a) Not required for isolation valves whose associated penetration flow path is isolated by at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.



3.3 INSTRUMENTATION

3.3.3.2 Remote Shutdown System

LC0 3.3.3.2 The Remote Shutdown System Functions shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

- NOTES-----
1. LC0 3.0.4 is not applicable.
 2. Separate Condition entry is allowed for each Function.
-

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required Functions inoperable.	A.1 Restore required Function to OPERABLE status.	30 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours



SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE		FREQUENCY
SR 3.3.3.2.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days
SR 3.3.3.2.2	Perform CHANNEL CALIBRATION for each required instrumentation channel, except the suppression pool water level instrumentation channel.	18 months
SR 3.3.3.2.3	Perform CHANNEL CALIBRATION for the suppression pool water level instrumentation channel.	24 months
SR 3.3.3.2.4	Verify each required control circuit and transfer switch is capable of performing the intended functions.	24 months



3.3 INSTRUMENTATION

3.3.4.1 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

LCO 3.3.4.1 a. Two channels per trip system for each EOC-RPT instrumentation Function listed below shall be OPERABLE:

1. Turbine Throttle Valve (TTV)—Closure; and
2. Turbine Governor Valve (TGV) Fast Closure, Trip Oil Pressure—Low.

OR

- b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for inoperable EOC-RPT as specified in the COLR are made applicable.

APPLICABILITY: THERMAL POWER \geq 30% RTP.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Restore channel to OPERABLE status.	72 hours
	<u>OR</u>	(continued)

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(b) $\frac{1}{\sqrt{2}}$

Group	Condition 1	Condition 2	Condition 3	Condition 4
Control	65	75	80	85
Low	65	80	75	60
High	65	85	80	65
Very High	65	90	85	70

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ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	<p>A.2 -----NOTE----- Not applicable if inoperable channel is the result of an inoperable breaker. -----</p> <p>Place channel in trip.</p>	72 hours
<p>B. One or more Functions with EOC-RPT trip capability not maintained.</p> <p><u>AND</u></p> <p>MCPR limit for inoperable EOC-RPT not made applicable.</p>	<p>B.1 Restore EOC-RPT trip capability.</p> <p><u>OR</u></p> <p>B.2 Apply the MCPR limit for inoperable EOC-RPT as specified in the COLR.</p>	<p>2 hours</p> <p>2 hours</p>
C. Required Action and associated Completion Time not met.	<p>C.1 Remove the associated recirculation pump from service.</p> <p><u>OR</u></p> <p>C.2 Reduce THERMAL POWER to < 30% RTP.</p>	<p>4 hours</p> <p>4 hours</p>

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1. The first part of the report is a summary of the work done during the year.

2. The second part is a detailed account of the work done during the year.

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4. The third part is a summary of the work done during the year.

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SURVEILLANCE REQUIREMENTS

-----NOTE-----
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains EOC-RPT trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.4.1.1	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.4.1.2	Perform CHANNEL CALIBRATION. The Allowable Values shall be: a. TTV—Closure: $\leq 7\%$ closed; and b. TGV Fast Closure, Trip Oil Pressure—Low: ≥ 1000 psig.	18 months
SR 3.3.4.1.3	Verify TTV—Closure and TGV Fast Closure, Trip Oil Pressure—Low Functions are not bypassed when THERMAL POWER is $\geq 30\%$ RTP.	18 months
SR 3.3.4.1.4	Perform LOGIC SYSTEM FUNCTIONAL TEST, including breaker actuation.	24 months
SR 3.3.4.1.5	-----NOTE----- Breaker arc suppression time may be assumed from the most recent performance of SR 3.3.4.1.6. ----- Verify the EOC-RPT SYSTEM RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

(continued)



SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.4.1.6	Determine RPT breaker arc suppression time.	60 months



3.3 INSTRUMENTATION

3.3.4.2 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation

LCO 3.3.4.2 Two channels per trip system for each ATWS-RPT instrumentation Function listed below shall be OPERABLE:

- a. Reactor Vessel Water Level—Low Low, Level 2; and
- b. Reactor Vessel Steam Dome Pressure—High.

APPLICABILITY: MODE 1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Restore channel to OPERABLE status.	7 days
	<p><u>OR</u></p> <p>A.2 -----NOTE----- Not applicable if inoperable channel is the result of an inoperable breaker. -----</p> <p>Place channel in trip.</p>	7 days

(continued)



ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One Function with ATWS-RPT trip capability not maintained.	B.1 Restore ATWS-RPT trip capability.	72 hours
C. Both Functions with ATWS-RPT trip capability not maintained.	C.1 Restore ATWS-RPT trip capability for one Function.	1 hour
D. Required Action and associated Completion Time not met.	D.1 Remove the associated recirculation pump from service.	6 hours
	<u>OR</u> D.2 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains ATWS-RPT trip capability.

SURVEILLANCE	FREQUENCY
SR 3.3.4.2.1 Perform CHANNEL CHECK for Reactor Vessel Water Level—Low Low, Level 2 Function.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.4.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.4.2.3	Perform CHANNEL CALIBRATION. The Allowable Values shall be: <ul style="list-style-type: none"> a. Reactor Vessel Water Level—Low Low, Level 2: ≥ -58 inches; and b. Reactor Vessel Steam Dome Pressure—High: ≤ 1143 psig. 	18 months
SR 3.3.4.2.4	Perform LOGIC SYSTEM FUNCTIONAL TEST, including breaker actuation.	24 months

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3.3 INSTRUMENTATION

3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation

LCO 3.3.5.1 The ECCS instrumentation for each Function in Table 3.3.5.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.5.1-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.5.1-1 for the channel.	Immediately
B. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	<p>B.1 -----NOTES-----</p> <p>1. Only applicable in MODES 1, 2, and 3.</p> <p>2. Only applicable for Functions 1.a, 1.b, 2.a, and 2.b.</p> <p>-----</p> <p>Declare supported feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.</p> <p><u>AND</u></p>	<p>1 hour from discovery of loss of initiation capability for feature(s) in both divisions</p> <p>(continued)</p>



ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	<p>B.2</p> <p>-----NOTES-----</p> <p>1. Only applicable in MODES 1, 2, and 3.</p> <p>2. Only applicable for Functions 3.a and 3.b.</p> <p>-----</p> <p>Declare High Pressure Core Spray (HPCS) System inoperable.</p>	1 hour from discovery of loss of HPCS initiation capability
	<p><u>AND</u></p> <p>B.3</p> <p>Place channel in trip.</p>	24 hours
C. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	<p>C.1</p> <p>-----NOTES-----</p> <p>1. Only applicable in MODES 1, 2, and 3.</p> <p>2. Only applicable for Functions 1.c, 1.d, 1.e, 1.f, 2.c, 2.d, 2.e, and 2.f.</p> <p>-----</p> <p>Declare supported feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.</p>	1 hour from discovery of loss of initiation capability for feature(s) in both divisions
	<p><u>AND</u></p>	(continued)

Figure 1 shows a 2D hexagonal lattice structure. A central atom is labeled 'A'. To its right is an atom labeled 'B'. Above the central atom is an atom labeled 'C'. Below the central atom is an atom labeled 'D'. To the left of the central atom is an atom labeled 'E'. To the right of the central atom is an atom labeled 'F'. The atoms are arranged in a regular hexagonal pattern.

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ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2 Restore channel to OPERABLE status.	24 hours
D. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	D.1 -----NOTE----- Only applicable if HPCS pump suction is not aligned to the suppression pool. ----- Declare HPCS System inoperable.	1 hour from discovery of loss of HPCS initiation capability
	<u>AND</u>	
	D.2.1 Place channel in trip.	24 hours
	<u>OR</u>	
	D.2.2 Align the HPCS pump suction to the suppression pool.	24 hours

(continued)

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	<p>E.1</p> <p>-----NOTES-----</p> <p>1. Only applicable in MODES 1, 2, and 3.</p> <p>2. Only applicable for Functions 1.g, 1.h, and 2.g.</p> <p>-----</p> <p>Declare supported feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.</p>	<p>1 hour from discovery of loss of initiation capability for feature(s) in both divisions</p> <p>7 days</p>
	<p><u>AND</u></p> <p>E.2 Restore channel to OPERABLE status.</p>	
F. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	<p>F.1 Declare Automatic Depressurization System (ADS) valves inoperable.</p>	<p>1 hour from discovery of loss of ADS initiation capability in both trip systems</p> <p>(continued)</p>
	<p><u>AND</u></p>	



ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. (continued)	F.2 Place channel in trip.	96 hours from discovery of inoperable channel concurrent with HPCS or reactor core isolation cooling (RCIC) inoperable <u>AND</u> 8 days
G. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	G.1 -----NOTE----- Only applicable for Functions 4.b, 4.d, 4.e, 5.b, and 5.d. ----- Declare ADS valves inoperable. <u>AND</u>	1 hour from discovery of loss of ADS initiation capability in both trip systems (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. (continued)	G.2 Restore channel to OPERABLE status.	96 hours from discovery of inoperable channel concurrent with HPCS or RCIC inoperable <u>AND</u> 8 days
H. Required Action and associated Completion Time of Condition B, C, D, E, F, or G not met.	H.1 Declare associated supported feature(s) inoperable.	Immediately



SURVEILLANCE REQUIREMENTS

NOTES

1. Refer to Table 3.3.5.1-1 to determine which SRs apply for each ECCS Function.
2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed as follows: (a) for up to 6 hours for Functions 3.c, 3.f, and 3.g; and (b) for up to 6 hours for Functions other than 3.c, 3.f, and 3.g provided the associated Function or the redundant Function maintains ECCS initiation capability.

SURVEILLANCE	FREQUENCY
SR 3.3.5.1.1 Perform CHANNEL CHECK.	12 hours
SR 3.3.5.1.2 Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.5.1.3 Perform CHANNEL CALIBRATION.	92 days
SR 3.3.5.1.4 Perform CHANNEL CALIBRATION.	18 months
SR 3.3.5.1.5 Perform CHANNEL CALIBRATION.	24 months
SR 3.3.5.1.6 Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.5.1.7 Verify the ECCS RESPONSE TIME for each required ECCS injection/spray subsystem is within limits.	24 months on a STAGGERED TEST BASIS

Table 3.3.5.1-1 (page 1 of 4)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Low Pressure Coolant Injection-A (LPCI) and Low Pressure Core Spray (LPCS) Subsystems					
a. Reactor Vessel Water Level - Low Low Low, Level 1	1,2,3, 4(a),5(a)	2(b)	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6 SR 3.3.5.1.7	≥ -148 inches
b. Drywell Pressure - High	1,2,3	2(b)	B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6 SR 3.3.5.1.7	≤ 1.88 psig
c. LPCS Pump Start - LOCA Time Delay Relay	1,2,3, 4(a),5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 8.53 seconds and ≤ 10.64 seconds
d. LPCI Pump A Start - LOCA Time Delay Relay	1,2,3, 4(a),5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 17.24 seconds and ≤ 21.53 seconds
e. LPCI Pump A Start - LOCA/LOOP Time Delay Relay	1,2,3, 4(a),5(a)	1	C	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 3.04 seconds and ≤ 6.00 seconds
f. Reactor Vessel Pressure - Low (Injection Permissive)	1,2,3 4(a),5(a)	1 per valve 1 per valve	C B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 448 psig and ≤ 492 psig ≥ 448 psig and ≤ 492 psig
g. LPCS Pump Discharge Flow - Low (Minimum Flow)	1,2,3, 4(a),5(a)	1	E	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 668 gpm and ≤ 1067 gpm
h. LPCI Pump A Discharge Flow - Low (Minimum Flow)	1,2,3, 4(a),5(a)	1	E	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 605 gpm and ≤ 984 gpm
i. Manual Initiation	1,2,3, 4(a),5(a)	2	C	SR 3.3.5.1.6	NA

(continued)

(a) When associated subsystem(s) are required to be OPERABLE.

(b) Also required to initiate the associated diesel generator (DG).

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Table 3.3.5.1-1 (page 2 of 4)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI B and LPCI C Subsystems					
a. Reactor Vessel Water Level - Low Low Low, Level 1	1,2,3, 4(a),5(a)	2(b)	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6 SR 3.3.5.1.7	≥ -148 inches
b. Drywell Pressure - High	1,2,3	2(b)	B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6 SR 3.3.5.1.7	≤ 1.88 psig
c. LPCI Pump B Start - LOCA Time Delay Relay	1,2,3, 4(a),5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 17.24 seconds and ≤ 21.53 seconds
d. LPCI Pump C Start - LOCA Time Delay Relay	1,2,3, 4(a),5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 8.53 seconds and ≤ 10.64 seconds
e. LPCI Pump B Start - LOCA/LOOP Time Delay Relay	1,2,3, 4(a),5(a)	1	C	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 3.04 seconds and ≤ 6.00 seconds
f. Reactor Vessel Pressure - Low (Injection Permissive)	1,2,3 4(a),5(a)	1 per valve 1 per valve	C B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 448 psig and ≤ 492 psig ≥ 448 psig and ≤ 492 psig
g. LPCI Pumps B & C Discharge Flow - Low (Minimum Flow)	1,2,3, 4(a),5(a)	1 per pump	E	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 605 gpm and ≤ 984 gpm
h. Manual Initiation	1,2,3, 4(a),5(a)	2	C	SR 3.3.5.1.6	NA
3. High Pressure Core Spray (HPCS) System					
a. Reactor Vessel Water Level - Low Low, Level 2	1,2,3, 4(a),5(a)	4(b)	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6 SR 3.3.5.1.7	≥ -58 inches

(continued)

(a) When associated subsystem(s) are required to be OPERABLE.

(b) Also required to initiate the associated DG.

Table 3.3.5.1-1 (page 3 of 4)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. HPCS System (continued)					
b. Drywell Pressure - High	1,2,3	4(b)	B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6 SR 3.3.5.1.7	≤ 1.88 psig
c. Reactor Vessel Water Level - High, Level 8	1,2,3, 4(a),5(a)	2	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≤ 56.0 inches
d. Condensate Storage Tank Level - Low	1,2,3, 4(c),5(c)	2	D	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 448 ft 1 inch elevation
e. Suppression Pool Water Level - High	1,2,3	2	D	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≤ 466 ft 11 inches elevation
f. HPCS System Flow Rate - Low (Minimum Flow)	1,2,3, 4(a),5(a)	1	E	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 1200 gpm and ≤ 1512 gpm
g. Manual Initiation	1,2,3, 4(a),5(a)	2	C	SR 3.3.5.1.6	NA
4. Automatic Depressurization System (ADS) Trip System A					
a. Reactor Vessel Water Level - Low Low Low, Level 1	1,2(d),3(d)	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ -148 inches
b. ADS Initiation Timer	1,2(d),3(d)	1	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≤ 115.0 seconds
c. Reactor Vessel Water Level - Low, Level 3 (Permissive)	1,2(d),3(d)	1	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 9.5 inches
d. LPCS Pump Discharge Pressure - High	1,2(d),3(d)	2	G	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 119 psig and ≤ 171 psig

(continued)

(a) When associated subsystem(s) are required to be OPERABLE.

(b) Also required to initiate the associated DG.

(c) When HPCS is OPERABLE for compliance with LCO 3.5.2, "ECCS - Shutdown," and aligned to the condensate storage tank while tank water level is not within the limit of SR 3.5.2.2.

(d) With reactor steam dome pressure > 150 psig.

Table 3.3.5.1-1 (page 4 of 4)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. ADS Trip System A (continued)					
e. LPCI Pump A Discharge Pressure - High	1,2(d),3(d)	2	G	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 116 psig and ≤ 134 psig
f. Accumulator Backup Compressed Gas System Pressure - Low	1,2(d),3(d)	3	F	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 151.4 psig
g. Manual Initiation	1,2(d),3(d)	4	G	SR 3.3.5.1.6	NA
5. ADS Trip System B					
a. Reactor Vessel Water Level - Low Low Low, Level 1	1,2(d),3(d)	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ -148 inches
b. ADS Initiation Timer	1,2(d),3(d)	1	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≤ 115.0 seconds
c. Reactor Vessel Water Level - Low, Level 3 (Permissive)	1,2(d),3(d)	1	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 9.5 inches
d. LPCI Pumps B & C Discharge Pressure - High	1,2(d),3(d)	2 per pump	G	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 116 psig and ≤ 134 psig
e. Accumulator Backup Compressed Gas System Pressure - Low	1,2(d),3(d)	3	F	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 151.4 psig
f. Manual Initiation	1,2(d),3(d)	4	G	SR 3.3.5.1.6	NA

(d) With reactor steam dome pressure > 150 psig.

3.3 INSTRUMENTATION

3.3.5.2 Reactor Core Isolation Cooling (RCIC) System Instrumentation

LCO 3.3.5.2 The RCIC System instrumentation for each Function in Table 3.3.5.2-1 shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 with reactor steam dome pressure > 150 psig.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.5.2-1 for the channel.	Immediately
B. As required by Required Action A.1 and referenced in Table 3.3.5.2-1.	B.1 Declare RCIC System inoperable.	1 hour from discovery of loss of RCIC initiation capability
	<u>AND</u> B.2 Place channel in trip.	24 hours
C. As required by Required Action A.1 and referenced in Table 3.3.5.2-1.	C.1 Restore channel to OPERABLE status.	24 hours

(continued)

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100-100000-100000

100-100000-100000

100-100000-100000

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. As required by Required Action A.1 and referenced in Table 3.3.5.2-1.	D.1 -----NOTE----- Only applicable if RCIC pump suction is not aligned to the suppression pool. ----- Declare RCIC System inoperable.	1 hour from discovery of loss of RCIC initiation capability
	<u>AND</u>	
	D.2.1 Place channel in trip.	24 hours
	<u>OR</u>	
	D.2.2 Align RCIC pump suction to the suppression pool.	24 hours
E. Required Action and associated Completion Time of Condition B, C, or D not met.	E.1 Declare RCIC System inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTES-----

1. Refer to Table 3.3.5.2-1 to determine which SRs apply for each RCIC Function.
 2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed as follows: (a) for up to 6 hours for Functions 2 and 4; and (b) for up to 6 hours for Functions 1 and 3 provided the associated Function maintains RCIC initiation capability.
-

SURVEILLANCE	FREQUENCY
SR 3.3.5.2.1 Perform CHANNEL CHECK.	12 hours
SR 3.3.5.2.2 Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.5.2.3 Perform CHANNEL CALIBRATION.	18 months
SR 3.3.5.2.4 Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

Table 3.3.5.2-1 (page 1 of 1)
Reactor Core Isolation Cooling System Instrumentation

FUNCTION	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low Low, Level 2	4	B	SR 3.3.5.2.1 SR 3.3.5.2.2 SR 3.3.5.2.3 SR 3.3.5.2.4	≥ -58 inches
2. Reactor Vessel Water Level - High, Level 8	2	C	SR 3.3.5.2.1 SR 3.3.5.2.2 SR 3.3.5.2.3 SR 3.3.5.2.4	≤ 56 inches
3. Condensate Storage Tank Level - Low	2	D	SR 3.3.5.2.1 SR 3.3.5.2.2 SR 3.3.5.2.3 SR 3.3.5.2.4	≥ 446 ft 0 inches elevation
4. Manual Initiation	2	C	SR 3.3.5.2.4	NA

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3.3 INSTRUMENTATION

3.3.6.1 Primary Containment Isolation Instrumentation

LCO 3.3.6.1 The primary containment isolation instrumentation for each Function in Table 3.3.6.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6.1-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	12 hours for Functions 2.a, 2.c, and 5.d <u>AND</u> 24 hours for Functions other than Functions 2.a, 2.c, and 5.d
B. One or more automatic Functions with isolation capability not maintained.	B.1 Restore isolation capability.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Enter the Condition referenced in Table 3.3.6.1-1 for the channel.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	D.1 Isolate associated main steam line (MSL).	12 hours
	<u>OR</u>	
	D.2.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	D.2.2 Be in MODE 4.	36 hours
E. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	E.1 Be in MODE 2.	6 hours
F. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	F.1 Isolate the affected penetration flow path(s).	1 hour
G. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	G.1 Isolate the affected penetration flow path(s).	24 hours

(continued)

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>H. Required Action and associated Completion Time of Condition F or G not met.</p> <p><u>OR</u></p> <p>As required by Required Action C.1 and referenced in Table 3.3.6.1-1.</p>	<p>H.1 Be in MODE 3.</p> <p><u>AND</u></p>	12 hours
	<p>H.2 Be in MODE 4.</p>	36 hours
<p>I. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.</p>	<p>I.1 Declare associated standby liquid control (SLC) subsystem inoperable.</p> <p><u>OR</u></p>	1 hour
	<p>I.2 Isolate the Reactor Water Cleanup (RWCU) System.</p>	1 hour
<p>J. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.</p>	<p>J.1 Initiate action to restore channel to OPERABLE status.</p> <p><u>OR</u></p>	Immediately
	<p>J.2 Initiate action to isolate the Residual Heat Removal (RHR) Shutdown Cooling (SDC) System.</p>	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTES-----

1. Refer to Table 3.3.6.1-1 to determine which SRs apply for each Primary Containment Isolation Function.
 2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains isolation capability.
-

SURVEILLANCE	FREQUENCY
SR 3.3.6.1.1 Perform CHANNEL CHECK.	12 hours
SR 3.3.6.1.2 Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.6.1.3 Perform CHANNEL FUNCTIONAL TEST.	184 days
SR 3.3.6.1.4 Perform CHANNEL CALIBRATION.	18 months
SR 3.3.6.1.5 Perform CHANNEL CALIBRATION.	24 months
SR 3.3.6.1.6 Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.6.1.7 Verify the ISOLATION SYSTEM RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

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Primary Containment Isolation Instrumentation 3.3.6.1

Table 3.3.6.1-1 (page 1 of 4)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level - Low Low, Level 2	1,2,3	2	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ -58 inches
b. Main Steam Line Pressure - Low	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 804 psig
c. Main Steam Line Flow - High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 124.4 psid
d. Condenser Vacuum - Low	1,2(a), 3(a)	2	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ 7.2 inches Hg vacuum
e. Main Steam Tunnel Temperature - High	1,2,3	2	D	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 170°F
f. Main Steam Tunnel Differential Temperature - High	1,2,3	2	D	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 90°F
g. Manual Initiation	1,2,3	4	G	SR 3.3.6.1.6	NA
2. Primary Containment Isolation					
a. Reactor Vessel Water Level - Low, Level 3	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ 9.5 inches
b. Reactor Vessel Water Level - Low Low, Level 2	1,2,3	2	H	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ -58 inches
c. Drywell Pressure - High	1,2,3	2	H	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 1.88 psig
d. Reactor Building Vent Exhaust Plenum Radiation - High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 16.0 mR/hr
e. Manual Initiation	1,2,3	4	G	SR 3.3.6.1.6	NA

(continued)

(a) With any turbine throttle valve not closed.

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Primary Containment Isolation Instrumentation 3.3.6.1

Table 3.3.6.1-1 (page 2 of 4)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. Reactor Core Isolation Cooling (RCIC) System Isolation					
a. RCIC Steam Line Flow - High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 250 inches wg
b. RCIC Steam Line Flow - Time Delay	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 3.00 seconds
c. RCIC Steam Supply Pressure - Low	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ 61 psig
d. RCIC Turbine Exhaust Diaphragm Pressure - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 20 psig
e. RCIC Equipment Room Area Temperature - High	1,2,3	1	F	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 180°F
f. RCIC Equipment Room Area Differential Temperature - High	1,2,3	1	F	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 60°F
g. RWCU/RCIC Steam Line Routing Area Temperature - High	1,2,3	1	F	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 180°F
h. Manual Initiation	1,2,3	1 ^(b)	G	SR 3.3.6.1.6	NA
4. RWCU System Isolation					
a. Differential Flow - High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 67.4 gpm
b. Differential Flow - Time Delay	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 46.5 seconds
c. Blowdown Flow - High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 271.7 gpm

(continued)

(b) RCIC Manual Initiation only inputs into one of the two trip systems.

1. The first part of the document is a list of names and addresses of the members of the committee.



Primary Containment Isolation Instrumentation 3.3.6.1

Table 3.3.6.1-1 (page 3 of 4)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. RWCU System Isolation (continued)					
d. Heat Exchanger Room Area Temperature - High	1,2,3	1	F	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 160°F
e. Heat Exchanger Room Area Ventilation Differential Temperature - High	1,2,3	1	F	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 70°F
f. Pump Room Area Temperature - High	1,2,3	1 per room	F	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 180°F
g. Pump Room Area Ventilation Differential Temperature - High	1,2,3	1 per room	F	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 100°F
h. RWCU/RCIC Line Routing Area Temperature - High	1,2,3	1	F	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 180°F
i. RWCU Line Routing Area Temperature - High	1,2,3	1 per room	F	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	
Room 409, 509 Areas					≤ 175°F
Room 408, 511 Areas					≤ 180°F
j. Reactor Vessel Water Level - Low Low, Level 2	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ -58 inches
k. SLC System Initiation	1,2	2 ^(c)	I	SR 3.3.6.1.6	NA
l. Manual Initiation	1,2,3	2	G	SR 3.3.6.1.6	NA
5. RHR SDC System Isolation					
a. Pump Room Area Temperature - High	3	1 per room ^(d)	F	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 150°F

(continued)

(c) SLC System Initiation only inputs into one of the two trip systems.

(d) Only the inboard trip system required in MODES 1, 2, and 3, as applicable, when the outboard valve control is transferred to the alternate remote shutdown panel and the outboard valve is closed.

Primary Containment Isolation Instrumentation
3.3.6.1

Table 3.3.6.1-1 (page 4 of 4)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. RHR SDC System Isolation (continued)					
b. Pump Room Area Ventilation Differential Temperature - High	3	1 per room ^(d)	F	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 70°F
c. Heat Exchanger Area Temperature - High	3	1 per room ^(d)	F	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	
Room 505 Area					≤ 140°F
Room 507 Area					≤ 160°F
Room 605 Area					≤ 150°F
Room 606 Area					≤ 140°F
d. Reactor Vessel Water Level - Low, Level 3	3,4,5	2 ^{(d)(e)}	J	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ 9.5 inches
e. Reactor Vessel Pressure - High	1,2,3	1 ^(d)	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 135 psig
f. Manual Initiation	1,2,3	2 ^(d)	G	SR 3.3.6.1.6	NA

(d) Only the inboard trip system required in MODES 1, 2, and 3, as applicable, when the outboard valve control is transferred to the alternate remote shutdown panel and the outboard valve is closed.

(e) Only one trip system required in MODES 4 and 5 with RHR Shutdown Cooling System integrity maintained.

Page 2 of 2



3.3 INSTRUMENTATION

3.3.6.2 Secondary Containment Isolation Instrumentation

LCO 3.3.6.2 The secondary containment isolation instrumentation for each Function in Table 3.3.6.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6.2-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Place channel in trip.	12 hours for Function 2 <u>AND</u> 24 hours for Functions other than Function 2
B. One or more automatic Functions with isolation capability not maintained.	B.1 Restore isolation capability.	1 hour
C. Required Action and associated Completion Time not met.	C.1.1 Isolate the associated penetration flow path(s). <u>OR</u>	1 hour (continued)

1. 2. 3. 4. 5. 6. 7. 8. 9. 10.



Secondary Containment Isolation Instrumentation
3.3.6.2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.1.2 Declare associated secondary containment isolation valve(s) inoperable..	1 hour
	<u>AND</u>	
	C.2.1 Place the associated standby gas treatment (SGT) subsystem in operation.	1 hour
	<u>OR</u>	
	C.2.2 Declare associated SGT subsystem inoperable.	1 hour

SURVEILLANCE REQUIREMENTS

-----NOTES-----

1. Refer to Table 3.3.6.2-1 to determine which SRs apply for each Secondary Containment Isolation Function.
 2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains isolation capability.
-

SURVEILLANCE	FREQUENCY
SR 3.3.6.2.1 Perform CHANNEL CHECK.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.3.6.2.2 Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.6.2.3 Perform CHANNEL CALIBRATION.	18 months
SR 3.3.6.2.4 Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

Secondary Containment Isolation Instrumentation 3.3.6.2

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES AND OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low Low, Level 2	1,2,3,(a)	2	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≥ -58 inches
2. Drywell Pressure - High	1,2,3	2	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 1.88 psig
3. Reactor Building Vent Exhaust Plenum Radiation - High	1,2,3, (a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 16.0 mR/hr
4. Manual Initiation	1,2,3, (a),(b)	4	SR 3.3.6.2.4	NA

(a) During operations with a potential for draining the reactor vessel.

(b) During CORE ALTERATIONS, and during movement of irradiated fuel assemblies in the secondary containment.

3.3 INSTRUMENTATION

3.3.7.1 Control Room Emergency Filtration (CREF) System Instrumentation

LCO 3.3.7.1 The CREF System instrumentation for each Function in Table 3.3.7.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.7.1-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.7.1-1 for the channel.	Immediately
B. As required by Required Action A.1 and referenced in Table 3.3.7.1-1.	B.1 Declare associated CREF subsystem inoperable.	1 hour from discovery of loss of CREF initiation capability in both trip systems
	<u>AND</u> B.2 Place channel in trip.	24 hours

(continued)



ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. As required by Required Action A.1 and referenced in Table 3.3.7.1-1.	C.1 Declare associated CREF subsystem inoperable.	1 hour from discovery of loss of CREF initiation capability in both trip systems
	<u>AND</u>	
	C.2 Place channel in trip.	12 hours
D. Required Action and associated Completion Time of Condition B or C not met.	D.1 Place associated CREF subsystem in the pressurization mode of operation.	1 hour
	<u>OR</u>	
	D.2 Declare associated CREF subsystem inoperable.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action A.1 and referenced in Table 3.3.7.1-1.	<p>-----NOTE----- LCO 3.0.4 is not applicable. -----</p>	
	<p>E.1 -----NOTE----- Enter applicable Conditions and Required Actions of LCO 3.7.3, "Control Room Emergency Filtration (CREF) System," when both remote air intakes are isolated. -----</p> <p>Isolate the associated remote air intake.</p>	1 hour from discovery of loss of radiation monitoring capability in a remote air intake
	<p><u>AND</u></p> <p>E.2 Restore channel to OPERABLE status.</p>	<p>7 days from discovery of inoperable channels associated with both remote air intakes</p> <p><u>AND</u></p> <p>30 days</p>
F. Required Action and associated Completion Time of Condition E not met.	F.1 Declare both CREF subsystems inoperable.	Immediately



SURVEILLANCE REQUIREMENTS

-----NOTES-----

1. Refer to Table 3.3.7.1-1 to determine which SRs apply for each CREF System Function.
2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains CREF initiation or radiation monitoring capability, as applicable.

SURVEILLANCE		FREQUENCY
SR 3.3.7.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.7.1.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.7.1.3	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.7.1.4	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

Table 3.3.7.1-1 (page 1 of 1)
Control Room Emergency Filtration System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low Low, Level 2	1,2,3, (a)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≥ -58 inches
2. Drywell Pressure - High	1,2,3	2	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≤ 1.88 psig
3. Reactor Building Vent Exhaust Plenum Radiation - High	1,2,3, (a),(b)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≤ 16.0 mR/hr
4. Main Control Room Ventilation Radiation Monitor	1,2,3, (a),(b)	2 per intake	E	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3	≤ 3800 cpm

(a) During operations with a potential for draining the reactor vessel.

(b) During CORE ALTERATIONS, and during movement of irradiated fuel assemblies in the secondary containment.



3.3 INSTRUMENTATION

3.3.8.1 Loss of Power (LOP) Instrumentation

LCO 3.3.8.1 The LOP instrumentation for each Function in Table 3.3.8.1-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
When the associated diesel generator (DG) is required to be
OPERABLE by LCO 3.8.2, "AC Sources—Shutdown."

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.8.1-1 for the channel.	Immediately
B. As required by Required Action A.1 and referenced in Table 3.3.8.1-1.	B.1 Declare associated DG inoperable.	1 hour from discovery of loss of initiation capability for the associated DG
	<u>AND</u> B.2 Restore channel to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. As required by Required Action A.1 and referenced in Table 3.3.8.1-1.	C.1 Place channel in trip.	1 hour
D. Required Action and associated Completion Time of Condition B or C not met.	D.1 Declare associated DG inoperable.	Immediately
	<u>OR</u>	
	-----NOTE----- Only applicable for Functions 1.c and 1.d. -----	
	D.2.1 Open offsite circuit supply breaker to associated 4.16 kV ESF bus.	Immediately
	<u>AND</u>	
	D.2.2 Declare associated offsite circuit inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTES-----

1. Refer to Table 3.3.8.1-1 to determine which SRs apply for each LOP Function.
 2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 2 hours provided the associated Function maintains initiation capability.
-

SURVEILLANCE	FREQUENCY
SR 3.3.8.1.1 Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.8.1.2 Perform CHANNEL CALIBRATION.	18 months
SR 3.3.8.1.3 Perform CHANNEL CALIBRATION.	24 months
SR 3.3.8.1.4 Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months



Table 3.3.8.1-1 (page 1 of 1)
Loss of Power Instrumentation

FUNCTION	REQUIRED CHANNELS PER DIVISION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Divisions 1 and 2 - 4.16 kV Emergency Bus Undervoltage				
a. TR-S Loss of Voltage - 4.16 kV Basis	2	B	SR 3.3.8.1.2 SR 3.3.8.1.4	≥ 2450 V and ≤ 3135 V
b. TR-S Loss of Voltage - Time Delay	2	B	SR 3.3.8.1.3 SR 3.3.8.1.4	≥ 2.95 seconds and ≤ 7.1 seconds
c. TR-B Loss of Voltage - 4.16 kV Basis	1	C	SR 3.3.8.1.3 SR 3.3.8.1.4	≥ 2450 V and ≤ 3135 V
d. TR-B Loss of Voltage - Time Delay	1	C	SR 3.3.8.1.3 SR 3.3.8.1.4	≥ 3.06 seconds and ≤ 9.28 seconds
e. Degraded Voltage - 4.16 kV Basis	2(a)	C	SR 3.3.8.1.1 SR 3.3.8.1.2 SR 3.3.8.1.4	≥ 3685 V and ≤ 3755 V
f. Degraded Voltage - Primary Time Delay	2(a)	C	SR 3.3.8.1.1 SR 3.3.8.1.2 SR 3.3.8.1.4	≥ 5.0 seconds and ≤ 5.3 seconds
g. Degraded Voltage - Secondary Time Delay	1	C	SR 3.3.8.1.2 SR 3.3.8.1.4	≥ 2.63 seconds and ≤ 3.39 seconds
2. Division 3 - 4.16 kV Emergency Bus Undervoltage				
a. Loss of Voltage - 4.16 kV Basis	2	B	SR 3.3.8.1.2 SR 3.3.8.1.4	≥ 2450 V and ≤ 3135 V
b. Loss of Voltage - Time Delay	2	B	SR 3.3.8.1.3 SR 3.3.8.1.4	≥ 1.87 seconds and ≤ 3.73 seconds
c. Degraded Voltage - 4.16 kV Basis	2	C	SR 3.3.8.1.2 SR 3.3.8.1.4	≥ 3685 V and ≤ 3755 V
d. Degraded Voltage - Time Delay	2	C	SR 3.3.8.1.2 SR 3.3.8.1.4	≥ 7.36 seconds and ≤ 8.34 seconds

(a) The Degraded Voltage - 4.16 kV Basis and - Primary Time Delay Functions must be associated with one another.



3.3 INSTRUMENTATION

3.3.8.2 Reactor Protection System (RPS) Electric Power Monitoring

LCO 3.3.8.2 Two RPS electric power monitoring assemblies shall be OPERABLE for each inservice RPS motor generator set or alternate power supply that supports equipment required to be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODES 4 and 5 with both residual heat removal (RHR) shutdown cooling (SDC) suction isolation valves open,
MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both required inservice power supplies with one electric power monitoring assembly inoperable.	A.1 Remove associated inservice power supply(s) from service.	72 hours
B. One or both required inservice power supplies with both electric power monitoring assemblies inoperable.	B.1 Remove associated inservice power supply(s) from service.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4.	12 hours 36 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met in MODE 4 or 5 with both RHR SDC suction isolation valves open.	D.1 Initiate action to restore one electric power monitoring assembly to OPERABLE status for inservice power supply(s) supplying required instrumentation.	Immediately
	<u>OR</u> D.2 Initiate action to isolate the RHR SDC System.	Immediately
E. Required Action and associated Completion Time of Condition A or B not met in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies.	E.1 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

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D SURVEILLANCE REQUIREMENTS

-----NOTE-----
When an RPS electric power monitoring assembly is placed in an inoperable status solely for performance of required Surveillances, entry into the associated Conditions and Required Actions may be delayed for up to 6 hours provided the other RPS electric power monitoring assembly for the associated power supply maintains trip capability.

SURVEILLANCE	FREQUENCY
<p>SR 3.3.8.2.1 -----NOTE----- Only required to be performed prior to entering MODE 2 or 3 from MODE 4, when in MODE 4 for ≥ 24 hours. ----- Perform CHANNEL FUNCTIONAL TEST.</p>	184 days
<p>SR 3.3.8.2.2 Perform CHANNEL CALIBRATION. The Allowable Values shall be:</p> <ul style="list-style-type: none"> a. Overvoltage ≤ 133.8 V, with time delay ≤ 3.46 seconds; b. Undervoltage ≥ 110.8 V, with time delay ≤ 3.46 seconds; and c. Underfrequency ≥ 57 Hz, with time delay ≤ 3.46 seconds. 	18 months
<p>SR 3.3.8.2.3 Perform a system functional test.</p>	18 months

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1 Two recirculation loops with matched flows shall be in operation in the "Unrestricted" Region of the power-to-flow map specified in the COLR.

OR

One recirculation loop shall be in operation in the "Unrestricted" Region of the power-to-flow map specified in the COLR with the following limits applied when the associated LCO is applicable:

- a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits specified in the COLR; and
- b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. No, one, or two recirculation loops in operation in Region A of the power-to-flow map.	A.1 Place the reactor mode switch in the shutdown position.	15 minutes

(continued)



ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. Two recirculation loops in operation in Region B or C of the power-to-flow map.</p> <p><u>OR</u></p> <p>One recirculation loop in operation in Region C of the power-to-flow map.</p>	<p>B.1 Verify the stability monitoring system decay ratio < 0.75.</p>	<p>15 minutes</p> <p><u>AND</u></p> <p>Once per hour thereafter</p>
<p>C. Required Action and associated Completion Time of Condition B not met.</p>	<p>C.1 Restore operation to the "Unrestricted" Region of the power-to-flow map.</p>	<p>1 hour</p>
<p>D. One recirculation loop in operation in Region B of the power-to-flow map.</p>	<p>D.1 Restore operation to Region C or the "Unrestricted" Region of the power-to-flow map.</p>	<p>1 hour</p>
<p>E. Recirculation loop flow mismatch not within limits.</p>	<p>E.1 Declare the recirculation loop with lower flow to be "not in operation."</p>	<p>2 hours</p>
<p>F. Requirements of the LCO not met for reasons other than Condition A, B, C, D, or E.</p>	<p>F.1 Satisfy the requirements of the LCO.</p>	<p>4 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>G. Required Action and associated Completion Time of Condition F not met.</p> <p><u>OR</u></p> <p>No recirculation loops in operation in a Region other than Region A of the power-to-flow map.</p>	G.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.1.1 -----NOTE----- Not required to be performed until 24 hours after both recirculation loops are in operation. -----</p> <p>Verify recirculation loop drive flow mismatch with both recirculation loops in operation is:</p> <p>a. $\leq 10\%$ of rated recirculation loop drive flow when operating at $< 70\%$ of rated core flow; and</p> <p>b. $\leq 5\%$ of rated recirculation loop drive flow when operating at $\geq 70\%$ of rated core flow.</p>	24 hours

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1. 2. 3. 4. 5. 6. 7. 8. 9. 10.

D SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.1.2	Verify operation is in the "Unrestricted" Region of the power-to-flow map specified in the COLR.	24 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 Jet Pumps

LCO 3.4.2 All jet pumps shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more jet pumps inoperable.	A.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.2.1 -----NOTES-----</p> <ol style="list-style-type: none"> 1. Not required to be performed until 4 hours after associated recirculation loop is in operation. 2. Not required to be performed until 24 hours after > 25% RTP. <p>-----</p> <p>Verify at least two of the following criteria (a, b, and c) are satisfied for each operating recirculation loop:</p> <ol style="list-style-type: none"> a. Recirculation loop drive flow versus recirculation pump speed differs by $\leq 10\%$ from established patterns. b. Recirculation loop drive flow versus total core flow differs by $\leq 10\%$ from established patterns. c. Each jet pump diffuser to lower plenum differential pressure differs by $\leq 20\%$ from established patterns, or each jet pump flow differs by $\leq 10\%$ from established patterns. 	<p>24 hours</p>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety/Relief Valves (SRVs) — \geq 25% RTP

LCO 3.4.3 The safety function of 12 SRVs shall be OPERABLE, with two SRVs in the lowest two lift setpoint groups OPERABLE.

APPLICABILITY: THERMAL POWER \geq 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required SRVs inoperable.	A.1 Reduce THERMAL POWER to $<$ 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the required SRVs are as follows:	In accordance with the Inservice Testing Program
	<u>Number of SRVs</u>	
	<u>Setpoint (psig)</u>	
	2	
	4	
	4	
	4	
	4	
	4	
	4	
	4	
SR 3.4.3.2 Verify each required SRV opens when manually actuated.		24 months



3.4.4 Safety/Relief Valves (SRVs) — < 25% RTP

LCO 3.4.4 The safety function of four SRVs shall be OPERABLE.

APPLICABILITY: MODE 1 with THERMAL POWER < 25% RTP,
MODES 2 and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required SRVs inoperable.	A.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	A.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY												
SR 3.4.4.1	Verify the safety function lift setpoints of the required SRVs are as follows:	In accordance with the Inservice Testing Program												
	<table><tr><th><u>Number of SRVs</u></th><th><u>Setpoint (psig)</u></th></tr><tr><td>2</td><td>1165 ± 34.9</td></tr><tr><td>4</td><td>1175 ± 35.2</td></tr><tr><td>4</td><td>1185 ± 35.5</td></tr><tr><td>4</td><td>1195 ± 35.8</td></tr><tr><td>4</td><td>1205 ± 36.1</td></tr></table>		<u>Number of SRVs</u>	<u>Setpoint (psig)</u>	2	1165 ± 34.9	4	1175 ± 35.2	4	1185 ± 35.5	4	1195 ± 35.8	4	1205 ± 36.1
<u>Number of SRVs</u>	<u>Setpoint (psig)</u>													
2	1165 ± 34.9													
4	1175 ± 35.2													
4	1185 ± 35.5													
4	1195 ± 35.8													
4	1205 ± 36.1													

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.4.4.2 -----NOTE----- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. ----- Verify each required SRV opens when manually actuated.</p>	<p>24 months</p>

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Operational LEAKAGE

LC0 3.4.5 RCS operational LEAKAGE shall be limited to:

- a. No pressure boundary LEAKAGE;
- b. ≤ 5 gpm unidentified LEAKAGE;
- c. ≤ 25 gpm total LEAKAGE averaged over the previous 24 hour period; and
- d. ≤ 2 gpm increase in unidentified LEAKAGE within the previous 24 hour period in MODE 1.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. Unidentified LEAKAGE not within limit.</p> <p><u>OR</u></p> <p>Total LEAKAGE not within limit.</p>	<p>A.1 Reduce LEAKAGE to within limits.</p>	4 hours
<p>B. Unidentified LEAKAGE increase not within limit.</p>	<p>B.1 Reduce unidentified LEAKAGE increase to within limit.</p> <p><u>OR</u></p>	<p>4 hours</p> <p>(continued)</p>

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ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Verify source of unidentified LEAKAGE increase is not service sensitive type 304 or type 316 austenitic stainless steel.	4 hours
C. Required Action and associated Completion Time of Condition A or B not met. <u>OR</u> Pressure boundary LEAKAGE exists.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.5.1 Verify RCS unidentified and total LEAKAGE and unidentified LEAKAGE increase are within limits.	12 hours

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Pressure Isolation Valve (PIV) Leakage

LCO 3.4.6 The leakage from each RCS PIV shall be within limit.

APPLICABILITY: MODES 1 and 2,
MODE 3, except valves in the residual heat removal shutdown
cooling flowpath when in, or during transition to or
from, the shutdown cooling mode of operation.

ACTIONS

- NOTES-----
1. Separate Condition entry is allowed for each flow path.
 2. Enter applicable Conditions and Required Actions for systems made inoperable by PIVs.
-

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more flow paths with leakage from one or more RCS PIVs not within limit.	-----NOTE----- Each check valve used to satisfy Required Action A.1 shall have been verified to meet SR 3.4.6.1 and be in the reactor coolant pressure boundary. -----	4 hours
	A.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, de-activated automatic, or check valve.	

(continued)

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.6.1 -----NOTE----- Only required to be performed in MODES 1 and 2. -----</p> <p>Verify equivalent leakage of each RCS PIV is ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm, at an RCS pressure of 1035 psig. The actual test pressure shall be ≥ 935 psig.</p>	In accordance with Inservice Testing Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Leakage Detection Instrumentation

LCO 3.4.7 The following RCS leakage detection instrumentation shall be OPERABLE:

- a. Drywell floor drain sump flow monitoring system; and
- b. One channel of either drywell atmospheric particulate or atmospheric gaseous monitoring system..

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Drywell floor drain sump flow monitoring system inoperable.	-----NOTE----- LCO 3.0.4 is not applicable. -----	
	A.1 Restore drywell floor drain sump flow monitoring system to OPERABLE status.	30 days
B. Required drywell atmospheric monitoring system inoperable.	-----NOTE----- LCO 3.0.4 is not applicable. -----	
	B.1 Analyze grab samples of drywell atmosphere.	Once per 12 hours
	<u>AND</u> B.2 Restore required drywell atmospheric monitoring system to OPERABLE status.	30 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3.	12 hours
	<u>AND</u> C.2 Be in MODE 4.	36 hours
D. All required leakage detection systems inoperable.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTE-----

When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the other required leakage detection instrumentation is OPERABLE.

SURVEILLANCE	FREQUENCY
SR 3.4.7.1 Perform CHANNEL CHECK of required drywell atmospheric monitoring system.	12 hours
SR 3.4.7.2 Perform CHANNEL FUNCTIONAL TEST of required leakage detection instrumentation.	31 days
SR 3.4.7.3 Perform CHANNEL CALIBRATION of required leakage detection instrumentation.	18 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Specific Activity

LCO 3.4.8 The specific activity of the reactor coolant shall be limited to DOSE EQUIVALENT I-131 specific activity $\leq 0.2 \mu\text{Ci/gm}$.

APPLICABILITY: MODE 1,
MODES 2 and 3 with any main steam line not isolated.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor coolant specific activity $> 0.2 \mu\text{Ci/gm}$ and $\leq 4.0 \mu\text{Ci/gm}$ DOSE EQUIVALENT I-131.	-----NOTE----- LCO 3.0.4 is not applicable. -----	
	A.1 Determine DOSE EQUIVALENT I-131.	Once per 4 hours
	<u>AND</u> A.2 Restore DOSE EQUIVALENT I-131 to within limits.	48 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Reactor coolant specific activity $> 4.0 \mu\text{Ci/gm}$ DOSE EQUIVALENT I-131.	B.1 Determine DOSE EQUIVALENT I-131.	Once per 4 hours
	<u>AND</u> B.2.1 Isolate all main steam lines.	12 hours
	<u>OR</u>	(continued)

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ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2.2.1 Be in MODE 3.	12 hours
	<u>AND</u> B.2.2.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.8.1 -----NOTE----- Only required to be performed in MODE 1. ----- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity is $\leq 0.2 \mu\text{Ci/gm.}$	7 days



3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Residual Heat Removal (RHR) Shutdown Cooling System — Hot Shutdown

LCO 3.4.9 Two RHR shutdown cooling subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one RHR shutdown cooling subsystem shall be in operation.

-----NOTES-----

1. Both RHR shutdown cooling subsystems and recirculation pumps may be removed from operation for up to 2 hours per 8 hour period.
 2. One RHR shutdown cooling subsystem may be inoperable for up to 2 hours for performance of Surveillances.
-

APPLICABILITY: MODE 3 with reactor steam dome pressure less than the RHR cut-in permissive pressure.

ACTIONS

-----NOTES-----

1. LCO 3.0.4 is not applicable.
 2. Separate Condition entry is allowed for each RHR shutdown cooling subsystem.
-

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two RHR shutdown cooling subsystems inoperable.	A.1 Initiate action to restore RHR shutdown cooling subsystem to OPERABLE status.	Immediately
	<u>AND</u>	(continued)



ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem.	1 hour
	<u>AND</u> A.3 Be in MODE 4.	24 hours
B. No RHR shutdown cooling subsystem in operation. <u>AND</u> No recirculation pump in operation.	B.1 Initiate action to restore one RHR shutdown cooling subsystem or one recirculation pump to operation.	Immediately
	<u>AND</u> B.2 Verify reactor coolant circulation by an alternate method.	1 hour from discovery of no reactor coolant circulation
	<u>AND</u> B.3 Monitor reactor coolant temperature and pressure.	<u>AND</u> Once per 12 hours thereafter Once per hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.9.1 -----NOTE----- Not required to be met until 2 hours after reactor steam dome pressure is less than the RHR cut-in permissive pressure. -----</p> <p>Verify one RHR shutdown cooling subsystem or recirculation pump is operating.</p>	<p>12 hours</p>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown

LCO 3.4.10 Two RHR shutdown cooling subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one RHR shutdown cooling subsystem shall be in operation.

-----NOTES-----

1. Both RHR shutdown cooling subsystems and recirculation pumps may be removed from operation for up to 2 hours per 8 hour period.
 2. One RHR shutdown cooling subsystem may be inoperable for up to 2 hours for the performance of Surveillances.
-

APPLICABILITY: MODE 4.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each RHR shutdown cooling subsystem.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two RHR shutdown cooling subsystems inoperable.	A.1 Verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem.	1 hour <u>AND</u> Once per 24 hours thereafter

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. No RHR shutdown cooling subsystem in operation. <u>AND</u> No recirculation pump in operation.	B.1 Verify reactor coolant circulating by an alternate method.	1 hour from discovery of no reactor coolant circulation <u>AND</u> Once per 12 hours thereafter
	<u>AND</u> B.2 Monitor reactor coolant temperature and pressure.	Once per hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.10.1 Verify one RHR shutdown cooling subsystem or recirculation pump is operating.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.11 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation loop temperature requirements shall be maintained within limits.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. -----NOTE----- Required Action A.2 shall be completed if this Condition is entered. ----- Requirements of the LCO not met in MODE 1, 2, or 3.</p>	A.1 Restore parameter(s) to within limits.	30 minutes
	<p><u>AND</u></p> <p>A.2 Determine RCS is acceptable for continued operation.</p>	72 hours
<p>B. Required Action and associated Completion Time of Condition A not met.</p>	B.1 Be in MODE 3.	12 hours
	<p><u>AND</u></p> <p>B.2 Be in MODE 4.</p>	36 hours

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CONDITION	REQUIRED ACTION	COMPLETION TIME
C. -----NOTE----- Required Action C.2 shall be completed if this Condition is entered. ----- Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.1 Initiate action to restore parameter(s) to within limits. <u>AND</u> C.2 Determine RCS is acceptable for operation.	Immediately Prior to entering MODE 2 or 3

SURVEILLANCE	FREQUENCY
<p>SR 3.4.11.1 -----NOTE----- Only required to be performed during RCS heatup and cooldown operations, and RCS inservice leak and hydrostatic testing. -----</p> <p>Verify:</p> <ol style="list-style-type: none"> RCS pressure and RCS temperature are within the applicable limits specified in Figures 3.4.11-1, 3.4.11-2, and 3.4.11-3; RCS heatup and cooldown rates are $\leq 100^{\circ}\text{F}$ in any 1 hour period; and RCS temperature change during inservice leak and hydrostatic testing is $\leq 20^{\circ}\text{F}$ in any 1 hour period when the RCS pressure and RCS temperature are not within the limits of Figure 3.4.11-2. 	<p>30 minutes</p>

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.4.11.2 Verify RCS pressure and RCS temperature are within the criticality limits specified in Figure 3.4.11-3.</p>	<p>Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality</p>
<p>SR 3.4.11.3 -----NOTE----- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. ----- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is $\leq 145^{\circ}\text{F}$.</p>	<p>Once within 15 minutes prior to each startup of a recirculation pump</p>
<p>SR 3.4.11.4 -----NOTE----- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. ----- Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is $\leq 50^{\circ}\text{F}$.</p>	<p>Once within 15 minutes prior to each startup of a recirculation pump</p>

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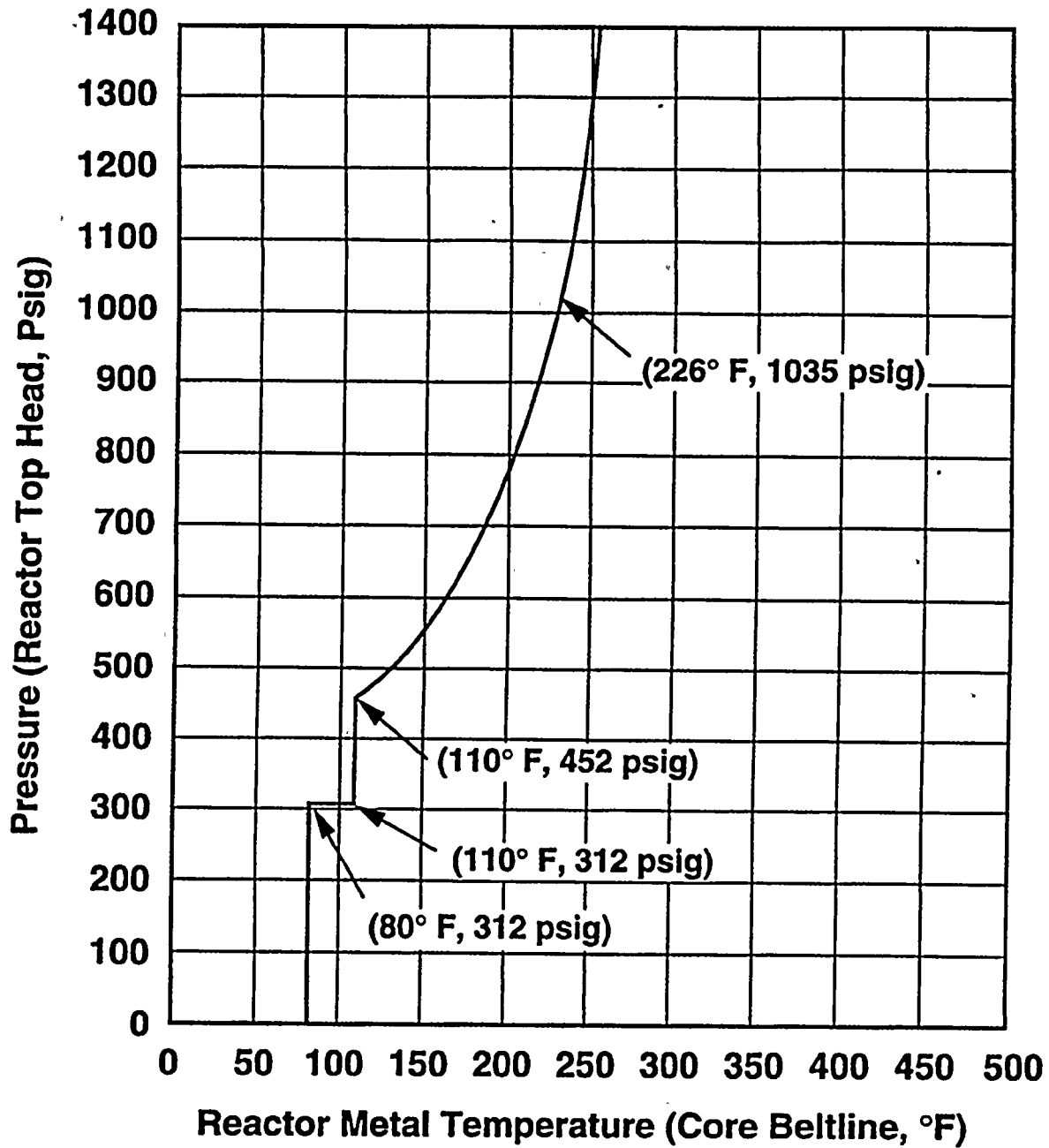
SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.4.11.5 -----NOTE----- Only required to be met in single loop operation with THERMAL POWER \leq 25% RTP or the operating recirculation loop flow \leq 10% rated loop flow. -----</p> <p>Verify the difference between the bottom head coolant temperature and the RPV coolant temperature is \leq 145°F.</p>	<p>Once within 15 minutes prior to an increase in THERMAL POWER or an increase in loop flow</p>
<p>SR 3.4.11.6 -----NOTE----- Only required to be met in single loop operation when the idle recirculation loop is not isolated from the RPV, and with THERMAL POWER \leq 25% RTP or the operating recirculation loop flow \leq 10% rated loop flow. -----</p> <p>Verify the difference between the reactor coolant temperature in the recirculation loop not in operation and the RPV coolant temperature is \leq 50°F.</p>	<p>Once within 15 minutes prior to an increase in THERMAL POWER or an increase in loop flow</p>

(continued)

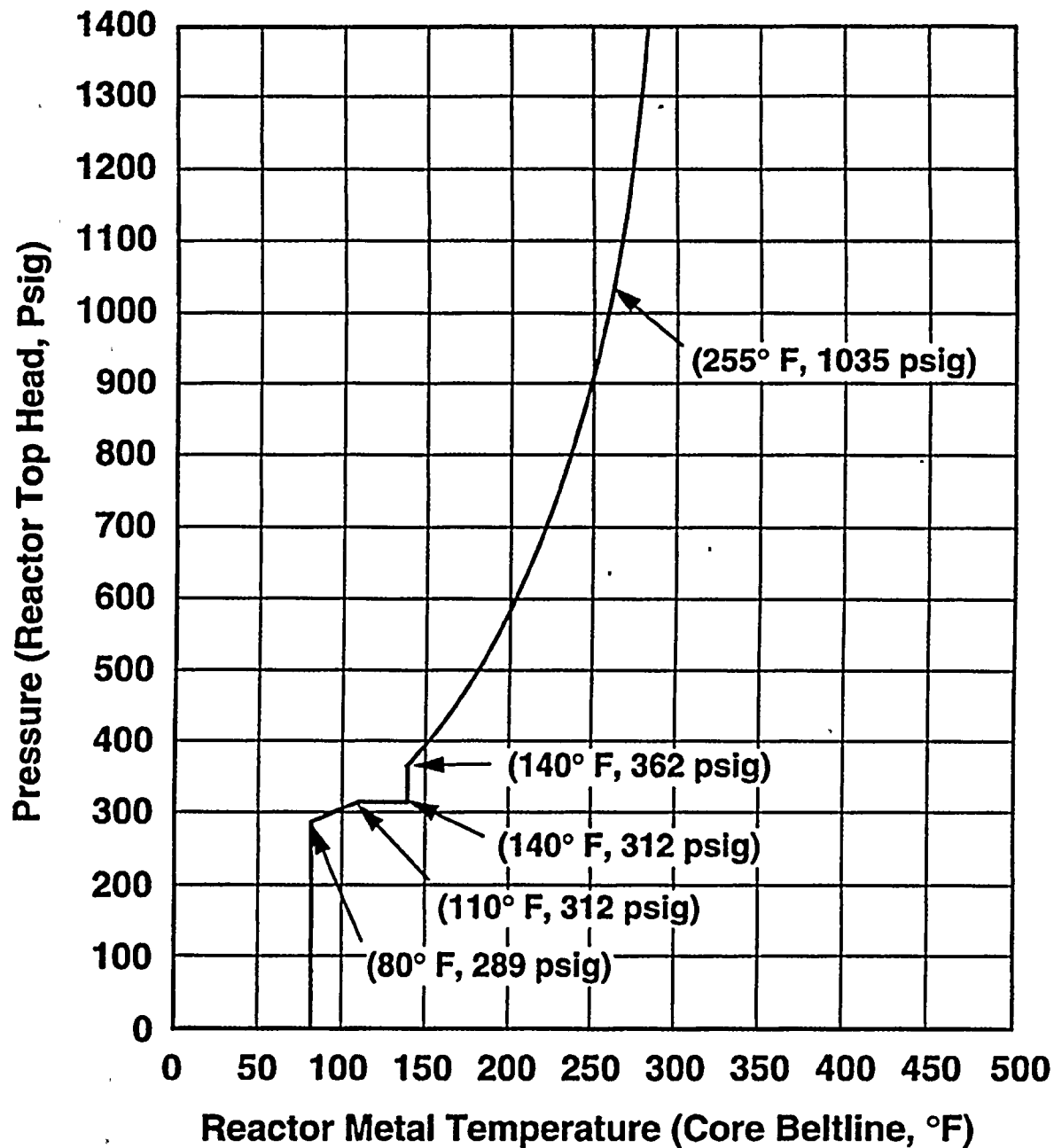
SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.4.11.7 -----NOTE----- Only required to be performed when tensioning the reactor vessel head bolting studs. -----</p> <p>Verify reactor vessel flange and head flange temperatures are $\geq 80^{\circ}\text{F}$.</p>	<p>30 minutes</p>
<p>SR 3.4.11.8 -----NOTE----- Not required to be performed until 30 minutes after RCS temperature $\leq 90^{\circ}\text{F}$ in MODE 4. -----</p> <p>Verify reactor vessel flange and head flange temperatures are $\geq 80^{\circ}\text{F}$.</p>	<p>30 minutes</p>
<p>SR 3.4.11.9 -----NOTE----- Not required to be performed until 12 hours after RCS temperature $\leq 100^{\circ}\text{F}$ in MODE 4. -----</p> <p>Verify reactor vessel flange and head flange temperatures are $\geq 80^{\circ}\text{F}$.</p>	<p>12 hours</p>



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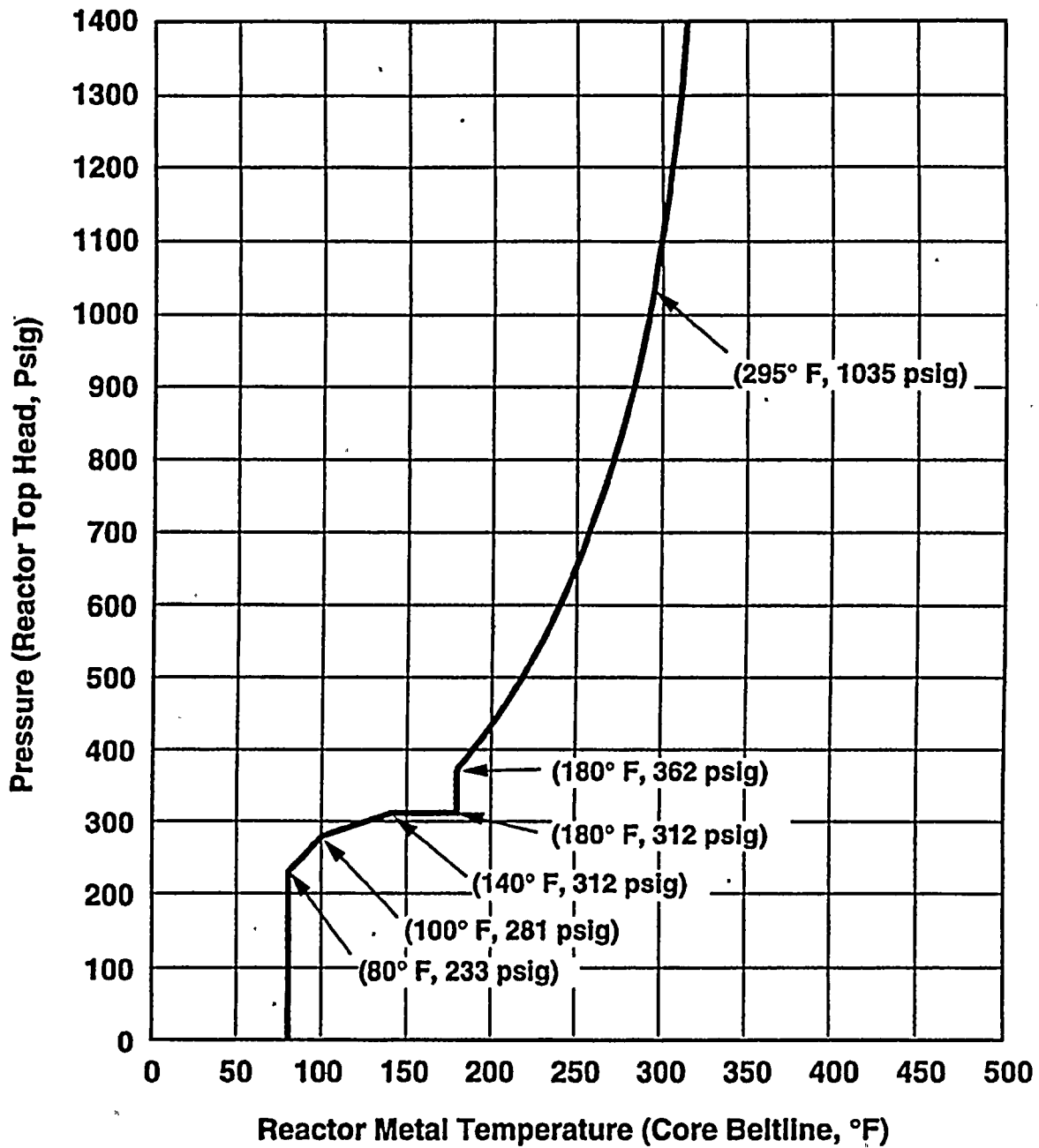
Figure 3.4.11-1 (Page 1 of 1)
Inservice Leak and Hydrostatic Testing Curve



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Figure 3.4-11-2 (Page 1 of 1)
Non-Nuclear Heating and Cooldown Curve





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Figure 3.4.11-3 (Page 1 of 1)
Nuclear Heating and Cooldown Curve



3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Reactor Steam Dome Pressure

LCO 3.4.12 The reactor steam dome pressure shall be ≤ 1035 psig.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor steam dome pressure not within limit.	A.1 Restore reactor steam dome pressure to within limit.	15 minutes
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.12.1 Verify reactor steam dome pressure is ≤ 1035 psig.	12 hours

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

3.5.1 ECCS—Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except ADS valves are not required to be
OPERABLE with reactor steam dome pressure \leq 150 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	7 days
B. High Pressure Core Spray (HPCS) System inoperable.	B.1 Verify by administrative means RCIC System is OPERABLE when RCIC System is required to be OPERABLE.	Immediately
	<u>AND</u> B.2 Restore HPCS System to OPERABLE status.	14 days

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. Two ECCS injection subsystems inoperable.</p> <p><u>OR</u></p> <p>One ECCS injection and one ECCS spray subsystem inoperable.</p>	<p>C.1 Restore one ECCS injection/spray subsystem to OPERABLE status.</p>	72 hours
<p>D. Required Action and associated Completion Time of Condition A, B, or C not met.</p>	<p>D.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>D.2 Be in MODE 4.</p>	<p>12 hours</p> <p>36 hours</p>
<p>E. One required ADS valve inoperable.</p>	<p>E.1 Restore ADS valve to OPERABLE status.</p>	14 days
<p>F. One required ADS valve inoperable.</p> <p><u>AND</u></p> <p>One low pressure ECCS injection/spray subsystem inoperable.</p>	<p>F.1 Restore ADS valve to OPERABLE status.</p> <p><u>OR</u></p> <p>F.2 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.</p>	<p>72 hours</p> <p>72 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>G. Required Action and associated Completion Time of Condition E or F not met.</p> <p><u>OR</u></p> <p>Two or more required ADS valves inoperable.</p>	<p>G.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>G.2 Reduce reactor steam dome pressure to ≤ 150 psig.</p>	<p>12 hours</p> <p>36 hours</p>
<p>H. HPCS and Low Pressure Core Spray (LPCS) Systems inoperable.</p> <p><u>OR</u></p> <p>Three or more ECCS injection/spray subsystems inoperable.</p> <p><u>OR</u></p> <p>HPCS System and one or more required ADS valves inoperable.</p> <p><u>OR</u></p> <p>Two or more ECCS injection/spray subsystems and one or more required ADS valves inoperable.</p>	<p>H.1 Enter LCO 3.0.3.</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.5.1.1 Verify, for each ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.</p>	<p>31 days</p>
<p>SR 3.5.1.2 -----NOTE----- Low pressure coolant injection (LPCI) subsystems may be considered OPERABLE during alignment and operation for decay heat removal with reactor steam dome pressure less than the residual heat removal cut-in permissive pressure in MODE 3, if capable of being manually realigned and not otherwise inoperable. ----- Verify each ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.</p>	<p>31 days</p>
<p>SR 3.5.1.3 Verify ADS accumulator backup compressed gas system average pressure in the required bottles is ≥ 2200 psig.</p>	<p>31 days</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE			FREQUENCY												
SR 3.5.1.4	Verify each ECCS pump develops the specified flow rate with the specified developed head. <table><thead><tr><th><u>SYSTEM</u></th><th><u>FLOW RATE</u></th><th><u>TOTAL DEVELOPED HEAD</u></th></tr></thead><tbody><tr><td>LPCS</td><td>≥ 6350 gpm</td><td>≥ 128 psid</td></tr><tr><td>LPCI</td><td>≥ 7450 gpm</td><td>≥ 26 psid</td></tr><tr><td>HPCS</td><td>≥ 6350 gpm</td><td>≥ 200 psid</td></tr></tbody></table>		<u>SYSTEM</u>	<u>FLOW RATE</u>	<u>TOTAL DEVELOPED HEAD</u>	LPCS	≥ 6350 gpm	≥ 128 psid	LPCI	≥ 7450 gpm	≥ 26 psid	HPCS	≥ 6350 gpm	≥ 200 psid	In accordance with the Inservice Testing Program
<u>SYSTEM</u>	<u>FLOW RATE</u>	<u>TOTAL DEVELOPED HEAD</u>													
LPCS	≥ 6350 gpm	≥ 128 psid													
LPCI	≥ 7450 gpm	≥ 26 psid													
HPCS	≥ 6350 gpm	≥ 200 psid													
SR 3.5.1.5	-----NOTE----- Vessel injection/spray may be excluded. ----- Verify each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.		24 months												
SR 3.5.1.6	-----NOTE----- Valve actuation may be excluded. ----- Verify the ADS actuates on an actual or simulated automatic initiation signal.		24 months												
SR 3.5.1.7	-----NOTE----- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. ----- Verify each required ADS valve opens when manually actuated.		24 months on a STAGGERED TEST BASIS for each valve solenoid												

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

3.5.2 ECCS—Shutdown

LCO 3.5.2 Two ECCS injection/spray subsystems shall be OPERABLE.

APPLICABILITY: MODE 4,
MODE 5 except with the spent fuel storage pool gates removed
and water level \geq 22 ft over the top of the reactor
pressure vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required ECCS injection/spray subsystem inoperable.	A.1 Restore required ECCS injection/spray subsystem to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action to suspend operations with a potential for draining the reactor vessel (OPDRVs).	Immediately
C. Two required ECCS injection/spray subsystems inoperable.	C.1 Initiate action to suspend OPDRVs.	Immediately
	<u>AND</u> C.2 Restore one ECCS injection/spray subsystem to OPERABLE status.	4 hours

(continued)



ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action C.2 and associated Completion Time not met.	D.1 Initiate action to restore secondary containment to OPERABLE status.	Immediately
	<u>AND</u>	
	D.2 Initiate action to restore one standby gas treatment subsystem to OPERABLE status.	Immediately
	<u>AND</u>	
	D.3 Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.5.2.1 Verify, for each required low pressure ECCS injection/spray subsystem, the suppression pool water level is \geq 18 ft 6 inches.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.5.2.2 Verify, for the required High Pressure Core Spray (HPCS) System, the:</p> <p> a. Suppression pool water level is ≥ 18 ft 6 inches; or</p> <p> b. Condensate storage tank (CST) water level is ≥ 13.25 ft in a single CST or ≥ 7.6 ft in each CST.</p>	12 hours
<p>SR 3.5.2.3 Verify, for each required ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.</p>	31 days
<p>SR 3.5.2.4 -----NOTE----- One low pressure coolant injection (LPCI) subsystem may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned and not otherwise inoperable. -----</p> <p>Verify each required ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.</p>	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE			FREQUENCY
SR 3.5.2.5	Verify each required ECCS pump develops the specified flow rate with the specified developed head.		In accordance with the Inservice Testing Program
		TOTAL DEVELOPED HEAD	
	<u>SYSTEM</u>	<u>FLOW RATE</u>	
	LPCS	≥ 6350 gpm	
	LPCI	≥ 7450 gpm	
	HPCS	≥ 6350 gpm	≥ 200 psid
SR 3.5.2.6	-----NOTE----- Vessel injection/spray may be excluded. ----- Verify each required ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.		24 months



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

3.5.3 RCIC System

LCO 3.5.3 The RCIC System shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 with reactor steam dome pressure > 150 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCIC System inoperable.	A.1 Verify by administrative means High Pressure Core Spray System is OPERABLE.	Immediately
	<u>AND</u> A.2 Restore RCIC System to OPERABLE status.	14 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours
	<u>AND</u> B.2 Reduce reactor steam dome pressure to ≤ 150 psig.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	Verify the RCIC System piping is filled with water from the pump discharge valve to the injection valve.	31 days
SR 3.5.3.2	Verify each RCIC System manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.3.3	-----NOTE----- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. Verify, with reactor pressure ≤ 1035 psig and ≥ 935 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure.	92 days
SR 3.5.3.4	-----NOTE----- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. Verify, with reactor pressure ≤ 165 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.5.3.5 -----NOTE----- Vessel injection may be excluded. ----- Verify the RCIC System actuates on an actual or simulated automatic initiation signal.</p>	<p>24 months</p>



3.6 CONTAINMENT SYSTEMS

3.6.1.1 Primary Containment

LCO 3.6.1.1 Primary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Primary containment inoperable.	A.1 Restore primary containment to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.1.1.1 Perform required visual examinations and leakage rate testing except for primary containment air lock testing, in accordance with the Primary Containment Leakage Rate Testing Program.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.1.2 Verify drywell to suppression chamber bypass leakage rate is less than or equal to the equivalent leakage rate through an orifice 0.005 ft ² at an initial differential pressure of ≥ 1.5 psid.	24 months <u>AND</u> -----NOTE----- Only required after two consecutive tests fail and continues until two consecutive tests pass ----- 12 months

3.6 CONTAINMENT SYSTEMS

3.6.1.2 Primary Containment Air Lock

LCO 3.6.1.2 The primary containment air lock shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

- NOTES-----
1. Entry and exit is permissible to perform repairs of the air lock components.
 2. Enter applicable Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," when air lock leakage results in exceeding overall containment leakage rate acceptance criteria.
-

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One primary containment air lock door inoperable.	<p>-----NOTES-----</p> <ol style="list-style-type: none"> 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days under administrative controls. <p>-----</p>	
	<p>A.1 Verify the OPERABLE door is closed.</p> <p><u>AND</u></p>	<p>1 hour</p> <p>(continued)</p>

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ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Lock the OPERABLE door closed.	24 hours
	<p><u>AND</u></p> <p>A.3 -----NOTE----- Air lock doors in high radiation areas or areas with limited access due to inerting may be verified locked closed by administrative means. -----</p> <p>Verify the OPERABLE door is locked closed.</p>	
B. Primary containment air lock interlock mechanism inoperable.	<p>-----NOTES-----</p> <p>1. Required Actions B.1, B.2, and B.3 are not applicable if both doors in the air lock are inoperable and Condition C is entered.</p> <p>2. Entry into and exit from primary containment is permissible under the control of a dedicated individual.</p> <p>-----</p>	1 hour
	<p>B.1 Verify an OPERABLE door is closed.</p> <p><u>AND</u></p>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Lock an OPERABLE door closed.	24 hours
	<p><u>AND</u></p> <p>B.3 -----NOTE----- Air lock doors in high radiation areas or areas with limited access due to inerting may be verified locked closed by administrative means. -----</p> <p>Verify an OPERABLE door is locked closed.</p>	Once per 31 days
C. Primary containment air lock inoperable for reasons other than Condition A or B.	C.1 Initiate action to evaluate primary containment overall leakage rate per LCO 3.6.1.1, using current air lock test results.	Immediately
	<p><u>AND</u></p> <p>C.2 Verify a door is closed.</p>	1 hour
	<p><u>AND</u></p> <p>C.3 Restore air lock to OPERABLE status.</p>	24 hours

(continued)



ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.6.1.2.1 -----NOTES-----</p> <ol style="list-style-type: none"> 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1.1. <p>-----</p> <p>Perform required primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.</p>	<p>In accordance with the Primary Containment Leakage Rate Testing Program</p>
<p>SR 3.6.1.2.2 Verify only one door in the primary containment air lock can be opened at a time.</p>	<p>24 months</p>

3.6 CONTAINMENT SYSTEMS

3.6.1.3 Primary Containment Isolation Valves (PCIVs)

LCO 3.6.1.3 Each PCIV, except reactor building-to-suppression chamber vacuum breakers, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
When associated instrumentation is required to be OPERABLE
per LCO 3.3.6.1, "Primary Containment Isolation
Instrumentation."

ACTIONS

-----NOTES-----

1. Penetration flow paths may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by PCIVs.
4. Enter applicable Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," when PCIV leakage results in exceeding overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. -----NOTE----- Only applicable to penetration flow paths with two PCIVs. -----</p> <p>One or more penetration flow paths with one PCIV inoperable except due to leakage not within limit.</p>	<p>A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.</p> <p><u>AND</u></p>	<p>4 hours except for main steam line</p> <p><u>AND</u></p> <p>8 hours for main steam line</p> <p>(continued)</p>



ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	<p>A.2</p> <p>-----NOTE----- Isolation devices in high radiation areas may be verified by use of administrative means. -----</p> <p>Verify the affected penetration flow path is isolated.</p>	<p>Once per 31 days for isolation devices outside primary containment</p> <p><u>AND</u></p> <p>Prior to entering MODE 2 or 3 from MODE 4 if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days, for isolation devices inside primary containment</p>

(continued)



ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. -----NOTE----- Only applicable to penetration flow paths with two PCIVs. -----</p> <p>One or more penetration flow paths with two PCIVs inoperable except due to leakage not within limit.</p>	<p>B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.</p>	<p>1 hour</p>
<p>C. -----NOTE----- Only applicable to penetration flow paths with only one PCIV. -----</p> <p>One or more penetration flow paths with one PCIV inoperable except due to leakage not within limit.</p>	<p>C.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.</p> <p><u>AND</u></p>	<p>4 hours except for excess flow check valves (EFCVs)</p> <p><u>AND</u> 12 hours for EFCVs</p> <p>(continued)</p>

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MEMORANDUM FOR THE RECORD

SUBJECT: [Illegible]

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18. [Illegible]

19. [Illegible]

20. [Illegible]

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	<p>C.2</p> <p>-----NOTE----- Isolation devices in high radiation areas may be verified by use of administrative means. -----</p> <p>Verify the affected penetration flow path is isolated.</p>	<p>Once per 31 days for isolation devices outside primary containment</p> <p><u>AND</u></p> <p>Prior to entering MODE 2 or 3 from MODE 4 if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days, for isolation devices inside primary containment</p>

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more penetration flow paths with secondary containment bypass leakage rate, MSIV leakage rate, or hydrostatically tested lines leakage rate not within limit.	D.1 Restore leakage rate to within limit.	4 hours except for main steam line <u>AND</u> 8 hours for main steam line
E. Required Action and associated Completion Time of Condition A, B, C, or D not met in MODE 1, 2, or 3.	E.1 Be in MODE 3. <u>AND</u> E.2 Be in MODE 4.	12 hours 36 hours
F. Required Action and associated Completion Time of Condition A, B, C, or D not met for PCIV(s) required to be OPERABLE during MODE 4 or 5.	F.1 Initiate action to suspend operations with a potential for draining the reactor vessel (OPDRVs). <u>OR</u> F.2 Initiate action to restore valve(s) to OPERABLE status.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.6.1.3.1 -----NOTE-----</p> <p>Not required to be met when the 24 inch and 30 inch primary containment purge valves are open for inerting, de-inerting, pressure control, ALARA or air quality considerations for personnel entry, or Surveillances that require the valves to be open.</p> <p>-----</p> <p>Verify each 24 inch and 30 inch primary containment purge valve is closed.</p>	<p>31 days</p>
<p>SR 3.6.1.3.2 -----NOTES-----</p> <ol style="list-style-type: none"> 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for PCIVs that are open under administrative controls. <p>-----</p> <p>Verify each primary containment isolation manual valve and blind flange that is located outside primary containment and is required to be closed during accident conditions is closed.</p>	<p>31 days</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.3	<p>-----NOTES-----</p> <ol style="list-style-type: none"> 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for PCIVs that are open under administrative controls. <p>-----</p> <p>Verify each primary containment isolation manual valve and blind flange that is located inside primary containment and is required to be closed during accident conditions is closed.</p>	<p>Prior to entering MODE 2 or 3 from MODE 4 if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days</p>
SR 3.6.1.3.4	Verify continuity of the traversing incore probe (TIP) shear isolation valve explosive charge.	31 days
SR 3.6.1.3.5	Verify the isolation time of each power operated and each automatic PCIV, except MSIVs, is within limits.	In accordance with the Inservice Testing Program
SR 3.6.1.3.6	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the Inservice Testing Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.8	Verify each EFCV actuates to the isolation position on an actual or simulated instrument line break signal.	24 months
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP System.	24 months on a STAGGERED TEST BASIS
SR 3.6.1.3.10	Verify the combined leakage rate for all secondary containment bypass leakage paths is ≤ 0.74 scfh when pressurized to $\geq P_a$.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.11	Verify leakage rate through each MSIV is ≤ 11.5 scfh when tested at ≥ 25.0 psig.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.12	Verify combined leakage rate through hydrostatically tested lines that penetrate the primary containment is within limits.	In accordance with the Primary Containment Leakage Rate Testing Program

3.6 CONTAINMENT SYSTEMS

3.6.1.4 Drywell Air Temperature

LCO 3.6.1.4 Drywell average air temperature shall be $\leq 135^{\circ}\text{F}$.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Drywell average air temperature not within limit.	A.1 Restore drywell average air temperature to within limit.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.1.4.1 Verify drywell average air temperature is within limit.	24 hours

3.6 CONTAINMENT SYSTEMS

3.6.1.5 Residual Heat Removal (RHR) Drywell Spray

LCO 3.6.1.5 Two RHR drywell spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR drywell spray subsystem inoperable.	A.1 Restore RHR drywell spray subsystem to OPERABLE status.	7 days
B. Two RHR drywell spray subsystems inoperable.	B.1 Restore one RHR drywell spray subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours
	<u>AND</u> C.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.5.1	Verify each RHR drywell spray subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position.	31 days
SR 3.6.1.5.2	Verify each spray nozzle is unobstructed.	10 years

3.6 CONTAINMENT SYSTEMS

3.6.1.6 Reactor Building-to-Suppression Chamber Vacuum Breakers

LCO 3.6.1.6 Each reactor building-to-suppression chamber vacuum breaker shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more lines with one reactor building-to-suppression chamber vacuum breaker not closed.	A.1 Close the open vacuum breaker.	72 hours
B. One or more lines with two reactor building-to-suppression chamber vacuum breakers not closed.	B.1 Close one open vacuum breaker.	1 hour
C. One line with one or more reactor building-to-suppression chamber vacuum breakers inoperable for opening.	C.1 Restore the vacuum breaker(s) to OPERABLE status.	72 hours

(continued)

Reactor Building-to-Suppression Chamber Vacuum Breakers
3.6.1.6

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two or more lines with one or more reactor building-to-suppression chamber vacuum breakers inoperable for opening.	D.1 Restore all vacuum breakers in two lines to OPERABLE status.	1 hour
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3. <u>AND</u>	12 hours
	E.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.1.6.1 -----NOTES----- 1. Not required to be met for vacuum breakers that are open during Surveillances. 2. Not required to be met for vacuum breakers open when performing their intended function. ----- Verify each vacuum breaker is closed.	14 days

(continued)

Reactor Building-to-Suppression Chamber Vacuum Breakers
3.6.1.6

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.6.2	Perform a functional test of each vacuum breaker.	In accordance with the Inservice Testing Program
SR 3.6.1.6.3	Verify the full open setpoint of each vacuum breaker is ≤ 0.5 psid.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.1.7 Suppression Chamber-to-Drywell Vacuum Breakers

LC0 3.6.1.7 Seven suppression chamber-to-drywell vacuum breakers shall be OPERABLE for opening.

AND

Nine suppression chamber-to-drywell vacuum breakers shall be closed, except when performing their intended function.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required suppression chamber-to-drywell vacuum breaker inoperable for opening.	A.1 Restore one vacuum breaker to OPERABLE status.	72 hours
B. -----NOTE----- Separate Condition entry is allowed for each suppression chamber-to-drywell vacuum breaker. ----- One or more suppression chamber-to-drywell vacuum breakers with one disk not closed.	B.1 Close the open vacuum breaker disk.	72 hours

(continued)

1. The first part of the document is a list of names and addresses of the members of the committee.

2. The second part of the document is a list of names and addresses of the members of the committee.

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Suppression Chamber-to-Drywell Vacuum Breakers
3.6.1.7

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more suppression chamber-to-drywell vacuum breakers with two disks not closed.	C.1 Close one open vacuum breaker disk.	2 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. <u>AND</u>	12 hours
	D.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.1.7.1 -----NOTE----- Not required to be met for vacuum breakers that are open during Surveillances. ----- Verify each vacuum breaker is closed.	14 days

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.6.1.7.2 Perform a functional test of each required vacuum breaker.	31 days <u>AND</u> Within 12 hours after any discharge of steam to the suppression chamber from the safety/relief valves
SR 3.6.1.7.3 Verify the full open setpoint of each required vacuum breaker is ≤ 0.5 psid.	24 months



3.6 CONTAINMENT SYSTEMS

3.6.1.8 Main Steam Isolation Valve Leakage Control (MSLC) System

LCO 3.6.1.8 Two MSLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MSLC subsystem inoperable.	A.1 Restore MSLC subsystem to OPERABLE status.	30 days
B. Two MSLC subsystems inoperable.	B.1 Restore one MSLC subsystem to OPERABLE status.	7 days
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours
	<u>AND</u> C.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.1.8.1 Operate each MSLC blower \geq 15 minutes.	31 days

(continued)

D SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.8.2	Verify electrical continuity of each inboard MSLC subsystem heater element circuitry.	31 days
SR 3.6.1.8.3	Perform a system functional test of each MSLC subsystem.	18 months

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3.6 CONTAINMENT SYSTEMS

3.6.2.1 Suppression Pool Average Temperature

LCO 3.6.2.1 Suppression pool average temperature shall be:

- a. $\leq 90^{\circ}\text{F}$ when THERMAL POWER is $> 1\%$ RTP and no testing that adds heat to the suppression pool is being performed;
- b. $\leq 105^{\circ}\text{F}$ when THERMAL POWER is $> 1\%$ RTP and testing that adds heat to the suppression pool is being performed; and
- c. $\leq 110^{\circ}\text{F}$ when THERMAL POWER is $\leq 1\%$ RTP.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Suppression pool average temperature $> 90^{\circ}\text{F}$ but $\leq 110^{\circ}\text{F}$. <u>AND</u> THERMAL POWER $> 1\%$ RTP. <u>AND</u> Not performing testing that adds heat to the suppression pool.	A.1 Verify suppression pool average temperature $\leq 110^{\circ}\text{F}$.	Once per hour
	<u>AND</u> A.2 Restore suppression pool average temperature to $\leq 90^{\circ}\text{F}$.	24 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to $\leq 1\%$ RTP.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. Suppression pool average temperature > 105°F.</p> <p><u>AND</u></p> <p>THERMAL POWER > 1% RTP.</p> <p><u>AND</u></p> <p>Performing testing that adds heat to the suppression pool.</p>	<p>C.1 Suspend all testing that adds heat to the suppression pool.</p>	<p>Immediately</p>
<p>D. Suppression pool average temperature > 110°F but ≤ 120°F.</p>	<p>D.1 Place the reactor mode switch in the shutdown position.</p> <p><u>AND</u></p> <p>D.2 Verify suppression pool average temperature ≤ 120°F.</p> <p><u>AND</u></p> <p>D.3 Be in MODE 4.</p>	<p>Immediately</p> <p>Once per 30 minutes</p> <p>36 hours</p>
<p>E. Suppression pool average temperature > 120°F.</p>	<p>E.1 Depressurize the reactor vessel to < 200 psig.</p> <p><u>AND</u></p> <p>E.2 Be in MODE 4.</p>	<p>12 hours</p> <p>36 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.2.1.1 Verify suppression pool average temperature is within the applicable limits.	24 hours <u>AND</u> 5 minutes when performing testing that adds heat to the suppression pool



3.6 CONTAINMENT SYSTEMS

3.6.2.2 Suppression Pool Water Level

LCO 3.6.2.2 Suppression pool water level shall be \geq 30 ft 9.75 inches and \leq 31 ft 1.75 inches.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Suppression pool water level not within limits.	A.1 Restore suppression pool water level to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.2.2.1 Verify suppression pool water level is within limits.	24 hours



3.6 CONTAINMENT SYSTEMS

3.6.2.3 Residual Heat Removal (RHR) Suppression Pool Cooling

LCO 3.6.2.3 Two RHR suppression pool cooling subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR suppression pool cooling subsystem inoperable.	A.1 Restore RHR suppression pool cooling subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Two RHR suppression pool cooling subsystems inoperable.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.3.1	Verify each RHR suppression pool cooling subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position.	31 days
SR 3.6.2.3.2	Verify each RHR pump develops a flow rate ≥ 7100 gpm through the associated heat exchanger while operating in the suppression pool cooling mode.	In accordance with the Inservice Testing Program

3.6 CONTAINMENT SYSTEMS

3.6.3.1 Primary Containment Hydrogen Recombiners

LCO 3.6.3.1 Two primary containment hydrogen recombiners shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One primary containment hydrogen recombinder inoperable.	<p>A.1 -----NOTE----- LCO 3.0.4 is not applicable. -----</p> <p>Restore primary containment hydrogen recombinder to OPERABLE status.</p>	30 days
B. Two primary containment hydrogen recombinders inoperable.	<p>B.1 Verify by administrative means that the hydrogen and oxygen control function is maintained.</p> <p><u>AND</u></p> <p>B.2 Restore one primary containment hydrogen recombinder to OPERABLE status.</p>	<p>1 hour</p> <p><u>AND</u></p> <p>Once per 12 hours thereafter</p> <p>7 days</p>
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1.1	Perform a system functional test for each primary containment hydrogen recombiner.	24 months
SR 3.6.3.1.2	Visually examine each primary containment hydrogen recombiner enclosure and verify there is no evidence of abnormal conditions.	24 months
SR 3.6.3.1.3	Perform a resistance to ground test for each heater phase.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.3.2 Primary Containment Atmosphere Mixing System

LCO 3.6.3.2 Two head area return fans shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One head area return fan inoperable.	-----NOTE----- LCO 3.0.4 is not applicable. -----	
	A.1 Restore head area return fan to OPERABLE status.	30 days
B. Two head area return fans inoperable.	B.1 Verify by administrative means that the hydrogen and oxygen control function is maintained.	1 hour
	<u>AND</u> B.2 Restore one head area return fan to OPERABLE status.	7 days
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.3.2.1 Operate each head area return fan for \geq 15 minutes.	92 days

3.6 CONTAINMENT SYSTEMS

3.6.3.3 Primary Containment Oxygen Concentration

LCO 3.6.3.3 The primary containment oxygen concentration shall be
< 3.5 volume percent.

APPLICABILITY: MODE 1 during the time period:

- a. From 24 hours after THERMAL POWER is > 15% RTP following startup, to
- b. 24 hours prior to reducing THERMAL POWER to < 15% RTP prior to the next scheduled reactor shutdown.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Primary containment oxygen concentration not within limit.	A.1 Restore oxygen concentration to within limit.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to \leq 15% RTP.	8 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.3.3.1 Verify primary containment oxygen concentration is within limits.	7 days

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the
secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor
vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1 Restore secondary containment to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Secondary containment inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	C.1 -----NOTE----- LCO 3.0.3 is not applicable. -----	
	Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
	C.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	C.3 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.4.1.1 Verify secondary containment vacuum is ≥ 0.25 inch of vacuum water gauge.	24 hours
SR 3.6.4.1.2 Verify all secondary containment equipment hatches are closed and sealed.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.3	Verify each secondary containment access inner door or each secondary containment access outer door in each access opening is closed.	31 days
SR 3.6.4.1.4	Verify each standby gas treatment (SGT) subsystem will draw down the secondary containment to ≥ 0.25 inch of vacuum water gauge in ≤ 120 seconds.	24 months on a STAGGERED TEST BASIS
SR 3.6.4.1.5	Verify each SGT subsystem can maintain ≥ 0.25 inch of vacuum water gauge in the secondary containment for 1 hour at a flow rate ≤ 2240 cfm.	24 months on a STAGGERED TEST BASIS

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

LCO 3.6.4.2 Each SCIV shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the
secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor
vessel (OPDRVs).

ACTIONS

-----NOTES-----

1. Penetration flow paths may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by SCIVs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more penetration flow paths with one SCIV inoperable.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	8 hours
	<u>AND</u>	(continued)



ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	<p>A.2 -----NOTE----- Isolation devices in high radiation areas may be verified by use of administrative means. -----</p> <p>Verify the affected penetration flow path is isolated.</p>	Once per 31 days
<p>B. -----NOTE----- Only applicable to penetration flow paths with two isolation valves. -----</p> <p>One or more penetration flow paths with two SCIVs inoperable.</p>	B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	4 hours
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	<p>C.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>C.2 Be in MODE 4.</p>	<p>12 hours</p> <p>36 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	D.1 -----NOTE----- LCO 3.0.3 is not applicable. -----	
	Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
	D.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	D.3 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.6.4.2.1 -----NOTES-----</p> <ol style="list-style-type: none"> 1. Valves and blind flanges in high radiation areas may be verified by use of administrative controls. 2. Not required to be met for SCIVs that are open under administrative controls. <p>-----</p> <p>Verify each secondary containment isolation manual valve and blind flange that is required to be closed during accident conditions is closed.</p>	<p>31 days</p>
<p>SR 3.6.4.2.2 Verify the isolation time of each power operated and each automatic SCIV is within limits.</p>	<p>In accordance with the Inservice Testing Program</p>
<p>SR 3.6.4.2.3 Verify each automatic SCIV actuates to the isolation position on an actual or simulated automatic isolation signal.</p>	<p>24 months</p>

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the
secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor
vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	36 hours
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	-----NOTE----- LCO 3.0.3 is not applicable. -----	Immediately (continued)
	C.1 Place OPERABLE SGT subsystem in operation. <u>OR</u>	

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
	C.2.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	C.2.3 Initiate action to suspend OPDRVs.	Immediately
D. Two SGT subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately
E. Two SGT subsystems inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	E.1 -----NOTE----- LCO 3.0.3 is not applicable. -----	
	Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
	E.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	E.3 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.3.1	Operate each SGT subsystem for ≥ 10 continuous hours with heaters operating.	31 days
SR 3.6.4.3.2	Perform required SGT filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.4.3.3	Verify each SGT subsystem actuates on an actual or simulated initiation signal.	24 months
SR 3.6.4.3.4	Verify each SGT filter cooling recirculation valve can be opened and the fan started.	24 months

3.7 PLANT SYSTEMS

3.7.1 Standby Service Water (SW) System and Ultimate Heat Sink (UHS).

LCO 3.7.1 Division 1 and 2 SW subsystems and UHS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Average sediment depth in one or both spray ponds ≥ 0.5 ft and < 1.0 ft.	A.1 Restore average sediment depth to within limits.	30 days

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One SW subsystem inoperable.	<p>B.1 -----NOTES-----</p> <ol style="list-style-type: none"> 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources—Operating," for diesel generator made inoperable by SW System. 2. Enter applicable Conditions and Required Actions of LCO 3.4.9, "Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown," for RHR shutdown cooling subsystem made inoperable by SW System. <p>-----</p> <p>Restore SW subsystem to OPERABLE status.</p>	72 hours

(continued)



ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met. <u>OR</u> Both SW subsystems inoperable. <u>OR</u> UHS inoperable for reasons other than Condition A.	C.1 Be in MODE 3.	12 hours
	<u>AND</u> C.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.1.1 Verify the water level of each UHS spray pond is \geq 432 ft 9 inches mean sea level.	24 hours
SR 3.7.1.2 Verify the average water temperature of each UHS spray pond is \leq 77°F.	24 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.7.1.3 -----NOTE----- Isolation of flow to individual components does not render SW subsystem inoperable. -----</p> <p>Verify each SW subsystem manual, power operated, and automatic valve in the flow path servicing safety related systems or components, that is not locked, sealed, or otherwise secured in position, is in the correct position.</p>	31 days
<p>SR 3.7.1.4 Verify average sediment depth in each UHS spray pond is < 0.5 ft.</p>	92 days
<p>SR 3.7.1.5 Verify each SW subsystem actuates on an actual or simulated initiation signal.</p>	24 months

3.7 PLANT SYSTEMS

3.7.2 High Pressure Core Spray (HPCS) Service Water (SW) System

LCO 3.7.2 The HPCS SW System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. HPCS SW System inoperable.	A.1 Declare HPCS System inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.2.1 -----NOTE----- Isolation of flow to individual components does not render HPCS SW System inoperable. ----- Verify each HPCS SW System manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.2.2 Verify the HPCS SW System actuates on an actual or simulated initiation signal.	24 months

3.7 PLANT SYSTEMS

3.7.3 Control Room Emergency Filtration (CREF) System

LCO 3.7.3 Two CREF subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the
secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor
vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREF subsystem inoperable.	A.1 Restore CREF subsystem to OPERABLE status.	7 days
B. Required Action and Associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	-----NOTE----- LCO 3.0.3 is not applicable. -----	
	C.1 Place OPERABLE CREF subsystem in pressurization mode.	Immediately
	<u>OR</u>	
	C.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
	C.2.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	C.2.3 Initiate action to suspend OPDRVs.	Immediately
D. Two CREF subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately

(continued)

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two CREF subsystems inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	-----NOTE----- LCO 3.0.3 is not applicable. -----	
	E.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
	E.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	E.3 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.3.1 Operate each CREF subsystem for ≥ 10 continuous hours with the heaters operating.	31 days
SR 3.7.3.2 Perform required CREF filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.3.3	Verify each CREF subsystem actuates on an actual or simulated initiation signal.	24 months
SR 3.7.3.4	Verify each CREF subsystem can maintain a positive pressure of $\geq 1/8$ inches water gauge relative to the radwaste and turbine buildings during the pressurization mode of operation at an outside air flow rate of ≤ 1000 cfm.	24 months on a STAGGERED TEST BASIS



3.7 PLANT SYSTEMS

3.7.4 Control Room Air Conditioning (AC) System

LCO 3.7.4 Two control room AC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the
secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor
vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One control room AC subsystem inoperable.	A.1 Restore control room AC subsystem to OPERABLE status.	30 days
B. Required Action and Associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	-----NOTE----- LCO 3.0.3 is not applicable. -----	
	C.1 Place OPERABLE control room AC subsystem in operation.	Immediately
	<u>OR</u>	
	C.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
	C.2.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	C.2.3 Initiate action to suspend OPDRVs.	Immediately
D. Two control room AC subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two control room AC subsystems inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	-----NOTE----- LCO 3.0.3 is not applicable. -----	
	E.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
	E.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	E.3 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.4.1 Verify each control room AC subsystem has the capability to remove the assumed heat load.	24 months

3.7 PLANT SYSTEMS

3.7.5 Main Condenser Offgas

LCO 3.7.5 The gross gamma activity rate of the noble gases measured at the main condenser air ejector shall be ≤ 332 mCi/second after decay of 30 minutes.

APPLICABILITY: MODE 1,
MODES 2 and 3 with any main steam line not isolated and
steam jet air ejector (SJAE) in operation.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Gross gamma activity rate of the noble gases not within limit.	A.1 Restore gross gamma activity rate of the noble gases to within limit.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Isolate all main steam lines.	12 hours
	<u>OR</u>	
	B.2 Isolate SJAE.	12 hours
	<u>OR</u>	
	B.3.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	B.3.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.7.5.1 -----NOTE----- Not required to be performed until 31 days after any main steam line not isolated and SJAE in operation. ----- Verify the gross gamma activity rate of the noble gases is ≤ 332 mCi/second after decay of 30 minutes.</p>	<p>31 days <u>AND</u> Once within 4 hours after a $\geq 50\%$ increase in the nominal steady state fission gas release after factoring out increases due to changes in THERMAL POWER level</p>



3.7 PLANT SYSTEMS

3.7.6 Main Turbine Bypass System

LCO 3.7.6 The Main Turbine Bypass System shall be OPERABLE.

OR

LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are made applicable.

APPLICABILITY: THERMAL POWER \geq 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.6.1 Verify one complete cycle of each main turbine bypass valve.	31 days

(continued)



SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.6.2	Perform a system functional test.	24 months
SR 3.7.6.3	Verify the TURBINE BYPASS SYSTEM RESPONSE TIME is within limits.	24 months

3.7 PLANT SYSTEMS

3.7.7 Spent Fuel Storage Pool Water Level

LCO 3.7.7 The spent fuel storage pool water level shall be ≥ 22 ft over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel storage pool water level not within limit.	<p>A.1 -----NOTE----- LCO 3.0.3 is not applicable. -----</p> <p>Suspend movement of irradiated fuel assemblies in the spent fuel storage pool.</p>	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.7.1 Verify the spent fuel storage pool water level is ≥ 22 ft over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.	7 days



3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources—Operating

LCO 3.8.1 The following AC electrical power sources shall be OPERABLE:

- a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electric Power Distribution System; and
- b. Three diesel generators (DGs).

APPLICABILITY: MODES 1, 2, and 3.

-----NOTE-----
Division 3 AC electrical power sources are not required to be OPERABLE when High Pressure Core Spray System is inoperable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 for OPERABLE offsite circuit. <u>AND</u>	1 hour <u>AND</u> Once per 8 hours thereafter (continued)

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ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Declare required feature(s) with no offsite power available inoperable when the redundant required feature(s) are inoperable.	24 hours from discovery of no offsite power to one division concurrent with inoperability of redundant required feature(s)
	<u>AND</u> A.3 Restore offsite circuit to OPERABLE status.	72 hours <u>AND</u> 6 days from discovery of failure to meet LCO
B. One required DG inoperable.	B.1 Perform SR 3.8.1.1 for OPERABLE offsite circuit(s). <u>AND</u>	1 hour <u>AND</u> Once per 8 hours thereafter (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Declare required feature(s), supported by the inoperable DG, inoperable when the redundant required feature(s) are inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	<u>AND</u>	
	B.3.1 Determine OPERABLE DG(s) are not inoperable due to common cause failure.	24 hours
	<u>OR</u>	
	B.3.2 Perform SR 3.8.1.2 for OPERABLE DG(s).	24 hours
	<u>AND</u>	
	B.4 Restore required DG to OPERABLE status.	72 hours
		<u>AND</u>
		6 days from discovery of failure to meet LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two offsite circuits inoperable.	C.1 Declare required feature(s) inoperable when the redundant required feature(s) are inoperable.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s)
	<u>AND</u> C.2 Restore one offsite circuit to OPERABLE status.	24 hours
D. One offsite circuit inoperable. <u>AND</u> One required DG inoperable.	<p>-----NOTE----- Enter applicable Conditions and Required Actions of LCO 3.8.7, "Distribution Systems—Operating," when Condition D is entered with no AC power source to any division. -----</p> <p>D.1 Restore offsite circuit to OPERABLE status.</p> <p><u>OR</u></p> <p>D.2 Restore required DG to OPERABLE status.</p>	<p>12 hours</p> <p>12 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two required DGs inoperable.	E.1 Restore one required DG to OPERABLE status.	2 hours <u>OR</u> 24 hours if Division 3 DG is inoperable
F. Required Action and Associated Completion Time of Condition A, B, C, D, or E not met.	F.1 Be in MODE 3. <u>AND</u> F.2 Be in MODE 4.	12 hours 36 hours
G. Three or more required AC sources inoperable.	G.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.1.1 Verify correct breaker alignment and indicated power availability for each offsite circuit.	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.2 -----NOTES-----</p> <ol style="list-style-type: none"> 1. All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. 2. A modified DG start involving idling and gradual acceleration to synchronous speed may be used for this SR as recommended by the manufacturer. When modified start procedures are not used, the time, voltage, and frequency tolerances of SR 3.8.1.7 must be met. <p>-----</p> <p>Verify each required DG starts from standby conditions and achieves steady state:</p> <ol style="list-style-type: none"> a. Voltage ≥ 3910 V and ≤ 4400 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz for DG-1 and DG-2; and b. Voltage ≥ 3740 V and ≤ 4400 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz for DG-3. 	<p>31 days</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.3	<p>-----NOTES-----</p> <ol style="list-style-type: none"> 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by, and immediately follow, without shutdown, a successful performance of SR 3.8.1.2 or SR 3.8.1.7. <p>-----</p> <p>Verify each required DG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 4000 kW and ≤ 4400 kW for DG-1 and DG-2, and ≥ 2340 kW and ≤ 2600 kW for DG-3.</p>	31 days
SR 3.8.1.4	Verify each required day tank contains ≥ 1400 gal of fuel oil.	31 days
SR 3.8.1.5	Check for and remove accumulated water from each required day tank.	31 days
SR 3.8.1.6	Verify each required fuel oil transfer subsystem operates to automatically transfer fuel oil from the storage tank to the day tank.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.7 -----NOTE----- All DG starts may be preceded by an engine prelube period. ----- Verify each required DG starts from standby condition and achieves:</p> <p>a. For DG-1 and DG-2 in ≤ 15 seconds, voltage ≥ 3910 V and frequency ≥ 58.8 Hz, and after steady state conditions are reached, maintains voltage ≥ 3910 V and ≤ 4400 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz; and</p> <p>b. For DG-3, in ≤ 15 seconds, voltage ≥ 3740 V and frequency ≥ 58.8 Hz, and after steady state conditions are reached, maintains voltage ≥ 3740 V and ≤ 4400 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.</p>	<p>184 days</p>
<p>SR 3.8.1.8 -----NOTE----- This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. ----- Verify automatic and manual transfer of the power supply to safety related buses from the startup offsite circuit to the backup offsite circuit.</p>	<p>24 months</p>

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.9 -----NOTES-----</p> <ol style="list-style-type: none"> 1. This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. 2. If performed with the DG synchronized with offsite power, it shall be performed at a power factor as close to the power factor of the single largest post-accident load as practicable. <p>-----</p> <p>Verify each required DG rejects a load greater than or equal to its associated single largest post-accident load, and following load rejection, the frequency is ≤ 66.75 Hz.</p>	<p>24 months</p>
<p>SR 3.8.1.10 -----NOTES-----</p> <ol style="list-style-type: none"> 1. This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. 2. If performed with the DG synchronized with offsite power, it shall be performed at the accident load power factor, or at a power factor as close to the accident load power factor as practicable with the field excitation current $\geq 90\%$ of the continuous rating. <p>-----</p> <p>Verify each required DG does not trip and voltage is maintained ≤ 4784 V during and following a load rejection of a load ≥ 4400 kW for DG-1 and DG-2 and ≥ 2600 kW for DG-3.</p>	<p>24 months</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.11 -----NOTES-----</p> <ol style="list-style-type: none"> 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, 2, or 3. However, credit may be taken for unplanned events that satisfy this SR. <p>-----</p> <p>Verify on an actual or simulated loss of offsite power signal:</p> <ol style="list-style-type: none"> a. De-energization of emergency buses; b. Load shedding from emergency buses for Divisions 1 and 2; and c. DG auto-starts from standby condition and: <ol style="list-style-type: none"> 1. energizes permanently connected loads in ≤ 15 seconds for DG-1 and DG-2, and in ≤ 18 seconds for DG-3, 2. energizes auto-connected shutdown loads, 3. maintains steady state voltage ≥ 3740 V and ≤ 4400 V, 4. maintains steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz, and 5. supplies permanently connected and auto-connected shutdown loads for ≥ 5 minutes. 	<p>24 months</p>

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SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | FREQUENCY |
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| <p>SR 3.8.1.12 -----NOTES-----</p> <ol style="list-style-type: none"> 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. <p>-----</p> <p>Verify on an actual or simulated Emergency Core Cooling System (ECCS) initiation signal each required DG auto-starts from standby condition and:</p> <ol style="list-style-type: none"> a. For DG-1 and DG-2, in ≤ 15 seconds achieves voltage ≥ 3910 V, and after steady state conditions are reached, maintains voltage ≥ 3910 V and ≤ 4400 V and, for DG-3, in ≤ 15 seconds achieves voltage ≥ 3740 V, and after steady state conditions are reached, maintains voltage ≥ 3740 V and ≤ 4400 V; b. In ≤ 15 seconds, achieves frequency ≥ 58.8 Hz and after steady state conditions are achieved, maintains frequency ≥ 58.8 Hz and ≤ 61.2 Hz; c. Operates for ≥ 5 minutes; d. Permanently connected loads remain energized from the offsite power system; and e. Emergency loads are auto-connected to the offsite power system. | <p>24 months</p> |

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SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | FREQUENCY |
|---|------------------|
| <p>SR 3.8.1.13 -----NOTE-----
 This Surveillance shall not be performed in
 MODE 1, 2, or 3. However, credit may be
 taken for unplanned events that satisfy
 this SR.

 Verify each required DG's automatic trips
 are bypassed on an actual or simulated ECCS
 initiation signal except:</p> <ul style="list-style-type: none"> a. Engine overspeed; b. Generator differential current; and c. Incomplete starting sequence. | <p>24 months</p> |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | FREQUENCY |
|---|------------------|
| <p>SR 3.8.1.14 -----NOTES-----</p> <ol style="list-style-type: none"> 1. Momentary transients outside the load, excitation current, and power factor ranges do not invalidate this test. 2. This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. 3. If performed with the DG synchronized with offsite power, it shall be performed at the accident load power factor, or at a power factor as close to the accident load power factor as practicable with the field excitation current $\geq 90\%$ of the continuous rating. <p>-----</p> <p>Verify each required DG operates for ≥ 24 hours:</p> <ol style="list-style-type: none"> a. For ≥ 2 hours loaded ≥ 4650 kW for DG-1 and DG-2, and ≥ 2850 kW for DG-3; and b. For the remaining hours of the test loaded ≥ 4400 kW for DG-1 and DG-2, and ≥ 2600 kW for DG-3. | <p>24 months</p> |

(continued)



SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | FREQUENCY |
|--|------------------|
| <p>SR 3.8.1.15 -----NOTES-----</p> <ol style="list-style-type: none"> 1. This Surveillance shall be performed within 5 minutes of shutting down the DG after the DG has operated ≥ 1 hour loaded ≥ 4000 kW for DG-1 and DG-2, and ≥ 2340 kW for DG-3. <p> Momentary transients outside of load range do not invalidate this test.</p> <ol style="list-style-type: none"> 2. All DG starts may be preceded by an engine prelube period. <p>-----</p> <p>Verify each required DG starts and achieves:</p> <ol style="list-style-type: none"> a. For DG-1 and DG-2, in ≤ 15 seconds, voltage ≥ 3910 V and frequency ≥ 58.8 Hz, and after steady state conditions are reached, maintains voltage ≥ 3910 V and ≤ 4400 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz; and b. For DG-3, in ≤ 15 seconds, voltage ≥ 3740 V and frequency ≥ 58.8 Hz, and after steady state conditions are reached, maintains voltage ≥ 3740 V and ≤ 4400 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz. | <p>24 months</p> |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | FREQUENCY |
|---|------------------|
| <p>SR 3.8.1.16 -----NOTE-----
 This Surveillance shall not be performed in
 MODE 1, 2, or 3. However, credit may be
 taken for unplanned events that satisfy
 this SR.
 -----</p> <p>Verify each required DG:</p> <ul style="list-style-type: none"> a. Synchronizes with offsite power source
while loaded with emergency loads upon
a simulated restoration of offsite
power; b. Transfers loads to offsite power
source; and c. Returns to ready-to-load operation. | <p>24 months</p> |
| <p>SR 3.8.1.17 -----NOTE-----
 This Surveillance shall not be performed in
 MODE 1, 2, or 3. However, credit may be
 taken for unplanned events that satisfy
 this SR.
 -----</p> <p>Verify, with a DG operating in test mode
and connected to its bus, an actual or
simulated ECCS initiation signal overrides
the test mode by:</p> <ul style="list-style-type: none"> a. Returning DG to ready-to-load
operation; and b. Automatically energizing the emergency
load from offsite power. | <p>24 months</p> |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | FREQUENCY |
|---|-------------------------------------|
| <p>SR 3.8.1.18 -----NOTE-----
 This Surveillance shall not be performed in
 MODE 1, 2, or 3. However, credit may be
 taken for unplanned events that satisfy
 this SR.

 Verify interval between each sequenced load
 block is within $\pm 10\%$ of design interval
 for each time delay relay.</p> | <p>24 months</p> |
| <p>SR 3.8.1.19 -----NOTES-----
 1. All DG starts may be preceded by an
 engine prelube period.
 2. This Surveillance shall not be
 performed in MODE 1, 2, or 3.
 However, credit may be taken for
 unplanned events that satisfy this SR.

 Verify, on an actual or simulated loss of
 offsite power signal in conjunction with an
 actual or simulated ECCS initiation signal:
 a. De-energization of emergency buses;
 b. Load shedding from emergency buses for
 DG-1 and DG-2; and
 c. DG auto-starts from standby condition
 and:
 1. energizes permanently connected
 loads in ≤ 15 seconds,
 2. energizes auto-connected emergency
 loads,
 3. maintains steady state voltage
 ≥ 3740 V and ≤ 4400 V,</p> | <p>24 months</p> <p>(continued)</p> |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|--|-----------------|
| <p>SR 3.8.1.19 (continued)</p> <p>4. maintains steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz, and</p> <p>5. supplies permanently connected and auto-connected emergency loads for ≥ 5 minutes.</p> | |
| <p>SR 3.8.1.20 -----NOTE-----
All DG starts may be preceded by an engine prelube period.
-----</p> <p>Verify, when started simultaneously from standby condition, DG-1 and DG-2 achieves, in ≤ 15 seconds, voltage ≥ 3910 V and frequency ≥ 58.8 Hz, and DG-3 achieves, in ≤ 15 seconds, voltage ≥ 3740 V and frequency ≥ 58.8 Hz.</p> | <p>10 years</p> |

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources—Shutdown

LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

- a. One qualified circuit between the offsite transmission network and the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems—Shutdown";
- b. One diesel generator (DG) capable of supplying one division of the Division 1 or 2 onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.8; and
- c. The Division 3 DG capable of supplying the Division 3 onsite Class 1E AC electrical power distribution subsystem, when the Division 3 onsite Class 1E electrical power distribution subsystem is required by LCO 3.8.8.

APPLICABILITY: MODES 4 and 5,
During movement of irradiated fuel assemblies in the
secondary containment.

ACTIONS

-----NOTE-----
LCO 3.0.3 is not applicable.

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|-----------------|
| A. Required offsite circuit inoperable. | -----NOTE-----
Enter applicable Condition and Required Actions of LCO 3.8.8, when any required division is de-energized as a result of Condition A.
----- | |
| | A.1 Declare affected required feature(s) with no offsite power available inoperable. | Immediately |
| | <u>OR</u> | |
| | A.2.1 Suspend CORE ALTERATIONS. | Immediately |
| | <u>AND</u> | |
| | A.2.2 Suspend movement of irradiated fuel assemblies in the secondary containment. | Immediately |
| | <u>AND</u> | |
| | A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel (OPDRVs). | Immediately |
| | <u>AND</u> | |
| | | (continued) |

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|--|---|-----------------|
| A. (continued) | A.2.4 Initiate action to restore required offsite power circuit to OPERABLE status. | Immediately |
| B. Division 1 or 2 required DG inoperable. | B.1 Suspend CORE ALTERATIONS. | Immediately |
| | <u>AND</u> | |
| | B.2 Suspend movement of irradiated fuel assemblies in secondary containment. | Immediately |
| | <u>AND</u> | |
| | B.3 Initiate action to suspend OPDRVs. | Immediately |
| | <u>AND</u> | |
| | B.4 Initiate action to restore required DG to OPERABLE status. | Immediately |
| C. Required Division 3 DG inoperable. | C.1 Declare High Pressure Core Spray System inoperable. | 72 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|--|
| <p>SR 3.8.2.1 -----NOTE-----</p> <p>The following SRs are not required to be performed: SR 3.8.1.3, SR 3.8.1.9 through SR 3.8.1.11, SR 3.8.1.13 through SR 3.8.1.16, SR 3.8.1.18, and SR 3.8.1.19.</p> <p>-----</p> <p>For AC sources required to be OPERABLE, the SRs of Specification 3.8.1, except SR 3.8.1.8, SR 3.8.1.17, and SR 3.8.1.20, are applicable.</p> | <p>In accordance with applicable SRs</p> |

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air

LCO 3.8.3 The stored diesel fuel oil, lube oil, and starting air subsystem shall be within limits for each required diesel generator (DG).

APPLICABILITY: When associated DG is required to be OPERABLE.

ACTIONS

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

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|--|---|-----------------|
| <p>A. One or more DGs with stored fuel oil level:</p> <ol style="list-style-type: none"> 1. For DG-1 or DG-2,
< 55,500 gal and
≥ 47,520 gal; and 2. For DG-3,
< 33,000 gal and
≥ 28,340 gal. | <p>A.1 Restore stored fuel oil level to within limit.</p> | 48 hours |
| <p>B. One or more DGs with lube oil inventory:</p> <ol style="list-style-type: none"> 1. For DG-1 or DG-2,
< 330 gal and
≥ 283 gal; and 2. For DG-3,
< 165 gal and
≥ 142 gal. | <p>B.1 Restore lube oil inventory to within limit.</p> | 48 hours |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|--|--|-----------------|
| C. One or more DGs with stored fuel oil total particulates not within limit. | C.1 Restore stored fuel oil total particulates to within limit. | 7 days |
| D. One or more DGs with new fuel oil properties not within limits. | D.1 Restore stored fuel oil properties to within limits. | 30 days |
| E. One or more DGs with required starting air receiver pressure:

1. For DG-1 and DG-2,
< 230 psig and
≥ 150 psig; and

2. For DG-3,
< 223 psig and
≥ 150 psig. | E.1 Restore required starting air receiver pressure to within limit. | 48 hours |
| F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met.

<u>OR</u>

One or more DGs with stored diesel fuel oil, lube oil, or starting air subsystem not within limits for reasons other than Condition A, B, C, D, or E. | F.1 Declare associated DG inoperable. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|--------------|--|--|
| SR 3.8.3.1 | Verify each fuel oil storage tank contains:
a. $\geq 55,500$ gal of fuel for DG-1 and DG-2; and
b. $\geq 33,000$ gal of fuel for DG-3. | 31 days |
| SR 3.8.3.2 | Verify lube oil inventory is:
a. ≥ 330 gal for DG-1 and DG-2; and
b. ≥ 165 gal for DG-3. | 31 days |
| SR 3.8.3.3 | Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program. | In accordance with the Diesel Fuel Oil Testing Program |
| SR 3.8.3.4 | Verify each required DG air start receiver pressure is:
a. ≥ 230 psig for DG-1 and DG-2; and
b. ≥ 223 psig for DG-3. | 31 days |
| SR 3.8.3.5 | Check for and remove accumulated water from each fuel oil storage tank. | 92 days |

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources—Operating

LCO 3.8.4 The Division 1, Division 2, and Division 3 DC electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|--|---|-----------------|
| A. Division 1 or 2 125 V DC electrical power subsystem inoperable. | A.1 Restore Division 1 and 2 125 V DC electrical power subsystems to OPERABLE status. | 2 hours |
| B. Division 3 DC electrical power subsystem inoperable. | B.1 Declare High Pressure Core Spray System inoperable. | Immediately |
| C. Division 1 250 V DC electrical power subsystem inoperable. | C.1 Declare associated supported features inoperable. | Immediately |
| D. Required Action and associated Completion Time not met. | D.1 Be in MODE 3. | 12 hours |
| | <u>AND</u>
D.2 Be in MODE 4. | 36 hours |



SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|-----------|
| <p>SR 3.8.4.1 Verify battery terminal voltage on float charge is:</p> <p> a. ≥ 126 V for the 125 V batteries; and</p> <p> b. ≥ 252 V for the 250 V battery.</p> | 7 days |
| <p>SR 3.8.4.2 Verify no visible corrosion at battery terminals and connectors.</p> <p> <u>OR</u></p> <p> Verify battery connection resistance is ≤ 24.4 E-6 ohms for inter-cell connectors of the Division 1 and 2 batteries, ≤ 169 E-6 ohms for inter-cell connectors of the Division 3 battery, and $\leq 20\%$ above the resistance as measured during installation for inter-tier and inter-rack connectors.</p> | 92 days |
| <p>SR 3.8.4.3 Verify battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration that degrades battery performance.</p> | 12 months |
| <p>SR 3.8.4.4 Remove visible corrosion, and verify battery cell to cell and terminal connections are coated with anti-corrosion material.</p> | 12 months |

(continued)

1. $\frac{1}{2} \times \frac{1}{2} = \frac{1}{4}$ 2. $\frac{1}{2} \times \frac{1}{2} = \frac{1}{4}$ 3. $\frac{1}{2} \times \frac{1}{2} = \frac{1}{4}$ 4. $\frac{1}{2} \times \frac{1}{2} = \frac{1}{4}$ 5. $\frac{1}{2} \times \frac{1}{2} = \frac{1}{4}$ 6. $\frac{1}{2} \times \frac{1}{2} = \frac{1}{4}$ 7. $\frac{1}{2} \times \frac{1}{2} = \frac{1}{4}$ 8. $\frac{1}{2} \times \frac{1}{2} = \frac{1}{4}$ 9. $\frac{1}{2} \times \frac{1}{2} = \frac{1}{4}$ 10. $\frac{1}{2} \times \frac{1}{2} = \frac{1}{4}$

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | FREQUENCY |
|---|-----------|
| <p>SR 3.8.4.5 Verify battery connection resistance is ≤ 24.4 E-6 ohms for inter-cell connectors of the Division 1 and 2 batteries, ≤ 169 E-6 ohms for inter-cell connectors of the Division 3 battery, and $\leq 20\%$ above the resistance as measured during installation for inter-tier and inter-rack connectors.</p> | 12 months |
| <p>SR 3.8.4.6 Verify each required battery charger supplies the required load for ≥ 1.5 hours at:</p> <p>a. ≥ 126 V for the 125 V battery chargers; and</p> <p>b. ≥ 252 V for the 250 V battery charger.</p> | 24 months |
| <p>SR 3.8.4.7 -----NOTES-----</p> <p>1. The modified performance discharge test in SR 3.8.4.8 may be performed in lieu of the service test in SR 3.8.4.7 once per 60 months.</p> <p>2. This Surveillance shall not be performed in MODE 1, 2, or 3. However, credit may be taken for unplanned events that satisfy this SR.</p> <p>-----</p> <p>Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.</p> | 24 months |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | FREQUENCY |
|---|--|
| <p>SR 3.8.4.8 -----NOTE-----
 This Surveillance shall not be performed in
 MODE 1, 2, or 3. However, credit may be
 taken for unplanned events that satisfy
 this SR.

 Verify battery capacity is $\geq 80\%$ of the
 manufacturer's rating for the 125 V
 batteries and $\geq 83.4\%$ of the manufacturer's
 rating for the 250 V battery, when
 subjected to a performance discharge test
 or a modified performance discharge test.</p> | <p>60 months</p> <p><u>AND</u></p> <p>12 months when
 battery shows
 degradation or
 has reached 85%
 of expected
 life with
 capacity < 100%
 of
 manufacturer's
 rating</p> <p><u>AND</u></p> <p>24 months when
 battery has
 reached 85% of
 the expected
 life with
 capacity $\geq 100\%$
 of
 manufacturer's
 rating</p> |

100-100-100-100

100-100-100-100



100-100-100-100

100-100-100-100

100-100-100-100

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources—Shutdown

LCO 3.8.5 DC electrical power subsystem(s) shall be OPERABLE to support the electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems—Shutdown."

APPLICABILITY: MODES 4 and 5,
During movement of irradiated fuel assemblies in the
secondary containment.

ACTIONS

-----NOTE-----
LCO 3.0.3 is not applicable.

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|--|--|-----------------|
| A. One or more required DC electrical power subsystems inoperable. | A.1 Declare affected required feature(s) inoperable. | Immediately |
| | <u>OR</u> | |
| | A.2.1 Suspend CORE ALTERATIONS. | Immediately |
| | <u>AND</u> | |
| | A.2.2 Suspend movement of irradiated fuel assemblies in the secondary containment. | Immediately |
| | <u>AND</u> | |
| | | (continued) |

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|----------------|---|-----------------|
| A. (continued) | A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel. | Immediately |
| | <p><u>AND</u></p> <p>A.2.4 Initiate action to restore required DC electrical power subsystems to OPERABLE status.</p> | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|--|-----------------------------------|
| <p>SR 3.8.5.1 -----NOTE-----</p> <p>The following SRs are not required to be performed: SR 3.8.4.7 and SR 3.8.4.8.</p> <p>-----</p> <p>For DC electrical power subsystems required to be OPERABLE the following SRs are applicable:</p> <p>SR 3.8.4.1, SR 3.8.4.2, SR 3.8.4.3, SR 3.8.4.4, SR 3.8.4.5, SR 3.8.4.6, SR 3.8.4.7, and SR 3.8.4.8.</p> | In accordance with applicable SRs |

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Cell Parameters

LCO 3.8.6 Battery cell parameters for the Division 1, 2, and 3 batteries shall be within the limits of Table 3.8.6-1.

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

ACTIONS

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

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|--|--|----------------------------|
| A. One or more batteries with one or more battery cell parameters not within Category A or B limits. | A.1 Verify pilot cell(s) electrolyte level and float voltage meet Table 3.8.6-1 Category C limits. | 1 hour |
| | <u>AND</u> | |
| | A.2 Verify battery cell parameters meet Table 3.8.6-1 Category C limits. | 24 hours |
| | <u>AND</u> | Once per 7 days thereafter |
| | A.3 Restore battery cell parameters to Category A and B limits of Table 3.8.6-1. | 31 days |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|--|---|--------------------|
| <p>B. Required Action and associated Completion Time of Condition A not met.</p> <p><u>OR</u></p> <p>One or more batteries with average electrolyte temperature of the representative cells $\leq 60^{\circ}\text{F}$.</p> <p><u>OR</u></p> <p>One or more batteries with one or more battery cell parameters not within Category C limits.</p> | <p>B.1 Declare associated battery inoperable.</p> | <p>Immediately</p> |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|--|---------------|
| <p>SR 3.8.6.1 Verify battery cell parameters meet Table 3.8.6-1 Category A limits.</p> | <p>7 days</p> |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | FREQUENCY |
|--|---|
| <p>SR 3.8.6.2 Verify battery cell parameters meet Table 3.8.6-1 Category B limits.</p> | <p>92 days</p> <p><u>AND</u></p> <p>Once within 24 hours after battery discharge < 110 V for 125 V batteries and < 220 V for the 250 V battery</p> <p><u>AND</u></p> <p>Once within 24 hours after battery overcharge > 150 V for 125 V batteries and > 300 V for the 250 V battery</p> |
| <p>SR 3.8.6.3 Verify average electrolyte temperature of representative cells is > 60°F.</p> | <p>92 days</p> |

Table 3.8.6-1 (page 1 of 1)
Battery Cell Parameter Requirements

| PARAMETER | CATEGORY A:
LIMITS FOR EACH
DESIGNATED PILOT
CELL | CATEGORY B:
LIMITS FOR EACH
CONNECTED CELL | CATEGORY C:
LIMITS FOR EACH
CONNECTED CELL |
|---------------------------|--|--|--|
| Electrolyte Level | > Minimum level
indication mark,
and $\leq \frac{1}{4}$ inch
above maximum
level indication
mark(a) | > Minimum level
indication mark,
and $\leq \frac{1}{4}$ inch
above maximum
level indication
mark(a) | Above top of
plates, and not
overflowing |
| Float Voltage | ≥ 2.13 V | ≥ 2.13 V | > 2.07 V |
| Specific
Gravity(b)(c) | ≥ 1.200 | ≥ 1.195
<u>AND</u>
Average of all
connected cells
> 1.205 | Not more than
0.020 below
average of all
connected cells
<u>AND</u>
Average of all
connected cells
≥ 1.195 |

- (a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum level during and following equalizing charges provided it is not overflowing.
- (b) Corrected for electrolyte temperature and level. Level correction is not required, however, when battery charging is < 2 amps when on float charge.
- (c) A battery charging current of < 2 amps when on float charge is acceptable for meeting specific gravity limits following a battery recharge, for a maximum of 7 days. When charging current is used to satisfy specific gravity requirements, specific gravity of each connected cell shall be measured prior to expiration of the 7 day allowance.



3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Distribution Systems—Operating

LCO 3.8.7 The following AC and DC electrical power distribution subsystems shall be OPERABLE:

- a. Division 1 and Division 2 AC electrical power distribution subsystems;
- b. Division 1 and Division 2 125 V DC electrical power distribution subsystems;
- c. Division 1 250 V DC electrical power distribution subsystem; and
- d. Division 3 AC and DC electrical power distribution subsystems.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|--|--|
| A. Division 1 or 2 AC electrical power distribution subsystem inoperable. | A.1 Restore Division 1 and 2 AC electrical power distribution subsystems to OPERABLE status. | 8 hours
<u>AND</u>
16 hours from discovery of failure to meet LCO 3.8.7.a or b |

(continued)

1. The first part of the document is a list of names and addresses of the members of the committee.

2. The second part of the document is a list of the names and addresses of the members of the committee.

3.

4.

5.

6.



7. The last part of the document is a list of the names and addresses of the members of the committee.

8.

9.

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|--|--|--|
| B. Division 1 or 2 125 V DC electrical power distribution subsystem inoperable. | B.1 Restore Division 1 and 2 125 V DC electrical power distribution subsystems to OPERABLE status. | 2 hours

<u>AND</u>
16 hours from discovery of failure to meet LCO 3.8.7.a or b |
| C. Required Action and associated Completion Time of Condition A or B not met. | C.1 Be in MODE 3.

<u>AND</u>
C.2 Be in MODE 4. | 12 hours

36 hours |
| D. Division 1 250 V DC electrical power distribution subsystem inoperable. | D.1 Declare associated supported feature(s) inoperable. | Immediately |
| E. One or more Division 3 AC or DC electrical power distribution subsystems inoperable. | E.1 Declare High Pressure Core Spray System inoperable. | Immediately |
| F. Two or more divisions with inoperable electrical power distribution subsystems that result in a loss of function. | F.1 Enter LCO 3.0.3. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|--------------|--|-----------|
| SR 3.8.7.1 | Verify correct breaker alignments and indicated power availability to required AC and DC electrical power distribution subsystems. | 7 days |

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Distribution Systems—Shutdown

LCO 3.8.8 The necessary portions of the Division 1, Division 2, and Division 3 AC and DC electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

APPLICABILITY: MODES 4 and 5,
 During movement of irradiated fuel assemblies in the
 secondary containment.

ACTIONS

-----NOTE-----
LCO 3.0.3 is not applicable.

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|-----------------|
| A. One or more required AC or DC electrical power distribution subsystems inoperable. | A.1 Declare associated supported required feature(s) inoperable. | Immediately |
| | <u>OR</u> | |
| | A.2.1 Suspend CORE ALTERATIONS. | Immediately |
| | <u>AND</u> | |
| | A.2.2 Suspend movement of irradiated fuel assemblies in the secondary containment. | Immediately |
| | <u>AND</u> | |
| | | (continued) |

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|----------------|---|-----------------|
| A. (continued) | A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel. | Immediately |
| | <u>AND</u> | |
| | A.2.4 Initiate actions to restore required AC and DC electrical power distribution subsystems to OPERABLE status. | Immediately |
| | <u>AND</u> | |
| | A.2.5 Declare associated required shutdown cooling subsystem(s) inoperable and not in operation. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|-----------|
| SR 3.8.8.1 Verify correct breaker alignments and indicated power availability to required AC and DC electrical power distribution subsystems. | 7 days |

3.9 REFUELING OPERATIONS

3.9.1 Refueling Equipment Interlocks

LCO 3.9.1 The refueling equipment interlocks associated with the
refuel position shall be OPERABLE.

APPLICABILITY: During in-vessel fuel movement with equipment associated
 with the interlocks when the reactor mode switch is in
 the refuel position.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|--|--|-----------------|
| A. One or more required refueling equipment interlocks inoperable. | A.1 Suspend in-vessel fuel movement with equipment associated with the inoperable interlock(s). | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---------------|
| <p>SR 3.9.1.1 Perform CHANNEL FUNCTIONAL TEST on each of the following required refueling equipment interlock inputs:</p> <ul style="list-style-type: none"> a. All-rods-in, b. Refueling platform position, c. Refueling platform fuel grapple fuel-loaded, d. Refueling platform frame-mounted hoist fuel-loaded, and e. Refueling platform trolley-mounted hoist fuel-loaded. | <p>7 days</p> |

3.9 REFUELING OPERATIONS

3.9.2 Refuel Position One-Rod-Out Interlock

LCO 3.9.2 The refuel position one-rod-out interlock shall be OPERABLE.

APPLICABILITY: MODE 5 with the reactor mode switch in the refuel position and any control rod withdrawn.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|--|---|-----------------|
| A. Refuel position one-rod-out interlock inoperable. | A.1 Suspend control rod withdrawal. | Immediately |
| | <u>AND</u>
A.2 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|--------------|---|-----------|
| SR 3.9.2.1 | Verify reactor mode switch locked in refuel position. | 12 hours |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | FREQUENCY |
|---|---------------|
| <p>SR 3.9.2.2 -----NOTE-----
Not required to be performed until 1 hour
after any control rod is withdrawn.

Perform CHANNEL FUNCTIONAL TEST.</p> | <p>7 days</p> |

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3.9 REFUELING OPERATIONS

3.9.3 Control Rod Position

LCO 3.9.3 All control rods shall be fully inserted.

APPLICABILITY: When loading fuel assemblies into the core.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|--|-----------------|
| A. One or more control rods not fully inserted. | A.1 Suspend loading fuel assemblies into the core. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|--|-----------|
| SR 3.9.3.1 Verify all control rods are fully inserted. | 12 hours |



3.9 REFUELING OPERATIONS

3.9.4 Control Rod Position Indication

LCO 3.9.4 Each control rod "full-in" position indication channel shall be OPERABLE.

APPLICABILITY: MODE 5.

ACTIONS

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

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|--|---|-----------------|
| A. One or more required control rod position indication channels inoperable. | A.1.1 Suspend in-vessel fuel movement. | Immediately |
| | <u>AND</u> | |
| | A.1.2 Suspend control rod withdrawal. | Immediately |
| | <u>AND</u> | |
| | A.1.3 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies. | Immediately |
| | <u>OR</u> | |
| | | (continued) |

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|----------------|---|-----------------|
| A. (continued) | A.2.1 Initiate action to fully insert the control rod associated with the inoperable position indicator. | Immediately |
| | <u>AND</u>
A.2.2 Initiate action to disarm the control rod drive associated with the fully inserted control rod. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|--|
| SR 3.9.4.1 Verify each channel has no "full-in" indication on each control rod that is not "full-in." | Each time the control rod is withdrawn from the "full-in" position |

3.9 REFUELING OPERATIONS

3.9.5 Control Rod OPERABILITY—Refueling

LCO 3.9.5 Each withdrawn control rod shall be OPERABLE.

APPLICABILITY: MODE 5.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|--|-----------------|
| A. One or more withdrawn control rods inoperable. | A.1 Initiate action to fully insert inoperable withdrawn control rods. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|--|-----------|
| SR 3.9.5.1 -----NOTE-----
Not required to be performed until 7 days after the control rod is withdrawn.

Insert each withdrawn control rod at least one notch. | 7 days |
| SR 3.9.5.2 Verify each withdrawn control rod scram accumulator pressure is \geq 940 psig. | 7 days |

3.9 REFUELING OPERATIONS

3.9.6 Reactor Pressure Vessel (RPV) Water Level—Irradiated Fuel

LC0 3.9.6 RPV water level shall be ≥ 22 ft above the top of the RPV flange.

APPLICABILITY: During movement of irradiated fuel assemblies within the RPV.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|--------------------------------------|---|-----------------|
| A. RPV water level not within limit. | A.1 Suspend movement of irradiated fuel assemblies within the RPV. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|-----------|
| SR 3.9.6.1 Verify RPV water level is ≥ 22 ft above the top of the RPV flange. | 24 hours |

3.9 REFUELING OPERATIONS

3.9.7 Reactor Pressure Vessel (RPV) Water Level—New Fuel or Control Rods

LCO 3.9.7 RPV water level shall be ≥ 22 ft above the top of irradiated fuel assemblies seated within the RPV.

APPLICABILITY: During movement of new fuel assemblies or handling of control rods within the RPV when irradiated fuel assemblies are seated within the RPV.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|--------------------------------------|--|-----------------|
| A. RPV water level not within limit. | A.1 Suspend movement of new fuel assemblies and handling of control rods within the RPV. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|--|-----------|
| SR 3.9.7.1 Verify RPV water level is ≥ 22 ft above the top of irradiated fuel assemblies seated within the RPV. | 24 hours |

3.9 REFUELING OPERATIONS

3.9.8 Residual Heat Removal (RHR)—High Water Level

LCO 3.9.8 One RHR shutdown cooling subsystem shall be OPERABLE and in operation.

-----NOTE-----
The required RHR shutdown cooling subsystem may be removed from operation for up to 2 hours per 8 hour period.

APPLICABILITY: MODE 5 with irradiated fuel in the reactor pressure vessel (RPV) and with the water level \geq 22 ft above the top of the RPV flange.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|--|--|
| A. Required RHR shutdown cooling subsystem inoperable. | A.1 Verify an alternate method of decay heat removal is available. | 1 hour
<u>AND</u>
Once per 24 hours thereafter |
| B. Required Action and associated Completion Time of Condition A not met. | B.1 Suspend loading irradiated fuel assemblies into the RPV.
<u>AND</u>
B.2 Initiate action to restore secondary containment to OPERABLE status.
<u>AND</u> | Immediately

Immediately

(continued) |

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|--|--|---|
| B. (continued) | B.3 Initiate action to restore one standby gas treatment subsystem to OPERABLE status. | Immediately |
| | <u>AND</u>
B.4 Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated. | Immediately |
| C. No RHR shutdown cooling subsystem in operation. | C.1 Verify reactor coolant circulation by an alternate method. | 1 hour from discovery of no reactor coolant circulation |
| | <u>AND</u>
C.2 Monitor reactor coolant temperature. | <u>AND</u>
Once per 12 hours thereafter

Once per hour |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|--------------|---|-----------|
| SR 3.9.8.1 | Verify one RHR shutdown cooling subsystem is operating. | 12 hours |

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3.9 REFUELING OPERATIONS

3.9.9 Residual Heat Removal (RHR)—Low Water Level

LCO 3.9.9 Two RHR shutdown cooling subsystems shall be OPERABLE, and one RHR shutdown cooling subsystem shall be in operation.

-----NOTE-----
The required operating shutdown cooling subsystem may be removed from operation for up to 2 hours per 8 hour period.

APPLICABILITY: MODE 5 with irradiated fuel in the reactor pressure vessel (RPV) and with the water level < 22 ft above the top of the RPV flange.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|--|
| A. One or two RHR shutdown cooling subsystems inoperable. | A.1 Verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem. | 1 hour
<u>AND</u>
Once per 24 hours thereafter |
| B. Required Action and associated Completion Time of Condition A not met. | B.1 Initiate action to restore secondary containment to OPERABLE status.

<u>AND</u> | Immediately

(continued) |

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|--|--|---|
| B. (continued) | B.2 Initiate action to restore one standby gas treatment subsystem to OPERABLE status. | Immediately |
| | <u>AND</u>
B.3 Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated. | Immediately |
| C. No RHR shutdown cooling subsystem in operation. | C.1 Verify reactor coolant circulation by an alternate method. | 1 hour from discovery of no reactor coolant circulation |
| | <u>AND</u>
C.2 Monitor reactor coolant temperature. | <u>AND</u>
Once per 12 hours thereafter

Once per hour |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|--|-----------|
| SR 3.9.9.1 Verify one RHR shutdown cooling subsystem
is operating. | 12 hours |

3.10 SPECIAL OPERATIONS

3.10.1 Inservice Leak and Hydrostatic Testing Operation

LCO 3.10.1 The average reactor coolant temperature specified in Table 1.1-1 for MODE 4 may be changed to "NA," and operation considered not to be in MODE 3; and the requirements of LCO 3.4.10, "Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown," may be suspended, to allow performance of an inservice leak or hydrostatic test provided the following MODE 3 LCOs are met:

- a. LCO 3.3.6.2, "Secondary Containment Isolation Instrumentation," Functions 1, 3, and 4 of Table 3.3.6.2-1;
- b. LCO 3.6.4.1, "Secondary Containment";
- c. LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)"; and
- d. LCO 3.6.4.3, "Standby Gas Treatment (SGT) System."

D APPLICABILITY: MODE 4 with average reactor coolant temperature > 200°F.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each requirement of the LCO.

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|--|-----------------|
| A. One or more of the above requirements not met. | <p>A.1 -----NOTE-----
Required Actions to be in MODE 4 include reducing average reactor coolant temperature to $\leq 200^{\circ}\text{F}$.
-----</p> <p>Enter the applicable Condition of the affected LCO.</p> | Immediately |
| | <p><u>OR</u></p> <p>A.2.1 Suspend activities that could increase the average reactor coolant temperature or pressure.</p> | Immediately |
| | <p><u>AND</u></p> <p>A.2.2 Reduce average reactor coolant temperature to $\leq 200^{\circ}\text{F}$.</p> | 24 hours |

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Inservice Leak and Hydrostatic Testing Operation
3.10.1

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---------------------------------------|
| SR 3.10.1.1 Perform the applicable SRs for the required
MODE 3 LCOs. | According to
the applicable
SRs |

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3.10 SPECIAL OPERATIONS

3.10.2 Reactor Mode Switch Interlock Testing

LCO 3.10.2 The reactor mode switch position specified in Table 1.1-1 for MODES 3, 4, and 5 may be changed to include the run, startup/hot standby, and refuel position, and operation considered not to be in MODE 1 or 2, to allow testing of instrumentation associated with the reactor mode switch interlock functions, provided:

- a. All control rods remain fully inserted in core cells containing one or more fuel assemblies; and
- b. No CORE ALTERATIONS are in progress.

APPLICABILITY: MODES 3 and 4 with the reactor mode switch in the run, startup/hot standby, or refuel position,
MODE 5 with the reactor mode switch in the run or startup/hot standby position.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|--|-----------------|
| A. One or more of the above requirements not met. | A.1 Suspend CORE ALTERATIONS except for control rod insertion. | Immediately |
| | <u>AND</u> | |
| | A.2 Fully insert all insertable control rods in core cells containing one or more fuel assemblies. | 1 hour |
| | <u>AND</u> | |
| | | (continued) |

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ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|----------------|--|-----------------|
| A. (continued) | A.3.1 Place the reactor mode switch in the shutdown position. | 1 hour |
| | <p style="text-align: center;"><u>OR</u></p> <p>A.3.2 -----NOTE-----
Only applicable in
MODE 5.
-----</p> <p>Place the reactor mode switch in the refuel position.</p> | 1 hour |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|--|-----------|
| SR 3.10.2.1 Verify all control rods are fully inserted in core cells containing one or more fuel assemblies. | 12 hours |
| SR 3.10.2.2 Verify no CORE ALTERATIONS are in progress. | 24 hours |

3.10 SPECIAL OPERATIONS

3.10.3 Single Control Rod Withdrawal—Hot Shutdown

LCO 3.10.3 The reactor mode switch position specified in Table 1.1-1 for MODE 3 may be changed to include the refuel position, and operation considered not to be in MODE 2, to allow withdrawal of a single control rod, provided the following requirements are met:

- a. LCO 3.9.2, "Refuel Position One-Rod-Out Interlock";
- b. LCO 3.9.4, "Control Rod Position Indication";
- c. All other control rods are fully inserted; and
- d. 1. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," MODE 5 requirements for Functions 1.a, 1.b, 7.a, 7.b, 10, and 11 of Table 3.3.1.1-1, and
LCO 3.9.5, "Control Rod OPERABILITY—Refueling,"

OR

2. All other control rods in a five by five array centered on the control rod being withdrawn are disarmed, at which time LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," MODE 3 requirements may be changed to allow the single control rod withdrawn to be assumed to be the highest worth control rod.

APPLICABILITY: MODE 3 with the reactor mode switch in the refuel position.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each requirement of the LCO.

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|-----------------|
| A. One or more of the above requirements not met. | <p>A.1 -----NOTES-----</p> <p>1. Required Actions to fully insert all insertable control rods include placing the reactor mode switch in the shutdown position.</p> <p>2. Only applicable if the requirement not met is a required LCO.</p> <p>-----</p> <p>Enter the applicable Condition of the affected LCO.</p> | Immediately |
| | <u>OR</u> | |
| | <p>A.2.1 Initiate action to fully insert all insertable control rods.</p> | Immediately |
| | <p><u>AND</u></p> <p>A.2.2 Place the reactor mode switch in the shutdown position.</p> | 1 hour |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|--------------|--|---------------------------------|
| SR 3.10.3.1 | Perform the applicable SRs for the required LCOs. | According to the applicable SRs |
| SR 3.10.3.2 | <p>-----NOTE-----
 Not required to be met if SR 3.10.3.1 is satisfied for LCO 3.10.3.d.1 requirements.
 -----</p> <p>Verify all control rods, other than the control rod being withdrawn, in a five by five array centered on the control rod being withdrawn, are disarmed.</p> | 24 hours |
| SR 3.10.3.3 | Verify all control rods, other than the control rod being withdrawn, are fully inserted. | 24 hours |

3.10 SPECIAL OPERATIONS

3.10.4 Single Control Rod Withdrawal—Cold Shutdown

LCO 3.10.4 The reactor mode switch position specified in Table 1.1-1 for MODE 4 may be changed to include the refuel position, and operation considered not to be in MODE 2, to allow withdrawal of a single control rod, and subsequent removal of the associated control rod drive (CRD) if desired, provided the following requirements are met:

- a. All other control rods are fully inserted;
- b. 1. LCO 3.9.2, "Refuel Position One-Rod-Out Interlock," and
LCO 3.9.4, "Control Rod Position Indication,"

OR

- 2. A control rod withdrawal block is inserted; and
- c. 1. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," MODE 5 requirements for Functions 1.a, 1.b, 7.a, 7.b, 10, and 11 of Table 3.3.1.1-1,
LCO 3.3.8.2, "Reactor Protection System (RPS) Electric Power Monitoring," MODE 5 requirements, and
LCO 3.9.5, "Control Rod OPERABILITY—Refueling,"

OR

- 2. All other control rods in a five by five array centered on the control rod being withdrawn are disarmed, at which time LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," MODE 5 requirements may be changed to allow the single control rod withdrawn to be assumed to be the highest worth control rod.

APPLICABILITY: MODE 4 with the reactor mode switch in the refuel position.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each requirement of the LCO.

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|--|---|-----------------|
| A. One or more of the above requirements not met with the affected control rod insertable. | <p>A.1 -----NOTES-----</p> <p>1. Required Actions to fully insert all insertable control rods include placing the reactor mode switch in the shutdown position.</p> <p>2. Only applicable if the requirement not met is a required LCO.</p> <p>-----</p> <p>Enter the applicable Condition of the affected LCO.</p> | Immediately |
| | <p><u>OR</u></p> <p>A.2.1 Initiate action to fully insert all insertable control rods.</p> | Immediately |
| | <p><u>AND</u></p> <p>A.2.2 Place the reactor mode switch in the shutdown position.</p> | 1 hour |
| | | |

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ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|--|--|-----------------|
| B. One or more of the above requirements not met with the affected control rod not insertable. | B.1 Suspend withdrawal of the control rod and removal of associated CRD. | Immediately |
| | <u>AND</u> | |
| | B.2.1 Initiate action to fully insert all control rods. | Immediately |
| | <u>OR</u> | |
| | B.2.2 Initiate action to satisfy the requirements of this LCO. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|-----------------------------|
| SR 3.10.4.1 Perform the applicable SRs for the required LCOs. | According to applicable SRs |
| SR 3.10.4.2 -----NOTE-----
Not required to be met if SR 3.10.4.1 is satisfied for LCO 3.10.4.c.1 requirements.

Verify all control rods, other than the control rod being withdrawn, in a five by five array centered on the control rod being withdrawn, are disarmed. | 24 hours |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|--------------|--|-----------|
| SR 3.10.4.3 | Verify all control rods, other than the control rod being withdrawn, are fully inserted. | 24 hours |
| SR 3.10.4.4 | -----NOTE-----
Not required to be met if SR 3.10.4.1 is satisfied for LCO 3.10.4.b.1 requirements.

Verify a control rod withdrawal block is inserted. | 24 hours |

3.10 SPECIAL OPERATIONS

3.10.5 Single Control Rod Drive (CRD) Removal—Refueling

LCO 3.10.5 The requirements of LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"; LCO 3.3.8.2, "Reactor Protection System (RPS) Electric Power Monitoring"; LCO 3.9.1, "Refueling Equipment Interlocks"; LCO 3.9.2, "Refuel Position One-Rod-Out Interlock"; LCO 3.9.4, "Control Rod Position Indication"; and LCO 3.9.5, "Control Rod OPERABILITY—Refueling," may be suspended in MODE 5 to allow the removal of a single CRD associated with a control rod withdrawn from a core cell containing one or more fuel assemblies, provided the following requirements are met:

- a. All other control rods are fully inserted;
- b. All other control rods in a five by five array centered on the withdrawn control rod are disarmed;
- c. A control rod withdrawal block is inserted, and LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," MODE 5 requirements may be changed to allow the single control rod withdrawn to be assumed to be the highest worth control rod; and
- d. No other CORE ALTERATIONS are in progress.

APPLICABILITY: MODE 5 with LCO 3.9.5 not met.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|-----------------|
| A. One or more of the above requirements not met. | A.1 Suspend removal of the CRD mechanism. | Immediately |
| | <u>AND</u> | (continued) |

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|----------------|---|-----------------|
| A. (continued) | A.2.1 Initiate action to fully insert all control rods. | Immediately |
| | <u>OR</u>
A.2.2 Initiate action to satisfy the requirements of this LCO. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|-----------|
| SR 3.10.5.1 Verify all controls rods, other than the control rod withdrawn for the removal of the associated CRD, are fully inserted. | 24 hours |
| SR 3.10.5.2 Verify all control rods, other than the control rod withdrawn for the removal of the associated CRD, in a five by five array centered on the control rod withdrawn for the removal of the associated CRD, are disarmed. | 24 hours |
| SR 3.10.5.3 Verify a control rod withdrawal block is inserted. | 24 hours |

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SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | FREQUENCY |
|--|----------------------------|
| SR 3.10.5.4 Perform SR 3.1.1.1. | According to
SR 3.1.1.1 |
| SR 3.10.5.5 Verify no other CORE ALTERATIONS are in
progress. | 24 hours |

3.10 SPECIAL OPERATIONS

3.10.6 Multiple Control Rod Withdrawal—Refueling

LCO 3.10.6 The requirements of LCO 3.9.3, "Control Rod Position"; LCO 3.9.4, "Control Rod Position Indication"; and LCO 3.9.5, "Control Rod OPERABILITY—Refueling," may be suspended, and the "full-in" position indicators may be bypassed for any number of control rods in MODE 5, to allow withdrawal of these control rods, removal of associated control rod drives (CRDs), or both, provided the following requirements are met:

- a. The four fuel assemblies are removed from the core cells associated with each control rod or CRD to be removed;
- b. All other control rods in core cells containing one or more fuel assemblies are fully inserted; and
- c. Fuel assemblies shall only be loaded in compliance with an approved spiral reload sequence.

APPLICABILITY: MODE 5 with LCO 3.9.3, LCO 3.9.4, or LCO 3.9.5 not met.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|--|-----------------|
| A. One or more of the above requirements not met. | A.1 Suspend withdrawal of control rods and removal of associated CRDs. | Immediately |
| | <u>AND</u> | |
| | A.2 Suspend loading fuel assemblies. | Immediately |
| | <u>AND</u> | |
| | | (continued) |



ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|----------------|--|-----------------|
| A. (continued) | A.3.1 Initiate action to fully insert all control rods in core cells containing one or more fuel assemblies. | Immediately |
| | <u>OR</u>
A.3.2 Initiate action to satisfy the requirements of this LCO. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|-----------|
| SR 3.10.6.1 Verify the four fuel assemblies are removed from core cells associated with each control rod or CRD removed. | 24 hours |
| SR 3.10.6.2 Verify all other control rods in core cells containing one or more fuel assemblies are fully inserted. | 24 hours |
| SR 3.10.6.3 -----NOTE-----
Only required to be met during fuel loading.

Verify fuel assemblies being loaded are in compliance with an approved spiral reload sequence. | 24 hours |

3.10 SPECIAL OPERATIONS

3.10.7 Control Rod Testing—Operating

LCO 3.10.7 The requirements of LCO 3.1.6, "Rod Pattern Control," may be suspended to allow performance of SDM demonstrations, control rod scram time testing, and control rod friction testing provided:

- a. The banked position withdrawal sequence requirements of SR 3.3.2.1.8 are changed to require the control rod sequence to conform to the specified test sequence.

OR

- b. The RWM is bypassed; the requirements of LCO 3.3.2.1, "Control Rod Block Instrumentation," Function 2 are suspended; and conformance to the approved control rod sequence for the specified test is verified by a second licensed operator or other qualified member of the technical staff.

APPLICABILITY: MODES 1 and 2 with LCO 3.1.6 not met.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|-------------------------------------|---|-----------------|
| A. Requirements of the LCO not met. | A.1 Suspend performance of the test and exception to LCO 3.1.6. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|--|
| <p>SR 3.10.7.1 -----NOTE-----
 Not required to be met if SR 3.10.7.2
 satisfied.
 -----</p> <p>Verify movement of control rods is in
 compliance with the approved control rod
 sequence for the specified test by a second
 licensed operator or other qualified member
 of the technical staff.</p> | <p>During control
 rod movement</p> |
| <p>SR 3.10.7.2 -----NOTE-----
 Not required to be met if SR 3.10.7.1
 satisfied.
 -----</p> <p>Verify control rod sequence input to the
 RWM is in conformance with the approved
 control rod sequence for the specified
 test.</p> | <p>Prior to
 control rod
 movement</p> |



3.10 SPECIAL OPERATIONS

3.10.8 SHUTDOWN MARGIN (SDM) Test—Refueling

LCO 3.10.8 The reactor mode switch position specified in Table 1.1-1 for MODE 5 may be changed to include the startup/hot standby position, and operation considered not to be in MODE 2, to allow SDM testing, provided the following requirements are met:

- a. LCO 3.3.1.1, "Reactor Protection System Instrumentation," MODE 2 requirements for Function 2.a and 2.d of Table 3.3.1.1-1;
- b. 1. LCO 3.3.2.1, "Control Rod Block Instrumentation," MODE 2 requirements for Function 2 of Table 3.3.2.1-1, with banked position withdrawal sequence requirements of SR 3.3.2.1.8 changed to require the control rod sequence to conform to the SDM test sequence,

OR

2. Conformance to the approved control rod sequence for the SDM test is verified by a second licensed operator or other qualified member of the technical staff;
- c. Each withdrawn control rod shall be coupled to the associated control rod drive (CRD);
- d. All control rod withdrawals during out of sequence control rod moves shall be made in notch out mode;
- e. No other CORE ALTERATIONS are in progress; and
- f. CRD charging water header pressure \geq 940 psig.

APPLICABILITY: MODE 5 with the reactor mode switch in startup/hot standby position.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|--|-------------------------------|
| <p>-----NOTE-----
Separate Condition entry
is allowed for each
control rod.
-----</p> <p>A. One or more control
rods not coupled to
its associated CRD.</p> | <p>-----NOTE-----
Rod worth minimizer may be
bypassed as allowed by LCO
3.3.2.1, "Control Rod Block
Instrumentation," if
required, to allow insertion
of inoperable control rod and
continued operation.
-----</p> <p>A.1 Fully insert
inoperable control
rod.</p> <p><u>AND</u></p> <p>A.2 Disarm the associated
CRD.</p> | <p>3 hours</p> <p>4 hours</p> |
| <p>B. One or more of the
above requirements not
met for reasons other
than Condition A.</p> | <p>B.1 Place the reactor
mode switch in the
shutdown or refuel
position.</p> | <p>Immediately</p> |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|--------------|---|---------------------------------|
| SR 3.10.8.1 | Perform the MODE 2 applicable SRs for LCO 3.3.1.1, Functions 2.a and 2.d of Table 3.3.1.1-1. | According to the applicable SRs |
| SR 3.10.8.2 | -----NOTE-----
Not required to be met if SR 3.10.8.3 satisfied.

Perform the MODE 2 applicable SRs for LCO 3.3.2.1, Function 2 of Table 3.3.2.1-1. | According to the applicable SRs |
| SR 3.10.8.3 | -----NOTE-----
Not required to be met if SR 3.10.8.2 satisfied.

Verify movement of control rods is in compliance with the approved control rod sequence for the SDM test by a second licensed operator or other qualified member of the technical staff. | During control rod movement |
| SR 3.10.8.4 | Verify no other CORE ALTERATIONS are in progress. | 12 hours |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | FREQUENCY |
|---|---|
| <p>SR 3.10.8.5 Verify each withdrawn control rod does not go to the withdrawn overtravel position.</p> | <p>Each time the control rod is withdrawn to "full out" position</p> <p><u>AND</u></p> <p>Prior to satisfying LCO 3.10.8.c requirement after work on control rod or CRD System that could affect coupling</p> |
| <p>SR 3.10.8.6 Verify CRD charging water header pressure \geq 940 psig.</p> | <p>7 days</p> |

4.0 DESIGN FEATURES

4.1 Site Location

4.1.1 Site and Exclusion Area Boundaries

The site area shall include the area enclosed by the exclusion area plus the plant property lines that fall outside the exclusion area, as shown in Figure 4.1-1. The exclusion area boundary is a circle with its center at the reactor and a radius of 1950 meters.

4.1.2 Low Population Zone

The low population zone is all the land within a circle with its center at the reactor and a radius of 4827 meters.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 764 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material and water rods or channels. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all safety design bases. A limited number of lead fuel assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Rod Assemblies

The reactor core shall contain 185 cruciform shaped control rod assemblies. The control material shall be boron carbide and hafnium metal as approved by the NRC.

(continued)

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

- a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.2 of the FSAR; and
- b. A nominal 6.5 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

- a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.1 of the FSAR; and
- b. A nominal fuel assembly center to center spacing of 7.0 inches within rows and 12.25 inches between rows in the new fuel storage racks.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 583 ft 1.25 inches.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2658 fuel assemblies.



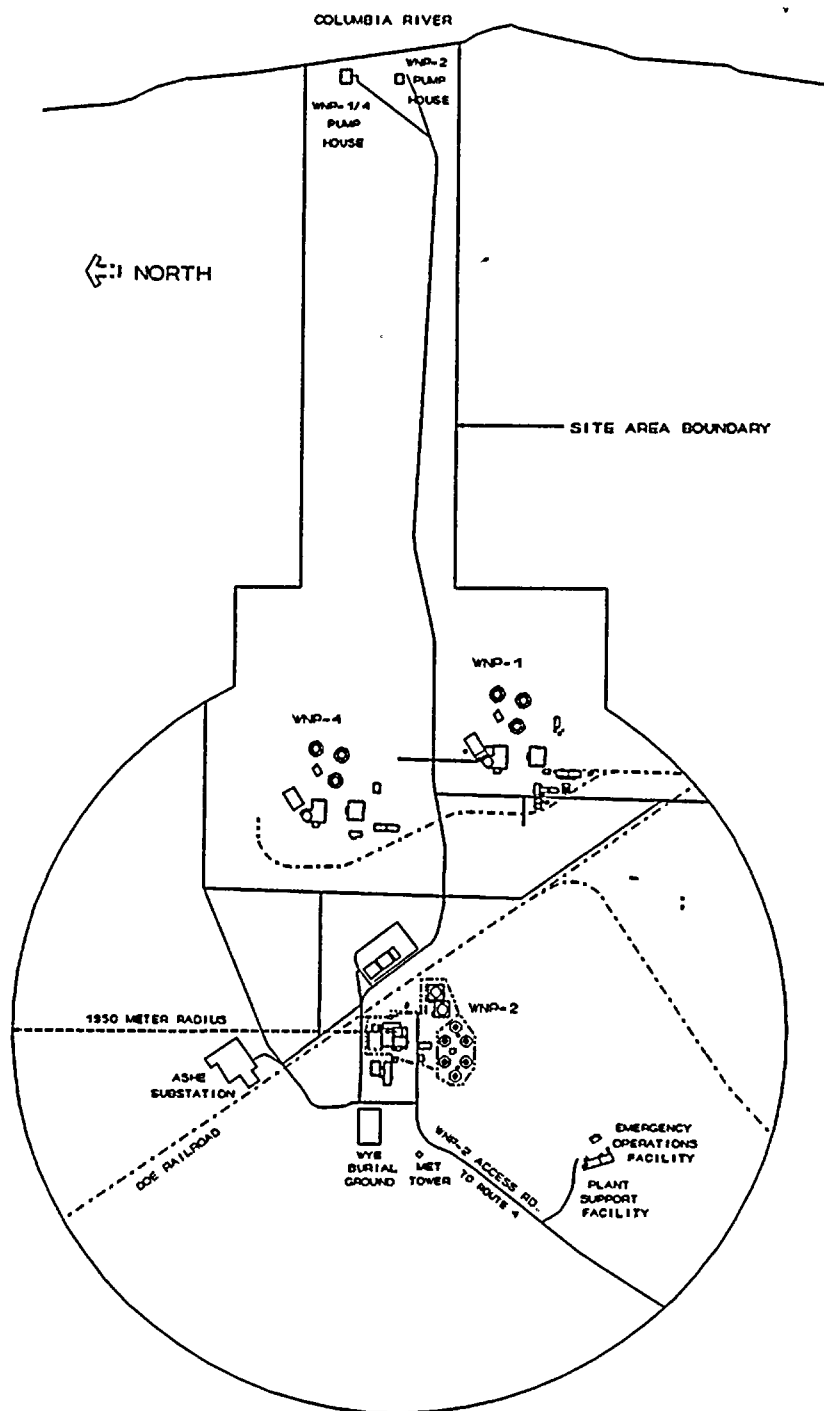


Figure 4.1-1
Site Area Boundary

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

- 5.1.1 The Plant General Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The Plant General Manager or his designee shall approve, prior to implementation, each proposed test, experiment, and modification to systems or equipment that affect nuclear safety.

- 5.1.2 The Shift Manager (SM) shall be responsible for the control room command function. During any absence of the SM from the control room while the unit is in MODE 1, 2, or 3, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the SM from the control room while the unit is in MODE 4 or 5, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.
-



5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

- a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the FSAR.
- b. The Plant General Manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.
- c. The Chief Executive Officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.
- d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2.2 Unit Staff

The unit staff organization shall include the following:

- a. At least two Equipment Operators shall be assigned when the unit is in MODE 1, 2, or 3; and at least one Equipment Operator shall be assigned when the unit is in MODE 4 or 5.

(continued)

5.2 Organization

5.2.2 Unit Staff (continued)

- b. At least one licensed Reactor Operator (RO) shall be present in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, or 3, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room.
- c. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and Specification 5.2.2.a for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.
- d. An individual qualified to implement radiation protection procedures shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.
- e. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, health physicists, equipment operators, and key maintenance personnel).

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a nominal 40 hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:

- 1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;
- 2. An individual should not be permitted to work more than 16 hours in any 24 hour period, nor more than 24 hours in any 48 hour period, nor more than 72 hours in any 7 day period, all excluding shift turnover time;
- 3. A break of at least 8 hours should be allowed between work periods, including shift turnover time;

(continued)

5.2 Organization

5.2.2 Unit Staff (continued)

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized in advance by the Plant General Manager or his designee, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Plant General Manager or his designee to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

- f. The Operations Manager or Assistant Operations Manager shall hold an SRO license.
 - g. The Shift Technical Advisor (STA) shall provide advisory technical support in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.
-

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS N18.1-1971, for comparable positions described in the FSAR, except for:

- a. The Operations Manager, who shall meet the requirements of ANSI/ANS N18.1-1971 with the exception that in lieu of meeting the stated ANSI/ANS requirement to hold a Senior Reactor Operator (SRO) license at the time of appointment to the position, the Operations Manager shall:
 - 1. Hold an SRO license at the time of appointment;
 - 2. Have held an SRO license; or
 - 3. Have been certified for equivalent SRO knowledge; and
 - b. The Radiation Protection Manager, who shall meet or exceed the qualifications of Regulatory Guide 1.8, Revision 1-R, May 1977.
-

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

- 5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:
- a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;
 - b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;
 - c. Quality assurance program for radioactive effluent and radiological environmental monitoring;
 - d. Fire Protection Program implementation; and
 - e. All programs specified in Specification 5.5.
-

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

- a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and
- b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release reports required by Specification 5.6.2 and Specification 5.6.3.
- c. Licensee initiated changes to the ODCM:
 1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:
 - (a) Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and
 - (b) A determination that the change(s) maintain the levels of radioactive effluent control required pursuant to 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and do not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;
 2. Shall become effective after review and acceptance by the Plant Operations Committee and the approval of the Plant General Manager; and

(continued)

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

3. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of, or concurrent with, the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include the Low Pressure Core Spray, High Pressure Core Spray, Residual Heat Removal, Reactor Core Isolation Cooling, hydrogen recombiner, process sampling, containment monitoring, and Standby Gas Treatment. The program shall include the following:

- a. Preventive maintenance and periodic visual inspection requirements; and
- b. Integrated leak test requirements for each system at 24 month intervals or less.

The provisions of SR 3.0.2 are applicable to the 24 month Frequency for performing integrated system leak test activities.

5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive iodines, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

- a. Training of personnel;
- b. Procedures for sampling and analysis; and

(continued)

5.5 Programs and Manuals

5.5.3 Post Accident Sampling (continued)

- c. Provisions for maintenance of sampling and analysis equipment.

5.5.4 Radioactive Effluent Controls Program

This program, conforming to 10 CFR 50.36a, provides for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

- a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;
- b. Limitations on the concentrations of radioactive material released in liquid effluents from the site to unrestricted areas, conforming to 10 times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001 - 20.2402;
- c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents pursuant to 10 CFR 20.1302 and with the methodology and parameters in the ODCM;
- d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;
- e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;
- f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

- g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary shall be limited to the following:
 - 1. For noble gases: less than or equal to a dose rate of 500 mrem/yr to the total body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and
 - 2. For iodine-131, iodine-133, tritium, and for all radionuclides in particulate form with half lives > 8 days: less than or equal to a dose rate of 1500 mrem/yr to any organ;
- h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;
- i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;
- j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190; and
- k. Limitations on venting and purging of the primary containment through the Standby Gas Treatment System to maintain releases as low as reasonably achievable.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the FSAR, Table 3.9-1, Note 1, cyclic and transient occurrences to ensure that components are maintained within the design limits.

(continued)

5.5 Programs and Manuals (continued)

5.5.6 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves.

- a. Testing Frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

| <u>ASME Boiler and Pressure
Vessel Code and
applicable Addenda
terminology for
inservice testing
activities</u> | <u>Required Frequencies
for performing inservice
testing activities</u> |
|---|---|
| Weekly | At least once per 7 days |
| Monthly | At least once per 31 days |
| Quarterly or every
3 months | At least once per 92 days |
| Semiannually or
every 6 months | At least once per 184 days |
| Every 9 months | At least once per 276 days |
| Yearly or annually | At least once per 366 days |
| Biennially or every
2 years | At least once per 731 days |

- b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;
- c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and
- d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5.7 Ventilation Filter Testing Program (VFTP)

The VFTP shall establish the required testing of Engineered Safety Feature (ESF) filter ventilation systems.

Tests described in Specification 5.5.7.a and 5.5.7.b shall be performed once per 24 months; after each complete or partial replacement of the HEPA filter train or charcoal adsorber filter;

(continued)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

after any structural maintenance on the system housing; and, following significant painting, fire, or chemical release in any ventilation zone communicating with the system while it is in operation.

Tests described in Specification 5.5.7.c shall be performed once per 24 months; after 720 hours of system operation; after any structural maintenance on the system housing; and, following significant painting, fire, or chemical release in any ventilation zone communicating with the system while it is in operation.

Tests described in Specification 5.5.7.d and 5.5.7.e shall be performed once per 24 months.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test Frequencies.

- a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass $< 0.05\%$ when tested in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989 at the system flowrate specified below:

| ESF Ventilation System | Flowrate (cfm) |
|------------------------|----------------|
| SGT System | 4012 to 4902 |
| CREF System | 900 to 1100 |

- b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass $< 0.05\%$ when tested in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989 at the system flowrate specified below:

| ESF Ventilation System | Flowrate (cfm) |
|------------------------|----------------|
| SGT System | 4012 to 4902 |
| CREF System | 900 to 1100 |

(continued)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

- c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1986 (Method B for the SGT System and Method A for the CREF System) at a relative humidity greater than or equal to the value specified below:

| ESF Ventilation System | Penetration (%) | RH (%) |
|------------------------|-----------------|--------|
| SGT System | 0.175 | 70 |
| CREF System | 1.0 | 70 |

- d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below:

| ESF Ventilation System | Delta P
(inches wg) | Flowrate
(cfm) |
|------------------------|------------------------|-------------------|
| SGT System | < 8 | 4012 to 4902 |
| CREF System | < 6 | 900 to 1100 |

- e. Demonstrate that the heaters for each of the ESF systems dissipate the nominal value specified below when tested in accordance with ASME N510-1989:

| ESF Ventilation System | Wattage (kW) |
|------------------------|--------------|
| SGT System | 18.6 to 22.8 |
| CREF System | 4.5 to 5.5 |

5.5.8 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Main Condenser Offgas Treatment System and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

(continued)

5.5 Programs and Manuals

5.5.8 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

- a. The limits for concentrations of hydrogen in the Main Condenser Offgas Treatment System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and
- b. A surveillance program to ensure that the quantity of radioactivity contained in all outside temporary liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations greater than the limits of Appendix B, Table 2, Column 2 to 10 CFR 20.1001 - 20.2402, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program Surveillance Frequencies.

5.5.9 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program shall establish the required testing of both new fuel oil and stored fuel oil. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

- a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:
 1. An API gravity, a specific gravity, or an absolute specific gravity within limits,
 2. A kinematic viscosity, if gravity was not determined by comparison with the supplier's certificate, and a flash point within limits for ASTM 2-D fuel oil,

(continued)

5.5 Programs and Manuals

5.5.9 Diesel Fuel Oil Testing Program (continued)

3. A water and sediment content within limits or a clear and bright appearance with proper color;
- b. Other properties for ASTM 2-D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and
- c. Total particulate concentration of the fuel oil in the storage tanks is ≤ 10 mg/l when tested every 31 days in accordance with ASTM D-2276, Method A-2 or A-3.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test Frequencies.

5.5.10 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases to these Technical Specifications.

- a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.
- b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:
 1. A change in the TS incorporated in the license; or
 2. A change to the FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.
- c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.
- d. Proposed changes that meet the criteria of 5.5.10.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

(continued)



5.5 Programs and Manuals (continued)

5.5.11 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6.

a. The SFDP shall contain the following:

1. Provisions for cross division checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;
2. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;
3. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and
4. Other appropriate limitations and remedial or compensatory actions.

b. A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or
2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or
3. A required system redundant to support system(s) for the supported systems described in b.1 and b.2 above is also inoperable.

(continued)

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

- c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.12 Primary Containment Leakage Rate Testing Program

The Primary Containment Leakage Rate Testing Program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception: Compensation for flow meter inaccuracies in excess of those specified in ANSI/ANS 56.8-1994 will be accomplished by increasing the actual instrument reading by the amount of the full scale inaccuracy when assessing the effect of local leak rates against the criteria established in Specification 5.5.12.a.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a , is 38 psig.

The maximum allowable primary containment leakage rate, L_a , at P_a , shall be 0.5% of primary containment air weight per day.

Leakage rate acceptance criteria are:

- a. Primary containment leakage rate acceptance criterion is $\leq 1.0 L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60 L_a$ for the Type B and Type C tests (except for main steam isolation valves) and $< 0.75 L_a$ for Type A tests;

(continued)

5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

- b. Primary containment air lock testing acceptance criteria are:
 - 1) Overall primary containment air lock leakage rate is $\leq 0.05 L_a$ when tested at $\geq P_a$; and
 - 2) For each door, leakage rate is $\leq 0.025 L_a$ when pressurized to ≥ 10 psig.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.



5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Radiation Exposure Report

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) for whom monitoring was performed, receiving an annual deep dose equivalent of > 100 mrem and the associated collective deep dose equivalent (reported in man-rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on electronic or pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions. The report shall be submitted by April 30 of each year.

5.6.2 Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the

(continued)



5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and the Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the safety/relief valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

- a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:
 1. The APLHGR for Specification 3.2.1;
 2. The MCPR for Specification 3.2.2;
 3. The LHGR for Specification 3.2.3; and
 4. The power-to-flow map for Specification 3.4.1.
- b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

1. ANF-1125(P)(A), and Supplements 1 and 2, "ANFB Critical Power Correlation," April 1990;
2. Letter, R.C. Jones (NRC) to R.A. Copeland (ANF), "NRC Approval of ANFB Additive Constants for ANF 9x9-9X BWR Fuel," dated November 14, 1990;
3. ANF-NF-524(P)(A), Revision 2 and Supplements 1 and 2, "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors," November 1990;
4. XN-NF-85-67(P)(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," September 1986;
5. ANF-89-014(P)(A), Revision 1 and Supplements 1 and 2, "Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels Corporation 9x9-IX and 9x9-9X BWR Reload Fuel," October 1991;
6. XN-NF-81-22(P)(A), "Generic Statistical Uncertainty Analysis Methodology," November 1983;
7. NEDE-24011-P-A-10-US, "General Electric Standard Application for Reactor Fuel," U.S. Supplement, March 1991;
8. NEDE-23785-1-PA, Revision 1, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Volume III, SAFER/GESTR Application Methodology," October 1984;
9. NEDO-20566A, "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K," September 1986;
10. EMF-CC-074(P)(A), "Volume 1 -- STAIF - A Computer Program for BWR Stability in the Frequency Domain, Volume 2 -- STAIF - A Computer Program for BWR Stability in the Frequency Domain, Code Qualification Report," July 1994;
11. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel," July 1996; and

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

12. WPPSS-FTS-131(A), Revision 1, "Applications Topical Report for BWR Design and Analysis," March 1996.
- c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.
- d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20.

5.7.1 High Radiation Areas with Dose Rates not Exceeding 1.0 rem/hour (at 30 centimeters from the radiation sources or from any surface penetrated by the radiation)

- a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.
- b. Access to, and activities in, each such area shall be controlled by means of a Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.
- c. Individuals qualified in radiation protection procedures (e.g., health physics technicians) and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are following plant radiation protection procedures for entry to, exit from, and work in such areas.
- d. Each individual or group entering such an area shall possess:
 1. A radiation monitoring device that continuously displays radiation dose rates in the area ("radiation monitoring and indicating device");
 2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached ("alarming dosimeter"), with an appropriate alarm setpoint;
 3. A radiation monitoring device that continuously transmits dose rate and cumulative dose to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area; or

(continued)

5.7 High Radiation Area

5.7.1 High Radiation Areas with Dose Rates not Exceeding 1.0 rem/hour (at 30 centimeters from the radiation sources or from any surface penetrated by the radiation) (continued)

4. A self-reading dosimeter and,
 - (a) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual at the work site, qualified in radiation protection procedures, equipped with a radiation monitoring and indicating device who is responsible for controlling personnel radiation exposure within the area, or
 - (b) Be under the surveillance, as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area.
- e. Except for individuals qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been established and entry personnel are knowledgeable of them.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation)

- a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked door, gate, or guard that prevents unauthorized entry, and in addition:
 1. All such door and gate keys shall be maintained under the administrative control of the Shift Manager or Health Physics supervision on duty; and
 2. Doors and gates shall remain locked or guarded except during periods of personnel or equipment entry or exit.

(continued)



5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation) (continued)

- b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.
- c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are following plant radiation protection procedures for entry to, exit from, and work in such areas.
- d. Each individual (whether alone or in a group) entering such an area shall possess:
 - 1. An alarming dosimeter with an appropriate alarm setpoint;
 - 2. A radiation monitoring device that continuously transmits dose rate and cumulative dose to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area;
 - 3. A self-reading dosimeter and,
 - (a) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring and indicating device who is responsible for controlling personnel exposure within the area, or
 - (b) Be under the surveillance, as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area; or

(continued)

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation) (continued)

4. A radiation monitoring and indicating device in those cases where the options of Specification 5.7.2.d.2 and 5.7.2.d.3, above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle.
 - e. Except for individuals qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been established and entry personnel are knowledgeable of them.
 - f. Such individual areas that are within a larger area that is controlled as a high radiation area, where no enclosure exists for purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, but shall be barricaded and conspicuously posted as a high radiation area, and a conspicuous, clearly visible flashing light shall be activated at the area as a warning device.
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B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires, and SLs ensure, that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (A00s).

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

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

While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross, rather than incremental, cladding deterioration. Therefore, the fuel cladding SL is defined with a margin to the conditions that would produce onset of transition boiling (i.e., MCPR = 1.00). These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity SL ensures that during normal operation and during A00s, at least 99.9% of the fuel rods in the core do not experience transition boiling.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp

(continued)

BASES

BACKGROUND
(continued)

reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The reactor vessel water level SL ensures that adequate core cooling capability is maintained during all MODES of reactor operation. Establishment of Emergency Core Cooling System initiation setpoints higher than this safety limit provides margin such that the safety limit will not be reached or exceeded.

APPLICABLE
SAFETY ANALYSES

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

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

2.1.1.1 Fuel Cladding Integrity

The use of the ANFB correlation is valid for critical power calculations at pressures > 600 psia and < 1500 psia and bundle mass fluxes $> 0.1 \times 10^6$ lb/hr-ft² and $< 1.5 \times 10^6$ lb/hr-ft² (Ref. 2). The use of the XL-S96 correlation is valid for critical power calculations at pressures > 392 psia and < 1262 psia and bundle mass fluxes $> 0.25 \times 10^6$ lb/hr-ft² and $< 1.55 \times 10^6$ lb/hr-ft² (Ref. 3). For operation at low pressures or low flows, the fuel cladding integrity SL is established by a limiting condition on core THERMAL POWER, with the following basis:

Provided that the water level in the vessel downcomer is maintained above the top of the active fuel, natural circulation is sufficient to ensure a minimum

(continued)

BASES

APPLICABLE
SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

bundle flow for all fuel assemblies that have a relatively high power and potentially can approach a critical heat flux condition. The minimum bundle flow is $> 28 \times 10^3$ lb/hr. The coolant minimum bundle flow and maximum flow area are such that the mass flux is $> 0.25 \times 10^6$ lb/hr-ft². Full scale critical power tests taken at pressures down to 14.7 psia indicate that the fuel assembly critical power at 0.25×10^6 lb/hr-ft² is approximately 3.35 Mwt. At 25% RTP, a bundle power of approximately 3.35 Mwt corresponds to a bundle radial peaking factor of > 2.9 , which is significantly higher than the expected peaking factor. Thus, a THERMAL POWER limit of 25% RTP for reactor pressures < 785 psig is conservative.

2.1.1.2 MCPR

The MCPR SL ensures sufficient conservatism in the operating MCPR limit that, in the event of an AOO from the limiting condition of operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (i.e., MCPR = 1.00) and the MCPR SL is based on a detailed statistical procedure that considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the SL is the uncertainty inherent in the critical power correlations. Reference 4 describes the methodology used in determining the MCPR SL for Siemens Power Corporation fuel. Reference 5 describes the methodology used in determining the MCPR SL for ABB CENO fuel.

The critical power correlations are based on a significant body of practical test data, providing a high degree of assurance that the critical power, as evaluated by the correlation, is within a small percentage of the actual critical power. As long as the core pressure and flow are within the range of validity of the critical power correlations, the assumed reactor conditions used in defining the SL introduce conservatism into the limit because bounding high radial power factors and bounding flat local peaking distributions are used to estimate the number

(continued)



BASES

APPLICABLE
SAFETY ANALYSES

2.1.1.2 MCPR (continued)

of rods in boiling transition. This conservatism and the inherent accuracy of the critical power correlations provide a reasonable degree of assurance that there would be no transition boiling in the core during sustained operation at the MCPR SL. If boiling transition were to occur, there is reason to believe that the integrity of the fuel would not be compromised. Significant test data accumulated by the NRC and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach. Much of the data indicate that BWR fuel can survive for an extended period of time in an environment of boiling transition.

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2, the reactor vessel water level is required to be above the top of the active irradiated fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes $< 2/3$ of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

SAFETY LIMITS

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

APPLICABILITY

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

(continued)



BASES (continued)

SAFETY LIMIT
VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 6). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and the probability of an accident occurring during this period is minimal.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
 2. ANF-1125(P)(A), Revision 0, including Supplements 1 and 2, April 1990.
 3. UR-89-210-P-A, "SVEA-96 Critical Power Experiments on a Full Scale 24-Rod Sub-Bundle," October 1993.
 4. ANF-524(P)(A), Revision 2, including Supplements 1 and 2, November 1990.
 5. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel," July 1996.
 6. 10 CFR 100.
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B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on reactor steam dome pressure protects the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to ensure that the design conditions are not exceeded during normal operation and anticipated operational occurrences (AOOs).

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

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

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

(continued)

BASES

BACKGROUND (continued)

The RCS pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to ASME, Boiler and Pressure Vessel Code, Section III, 1971 Edition, including Addenda through the summer of 1971 (Ref. 5), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The SL of 1325 psig, as measured in the reactor steam dome, is equivalent to 1375 psig at the lowest elevation of the RCS. The RCS is designed to ASME Code, Section III, 1971 Edition, including Addenda through the summer of 1971 (Ref. 5), for the reactor recirculation piping, which permits a maximum pressure transient of 125% of design pressures of 1250 psig for suction piping and 1550 psig for discharge piping. The RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings is 125% of design pressures of 1250 psig for suction piping and 1550 psig for discharge piping. The most limiting of these allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is established at 1325 psig as measured at the reactor steam dome.

APPLICABILITY

SL 2.1.2 applies in all MODES.

SAFETY LIMIT VIOLATIONS

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

(continued)

BASES (continued)

- REFERENCES
1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
 2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
 3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IW-5000.
 4. 10 CFR 100.
 5. ASME, Boiler and Pressure Vessel Code, 1971 Edition, Addenda, summer of 1971.
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B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

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

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

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

- a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and
- b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

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

(continued)

BASES

LCO 3.0.2
(continued)

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

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

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Condition no longer exists. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.11, "RCS Pressure and Temperature (P/T) Limits."

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

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

(continued)

BASES (continued)

LCO 3.0.3

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

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

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

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

(continued)

BASES

LCO 3.0.3
(continued)

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

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

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

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

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.7, "Spent Fuel Pool Water Level." LCO 3.7.7 has an Applicability of "During movement of irradiated fuel

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BASES

LCO 3.0.3
(continued)

assemblies in the associated fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.7 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.7 of "Suspend movement of irradiated fuel assemblies in the spent fuel storage pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

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

- a. Unit conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and
- b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the unit being required to exit the Applicability desired to be entered to comply with the Required Actions.

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

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BASES

LCO 3.0.4
(continued)

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

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

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

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

LCO 3.0.5

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

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

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D BASES

LCO 3.0.5
(continued)

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

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

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

D LCO 3.0.6

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

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements

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BASES

LCO 3.0.6
(continued)

related to the entry into multiple support and supported systems' LCO's Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the plant is maintained in a safe condition in the support system's Required Actions.

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

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

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

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform

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BASES

LCO 3.0.7
(continued)

special evolutions. Special Operations LCOs in Section 3.10 allow specified TS requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

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



B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

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| SRs | SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications in Sections 3.1 through 3.10 and apply at all times, unless otherwise stated. |
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| SR 3.0.1 | SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO. |
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Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

- a. The systems or components are known to be inoperable, although still meeting the SRs; or
- b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

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

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

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BASES

SR 3.0.1
(continued)

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

- a. Control rod drive maintenance during refueling that requires scram testing at > 800 psi. However, if other appropriate testing is satisfactorily completed and the scram time testing of SR 3.1.4.3 is satisfied, the control rod can be considered OPERABLE. This allows startup to proceed to reach 800 psi to perform other necessary testing.
 - b. Reactor Core Isolation Cooling (RCIC) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with RCIC considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.
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SR 3.0.2

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

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

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BASES

SR 3.0.2
(continued)

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

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

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

SR 3.0.3

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

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BASES

SR 3.0.3
(continued)

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

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

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

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

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

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

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

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BASES (continued)

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

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

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

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

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

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows

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BASES

SR 3.0.4
(continued)

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

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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

SDM requirements are specified to ensure:

- a. The reactor can be made subcritical from all operating conditions and transients and Design Basis Events;
- b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits; and
- c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

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

APPLICABLE SAFETY ANALYSES

Having sufficient SDM assures that the reactor will become and remain subcritical after all design basis accidents and transients. For example, SDM is assumed as an initial condition for the control rod removal error during a refueling accident (Ref. 2). The analysis of this reactivity insertion event assumes the refueling interlocks are OPERABLE when the reactor is in the refueling mode of operation. These interlocks prevent the withdrawal of more than one control rod from the core during refueling. (Special consideration and requirements for multiple control rod withdrawal during refueling are covered in Special Operations LCO 3.10.6, "Multiple Control Rod Withdrawal—Refueling.") The analysis assumes this condition is acceptable since the core will be shut down with the highest worth control rod withdrawn, if adequate SDM has been demonstrated.

SDM satisfies Criterion 2 of Reference 3.

(continued)



BASES (continued)

LCO The specified SDM limit accounts for the uncertainty in the demonstration of SDM by testing. Separate SDM limits are provided for testing where the highest worth control rod is determined analytically or by measurement. This is due to the reduced uncertainty in the SDM test when the highest worth control rod is determined by measurement. When SDM is demonstrated by calculations not associated with a test (e.g., to confirm SDM during the fuel loading sequence), additional margin is included to account for uncertainties in the calculation. To ensure adequate SDM, a design margin is included to account for uncertainties in the design calculations (Ref. 4).

APPLICABILITY In MODES 1 and 2, SDM must be provided to assure shutdown capability. In MODES 3 and 4, SDM is required to ensure the reactor will be held subcritical with margin for a single withdrawn control rod. SDM is required in MODE 5 to prevent an inadvertent criticality during the withdrawal of a single control rod from a core cell containing one or more fuel assemblies (Ref. 2).

ACTIONS

A.1

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

B.1

If the SDM cannot be restored, the plant must be brought to MODE 3 within 12 hours, to prevent the potential for further reductions in available SDM (e.g., additional stuck control rods). The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS
(continued)C.1

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

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

With SDM not within limits in MODE 4, the operator must immediately initiate action to fully insert all insertable control rods. Action must continue until all insertable control rods are fully inserted. This action results in the least reactive condition for the core. Actions must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at least one Standby Gas Treatment (SGT) subsystem is OPERABLE; and secondary containment isolation capability is available in each associated secondary containment penetration flow path not isolated that is assumed to be isolated to mitigate radioactivity releases (i.e., at least one secondary containment isolation valve and associated instrumentation are OPERABLE, or other acceptable administrative controls to assure isolation capability. These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated). This may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the Surveillances needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, SRs may need to be performed to restore the component to OPERABLE status. Actions must continue until all required components are OPERABLE.

(continued)

BASES

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

With SDM not within limits in MODE 5, the operator must immediately suspend CORE ALTERATIONS that could reduce SDM, e.g., insertion of fuel in the core or the withdrawal of control rods. Suspension of these activities shall not preclude completion of movement of a component to a safe condition. Inserting control rods or removing fuel from the core will reduce the total reactivity and are therefore excluded from the suspended actions.

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

Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at least one SGT subsystem is OPERABLE; and secondary containment isolation capability is available in each associated secondary containment penetration flow path not isolated that is assumed to be isolated to mitigate radioactivity releases (i.e., at least one secondary containment isolation valve and associated instrumentation are OPERABLE or other acceptable administrative controls to assure isolation capability. These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated). This may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the Surveillances needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, SRs may need to be performed to restore the component to OPERABLE status. Actions must continue until all required components are OPERABLE.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTSSR 3.1.1.1

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

The SDM may be demonstrated during an in-sequence control rod withdrawal, in which the highest worth control rod is analytically determined, or during local criticals, where the highest worth control rod is determined by testing. Local critical tests require the withdrawal of out of sequence control rods. This testing would therefore require bypassing of the rod worth minimizer to allow the out of sequence withdrawal, and therefore additional requirements must be met (see LCO 3.10.7, "Control Rod Testing—Operating").

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

During MODES 3 and 4, analytical calculation of SDM may be used to assure the requirements of SR 3.1.1.1 are met. During MODE 5, adequate SDM is also required to ensure the reactor does not reach criticality during control rod

(continued)



BASES

SURVEILLANCE
REQUIREMENTSSR 3.1.1.1 (continued)

withdrawals. An evaluation of each in vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload or reload sequences inherently satisfy the SR, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. FSAR, Section 15.4.1.1.
3. 10 CFR 50.36(c)(2)(ii).
4. FSAR, Section 4.3.2.4.1.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Anomalies

BASES

BACKGROUND

In accordance with GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable such that subcriticality is maintained under cold conditions and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Reactivity Anomalies is used as a measure of the predicted versus measured core reactivity during power operation. The continual confirmation of core reactivity is necessary to ensure that the Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity anomaly could be the result of unanticipated changes in fuel reactivity, control rod worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in ensuring the reactor can be brought safely to cold, subcritical conditions.

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

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

(continued)

D
BASES

BACKGROUND
(continued)

the excess positive reactivity is compensated by burnable absorbers (e.g., gadolinia), control rods, and whatever neutron poisons (mainly xenon and samarium) are present in the fuel.

The predicted core reactivity, as represented by k_{eff} , is calculated by a 3D core simulator code as a function of cycle exposure. This calculation is performed for projected operating states and conditions throughout the cycle. The monitored k_{eff} is calculated by the core monitoring system for actual plant conditions and is then compared to the predicted value for the cycle exposure.

APPLICABLE
SAFETY ANALYSES

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations (Ref. 2). In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod drop accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity anomaly provides additional assurance that the nuclear methods provide an accurate representation of the core reactivity.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted k_{eff} for identical core conditions at BOC do not reasonably agree, then the assumptions used in the reload cycle design analysis or the calculation models used to predict k_{eff} may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured value. Thereafter, any significant deviations in the measured k_{eff} from the predicted k_{eff} that develop during fuel depletion may be an indication that the assumptions of the DBA and transient analyses are no longer valid, or that an unexpected change in core conditions has occurred.

Reactivity Anomalies satisfy Criterion 2 of Reference 3.

(continued)

BASES (continued)

LCO

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

APPLICABILITY

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

ACTIONS

A.1

Should an anomaly develop between measured and predicted core reactivity, the core reactivity difference must be restored to within the limit to ensure continued operation is within the core design assumptions. Restoration to within the limit could be performed by an evaluation of the core design and safety analysis to determine the reason for the anomaly. This evaluation normally reviews the core conditions to determine their consistency with input to design calculations. Measured core and process parameters

(continued)

BASES

ACTIONS

A.1 (continued)

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

B.1

If the core reactivity cannot be restored to within the 1% $\Delta k/k$ limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.2.1

Verifying the reactivity difference between the monitored and predicted core k_{eff} is within the limits of the LCO provides further assurance that plant operation is maintained within the assumptions of the DBA and transient analyses. The core monitoring system calculates the core k_{eff} for the reactor conditions obtained from plant instrumentation. A comparison of the monitored core k_{eff} to the predicted core k_{eff} at the same cycle exposure is used to calculate the reactivity difference. The comparison is required when the core reactivity has potentially changed by a significant amount. This may occur following a refueling in which new fuel assemblies are loaded, fuel assemblies are shuffled within the core, or control rods are replaced or shuffled. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from another core location. Also, core reactivity changes during the cycle. The 24 hour interval after

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.2.1 (continued)

reaching equilibrium conditions following a startup is based on the need for equilibrium xenon concentrations in the core, such that an accurate comparison between the monitored and predicted core k_{eff} values can be made. For the purposes of this SR, the reactor is assumed to be at equilibrium conditions when steady state operations (no control rod movement or core flow changes) at $\geq 75\%$ RTP have been obtained. The 1000 MWD/T Frequency was developed, considering the relatively slow change in core reactivity with exposure and operating experience related to variations in core reactivity. This comparison requires the core to be operating at power levels which minimize the uncertainties and measurement errors, in order to obtain meaningful results. Therefore, the comparison is only done when in MODE 1.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.
 2. FSAR, Chapters 15 and 15.F.
 3. 10 CFR 50.36(c)(2)(ii).
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Control Rod OPERABILITY

BASES

BACKGROUND

Control rods are components of the Control Rod Drive (CRD) System, which is the primary reactivity control system for the reactor. In conjunction with the Reactor Protection System, the CRD System provides the means for the reliable control of reactivity changes to ensure that under conditions of normal operation, including anticipated operational occurrences, specified acceptable fuel design limits are not exceeded. In addition, the control rods provide the capability to hold the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the CRD System. The CRD System is designed to satisfy the requirements of GDC 26, GDC 27, GDC 28, and GDC 29, (Ref. 1).

The CRD System consists of 185 locking piston control rod drive mechanisms (CRDMs) and a hydraulic control unit for each drive mechanism. The locking piston type CRDM is a double-acting hydraulic piston, which uses condensate water as the operating fluid. Accumulators provide additional energy for scram. An index tube and piston, coupled to the control rod, are locked at fixed increments by a collet mechanism. The collet fingers engage notches in the index tube to prevent unintentional withdrawal of the control rod, but without restricting insertion.

This Specification, along with LCO 3.1.4, "Control Rod Scram Times," and LCO 3.1.5, "Control Rod Scram Accumulators," ensure that the performance of the control rods in the event of a Design Basis Accident (DBA) or transient meets the assumptions used in the safety analyses of References 2, 3, 4, 5, and 6.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in the evaluations involving control rods are presented in References 2, 3, 4, 5, and 6. The control rods provide the primary means for rapid reactivity control (reactor scram), for maintaining the reactor subcritical, and for limiting the potential effects of reactivity insertion events caused by malfunctions in the CRD System.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

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

The control rods also protect the fuel from damage that could result in release of radioactivity. The limits protected are the MCPR Safety Limit (SL) (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), and the fuel damage limit (see Bases for LCO 3.1.6, "Rod Pattern Control") during reactivity insertion events.

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

Control rod OPERABILITY satisfies Criterion 3 of Reference 7.

LCO

OPERABILITY of an individual control rod is based on a combination of factors, primarily the scram insertion times, the control rod coupling integrity, and the ability to determine the control rod position. Accumulator OPERABILITY is addressed by LCO 3.1.5. The associated scram accumulator status for a control rod only affects the scram insertion times and therefore an inoperable accumulator does not immediately require declaring a control rod inoperable. Although not all control rods are required to be OPERABLE to

(continued)

BASES

LCO
(continued) satisfy the intended reactivity control requirements, strict control over the number and distribution of inoperable control rods is required to satisfy the assumptions of the DBA and transient analyses.

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

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

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

A control rod is considered stuck if it will not insert by either CRD drive water or scram pressure. With a fully inserted control rod stuck, no actions are required as long as the control rod remains fully inserted. The Required Actions are modified by a Note that allows the RWM to be bypassed if required to allow continued operation. LCO 3.3.2.1, "Control Rod Block Instrumentation," provides additional requirements when the RWM is bypassed to ensure compliance with the CRDA analysis. With one withdrawn control rod stuck, the local scram reactivity rate assumptions may not be met if the stuck control rod separation criteria are not met. Therefore, a verification that the separation criteria are met must be performed immediately. The stuck control rod separation criteria are met if: a) the stuck control rod is separated in all directions from each "slow" control rod by any combination of two or more fully inserted control rods or OPERABLE, withdrawn control rods that are not "slow"; and b) two or

(continued)

BASES

ACTIONS

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

less inoperable or "slow" control rods are in the same group as the stuck control rod. The description of "slow" control rod is provided in LCO 3.1.4, "Control Rod Scram Times." In addition, the associated control rod drive must be disarmed within 2 hours. The allowed Completion Time of 2 hours is acceptable, considering the reactor can still be shut down, assuming no additional control rods fail to insert, and provides a reasonable amount of time to perform the Required Action in an orderly manner. The control rod must be isolated from both scram and normal insert and withdraw pressure. Isolating the control rod from scram and normal insert and withdraw pressure prevents damage to the CRDM. The control rod should be isolated from scram by isolating the hydraulic control unit from scram and normal insert and withdraw pressure, while maintaining cooling water to the CRD.

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

To allow continued operation with a withdrawn control rod stuck, an evaluation of adequate SDM is also required within 72 hours. Should a DBA or transient require a shutdown, to preserve the single failure criterion an additional control rod would have to be assumed to have failed to insert when required. Therefore, the original SDM demonstration may not

(continued)

BASES

ACTIONS

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

be valid. The SDM must therefore be evaluated (by measurement or analysis) with the stuck control rod at its stuck position and the highest worth OPERABLE control rod assumed to be fully withdrawn.

The allowed Completion Time of 72 hours to verify SDM is adequate, considering that with a single control rod stuck in a withdrawn position, the remaining OPERABLE control rods are capable of providing the required scram and shutdown reactivity. Failure to reach MODE 4 is only likely if an additional control rod adjacent to the stuck control rod also fails to insert during a required scram. Even with the postulated additional single failure of an adjacent control rod to insert, sufficient reactivity control remains to reach and maintain MODE 3 conditions (Ref. 8).

B.1

With two or more withdrawn control rods stuck, the plant must be brought to MODE 3 within 12 hours. The occurrence of more than one control rod stuck at a withdrawn position increases the probability that the reactor cannot be shut down if required. Insertion of all insertable control rods eliminates the possibility of an additional failure of a control rod to insert. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

C.1, C.2, and C.3

With one or more control rods inoperable for reasons other than being stuck in the withdrawn position, operation may continue, provided the overall scram reactivity rate is met. To ensure the overall scram reactivity rate is met, the total number of "slow" and inoperable control rods must be immediately verified to be \leq eight. This ensures that the safety analysis assumptions are met (the safety analysis assumes a total of eight control rods are "slow," one is stuck, and another fails to scram. Therefore, ensuring the total number of "slow" and inoperable is \leq eight is conservative since the inoperable control rods are already

(continued)



BASES

ACTIONS

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

fully inserted). In addition, the control rods must be fully inserted within 3 hours and disarmed (electrically or hydraulically) within 4 hours. Inserting a control rod ensures the shutdown and scram capabilities are not adversely affected. The control rod is disarmed to prevent inadvertent withdrawal during subsequent operations. The control rods can be hydraulically disarmed by closing the drive water and exhaust water isolation valves. Electrically, the control rods can be disarmed by disconnecting power from all four directional control valve solenoids. Required Action C.2 is modified by a Note that allows the RWM to be bypassed if required to allow insertion of the inoperable control rods and continued operation. LCO 3.3.2.1 provides additional requirements when the RWM is bypassed to ensure compliance with the CRDA analysis.

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

D.1 and D.2

Out of sequence control rods may increase the potential reactivity worth of a dropped control rod during a CRDA. At $\leq 10\%$ RTP, the generic banked position withdrawal sequence (BPWS) analysis (Ref. 8) requires inserted control rods not in compliance with BPWS to be separated by at least two OPERABLE control rods in all directions, including the diagonal. Therefore, if two or more inoperable control rods are not in compliance with BPWS and not separated by at least two OPERABLE control rods, action must be taken to restore compliance with BPWS or restore the control rods to OPERABLE status. A Note has been added to the Condition to clarify that the Condition is not applicable when $> 10\%$ RTP since the BPWS is not required to be followed under these conditions, as described in the Bases for LCO 3.1.6. The allowed Completion Time of 4 hours is acceptable, considering the low probability of a CRDA occurring.

(continued)

BASES

ACTIONS
(continued)

E.1

In addition to the separation requirements for inoperable control rods, an assumption in the CRDA analysis for Siemens fuel is that no more than three inoperable control rods are allowed in any one BPWS group. - Therefore, with one or more BPWS groups having four or more inoperable control rods, the control rods must be restored to OPERABLE status. Required Action E.1 is modified by a Note indicating that the Condition is not applicable when THERMAL POWER is > 10% RTP since the BPWS is not required to be followed under these conditions, as described in the Bases for LCO 3.1.6. The allowed Completion Time of 4 hours is acceptable, considering the low probability of a CRDA occurring.

F.1

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

SURVEILLANCE
REQUIREMENTS

SR 3.1.3.1

The position of each control rod must be determined, to ensure adequate information on control rod position is available to the operator for determining control rod OPERABILITY and controlling rod patterns. Control rod position may be determined by the use of OPERABLE position indicators, by moving control rods to a position with an

(continued)

BASES.

SURVEILLANCE
REQUIREMENTS

SR 3.1.3.1 (continued)

OPERABLE indicator, or by the use of other appropriate methods. The 24 hour Frequency of this SR is based on operating experience related to expected changes in control rod position and the availability of control rod position indications in the control room.

SR 3.1.3.2 and SR 3.1.3.3

Control rod insertion capability is demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. The control rod may then be returned to its original position. This ensures the control rod is not stuck and is free to insert on a scram signal. These Surveillances are not required when THERMAL POWER is less than or equal the actual LPSP of the RWM since the notch insertions may not be compatible with the requirements of the banked position withdrawal sequence (BPWS) (LCO 3.1.6) and the RWM (LCO 3.3.2.1). The 7 day Frequency of SR 3.1.3.2 is based on operating experience related to the changes in CRD performance and the ease of performing notch testing for fully withdrawn control rods. Partially withdrawn control rods are tested at a 31 day Frequency, based on the potential power reduction required to allow the control rod movement, and considering the large testing sample of SR 3.1.3.2. Furthermore, the 31 day Frequency takes into account operating experience related to changes in CRD performance. At any time, if a control rod is immovable, a determination of that control rod's trippability (OPERABILITY) must be made and appropriate action taken.

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

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.1.3.4

Verifying the scram time for each control rod to notch position 5 is ≤ 7 seconds provides reasonable assurance that the control rod will insert when required during a DBA or transient, thereby completing its shutdown function. This SR is performed in conjunction with the control rod scram time testing of SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," and the functional testing of SDV vent and drain valves in LCO 3.1.8, "Scram Discharge Volume (SDV) Vent and Drain Valves," overlap this Surveillance to provide complete testing of the assumed safety function. The associated Frequencies are acceptable, considering the more frequent testing performed to demonstrate other aspects of control rod OPERABILITY and operating experience, which shows scram times do not significantly change over an operating cycle.

SR 3.1.3.5

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 27, GDC 28, and GDC 29.
2. FSAR, Section 4.3.2.5.

(continued)

BASES

REFERENCES
(continued)

3. FSAR, Section 4.6.1.1.2.5.3.
 4. FSAR, Section 5.2.2.2.3.
 5. FSAR, Section 15.4.1.1.
 6. FSAR, Section 15.F.4.3.
 7. 10 CFR 50.36(c)(2)(ii).
 8. NEDO-21231, "Banked Position Withdrawal Sequence,"
Section 7.2, January 1977.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Control Rod Scram Times

BASES

BACKGROUND

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

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

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the control rod scram function are presented in References 2, 3, 4, 5, and 6. The Design Basis Accident (DBA) and transient analyses assume that all of the control rods scram at a specified insertion rate. The resulting negative scram reactivity forms the basis for the determination of plant thermal limits (e.g., the MCPR). Other distributions of scram times (e.g., several control rods scrambling slower than the average time, with several control rods scrambling faster than the average time) can also provide sufficient scram reactivity. Surveillance of each individual control rod's scram time averaged into each associated two-by-two array, ensures the scram reactivity assumed in the DBA and transient analyses can be met.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The scram function of the CRD System protects the MCPR Safety Limit (SL) (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), and the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), which ensure that no fuel damage will occur if these limits are not exceeded. Above 800 psig, the scram function is designed to insert negative reactivity at a rate fast enough to prevent the actual MCPR from becoming less than the MCPR SL during the analyzed limiting power transient. Below 800 psig, the scram function is assumed to perform during the control rod drop accident (Ref. 6) and, therefore, also provides protection against violating fuel damage limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control"). For the reactor vessel overpressure protection analysis (Ref. 4), the scram function, along with the safety/relief valves, ensure that the peak vessel pressure is maintained within the applicable ASME Code limits.

Control rod scram times satisfy Criterion 3 of Reference 7.

LCO

The scram times specified in Table 3.1.4-1 are required to ensure that the scram reactivity assumed in the DBA and transient analysis is met. The scram times have a margin to allow up to eight of the control rods to have scram times that exceed the specified limits (i.e., "slow" control rods in a two-by-two array that do not meet the average scram time limits) assuming a single stuck control rod (as allowed by LCO 3.1.3, "Control Rod OPERABILITY") and an additional control rod failing to scram per the single failure criterion. The scram times are specified as a function of reactor steam dome pressure to account for the pressure dependence of the scram times. The scram times are specified relative to measurements based on reed switch positions, which provide the control rod position indication. The reed switch closes ("pickup") when the index tube passes a specific location and then opens ("dropout") as the index tube travels upward. Verification of the specified scram times in Table 3.1.4-1 is accomplished through measurement of the "dropout" times.

Table 3.1.4-1 is modified by a Note, which states that control rods with scram times > 7 seconds are considered inoperable as required by SR 3.1.3.4.

(continued)

BASES

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

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

ACTIONS The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each two-by-two array. This is acceptable since the scram times are applicable for a two-by-two array, and the Required Actions for each Condition provide appropriate compensatory action for each two-by-two array not within the average scram time limits. Complying with the Required Actions may allow for continued operation and subsequent two-by-two arrays not within the average scram time limits governed by subsequent Condition entry and application of associated Required Actions.

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

With one or more two-by-two arrays not within the average scram time limits of Table 3.1.4-1, the scram reactivity rate assumptions in the safety analysis may not be met. Therefore, each control rod in the two-by-two array, with a scram time slower than the average scram time limits must be declared "slow" immediately. To ensure the overall scram reactivity rate is met, the total number of "slow" and inoperable control rods must be immediately verified to be \leq eight. This ensures that the safety analysis assumptions are met (the safety analysis has sufficient margin to assume a total of eight control rods are "slow," one is stuck, and another fails to scram. Therefore, ensuring the total

(continued)

BASES

ACTIONS

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

number of "slow" and inoperable is \leq eight is conservative since the inoperable control rods are already fully inserted). To ensure the local scram reactivity rate is met, each "slow" control rod must be immediately verified to meet the "slow" control rod separation criteria. The "slow" control rod separation criteria are met if: a) the "slow" control rod is separated in all directions from each "slow" control rod and each stuck control rod by any combination of two or more fully inserted control rods or OPERABLE, withdrawn control rods that are not "slow"; and b) two or less additional inoperable or "slow" control rods are in the same group as the "slow" control rod.

With the verifications described above performed satisfactorily, the scram reactivity rate assumptions in the safety analysis will be met and continued operation is allowed.

B.1

When any Required Action and associated Completion Time is not met, the rate of negative reactivity insertion during a scram may not be within the assumptions of the safety analyses. Therefore, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

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

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

In addition, the scram times in Table 3.1.4-1 are the average of a two-by-two array. Therefore, a control rod scram time, as determined by the following SRs, must be factored into the average scram time for all applicable two-by-two arrays.

SR 3.1.4.1

The scram reactivity used in DBA and transient analyses is based on assumed control rod scram time. Measurement of the scram times with reactor steam dome pressure ≥ 800 psig demonstrates acceptable scram times for the transients analyzed in References 5 and 6.

Maximum scram insertion times occur at a reactor pressure of approximately 800 psig because of the competing effects of reactor steam dome pressure and stored accumulator energy. Therefore, demonstration of adequate scram times at reactor steam dome pressure ≥ 800 psig ensures that the scram times will be within the specified limits at higher pressures. Limits are specified as a function of reactor pressure to account for the sensitivity of the scram insertion times with pressure and to allow a range of pressures over which scram time testing can be performed. To ensure scram time testing is performed within a reasonable time following a refueling or after a shutdown ≥ 120 days, all control rods are required to be tested before exceeding 40% RTP. This Frequency is acceptable, considering the additional surveillances performed for control rod OPERABILITY, the frequent verification of adequate accumulator pressure, and the required testing of control rods affected by work on control rods or the CRD System.

SR 3.1.4.2

Additional testing of a sample of control rods is required to verify the continued performance of the scram function during the cycle. A representative sample contains at least 10% of the control rods. The sample remains representative if no more than 20% of the control rods in the sample tested are determined to be "slow." If more than 20% of the sample is declared to be "slow" per the criteria in Table 3.1.4-1, additional control rods are tested until this 20% criterion (i.e., 20% of the entire sample size) is satisfied, or until the total number of "slow" control rods (throughout the

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.4.2 (continued)

core, from all Surveillances) exceeds the LCO limit taken. For planned testing, the control rods selected for the sample should be different for each test. Data from inadvertent scrams should be used whenever possible to avoid unnecessary testing at power, even if the control rods with data were previously tested in a sample. The 120 day Frequency is based on operating experience that has shown control rod scram times do not significantly change over an operating cycle. This Frequency is also reasonable, based on the additional Surveillances done on the CRDs at more frequent intervals in accordance with LCO 3.1.3 and LCO 3.1.5, "Control Rod Scram Accumulators."

SR 3.1.4.3

When work that could affect the scram insertion time is performed on a control rod or the CRD System, testing must be done to demonstrate that each affected control rod retains adequate scram performance over the range of applicable reactor pressures from zero to the maximum permissible pressure. The scram testing must be performed once before declaring the control rod OPERABLE. The required scram time testing must demonstrate that the affected control rod is still within acceptable limits. The limits for reactor pressures < 800 psig are found in the Licensee Controlled Specifications Manual (Ref. 8), and are established based on a high probability of meeting the acceptance criteria at reactor pressures \geq 800 psig. Limits for \geq 800 psig are found in Table 3.1.4-1. If testing demonstrates the affected control rod does not meet these limits, but is within the 7-second limit of the Note to Table 3.1.4-1, the control rod can be declared OPERABLE and "slow."

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

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.4.3 (continued)

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

SR 3.1.4.4

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

The Frequency of once prior to exceeding 40% RTP is acceptable because of the capability of testing the control rod at the different conditions and the more frequent surveillances on other aspects of control rod OPERABILITY.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. FSAR, Section 4.3.2.5.
3. FSAR, Section 4.6.1.1.2.5.3.
4. FSAR, Section 5.2.2.2.3.
5. FSAR, Section 15.4.1.1.

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BASES

REFERENCES
(continued)

6. FSAR, Section 15.F.4.3.
 7. 10 CFR 50.36(c)(2)(ii).
 8. Licensee Controlled Specifications Manual.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Control Rod Scram Accumulators

BASES

BACKGROUND

The control rod scram accumulators are part of the Control Rod Drive (CRD) System and are provided to ensure that the control rods scram under varying reactor conditions. The control rod scram accumulators store sufficient energy to fully insert a control rod at any reactor vessel pressure. The accumulator is a hydraulic cylinder with a free floating piston. The piston separates the water used to scram the control rods from the nitrogen, which provides the required energy. The scram accumulators are necessary to scram the control rods within the required insertion times of LCO 3.1.4, "Control Rod Scram Times."

APPLICABLE
SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the control rod scram function are presented in References 1, 2, 3, 4, and 5. The Design Basis Accident (DBA) and transient analyses assume that all of the control rods scram at a specified insertion rate. OPERABILITY of each individual control rod scram accumulator, along with LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.4, ensures that the scram reactivity assumed in the DBA and transient analyses can be met. The existence of an inoperable accumulator may invalidate prior scram time measurements for the associated control rod.

The scram function of the CRD System, and, therefore, the OPERABILITY of the accumulators, protects the MCPR Safety Limit (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), which ensure that no fuel damage will occur if these limits are not exceeded (see Bases for LCO 3.1.4). Also, the scram function at low reactor vessel pressure (i.e., startup conditions) provides protection against violating fuel design limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control").

Control rod scram accumulators satisfy Criterion 3 of Reference 6.

(continued)

D BASES (continued)

LCO The OPERABILITY of the control rod scram accumulators is required to ensure that adequate scram insertion capability exists when needed over the entire range of reactor pressures. The OPERABILITY of the scram accumulators is based on maintaining adequate accumulator pressure.

APPLICABILITY In MODES 1 and 2, the scram function is required for mitigation of DBAs and transients and, therefore, the scram accumulators must be OPERABLE to support the scram function. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod scram accumulator OPERABILITY under these conditions. Requirements for scram accumulators in MODE 5 are contained in LCO 3.9.5, "Control Rod OPERABILITY—Refueling."

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

A.1 and A.2

With one control rod scram accumulator inoperable and the reactor steam dome pressure ≥ 900 psig, the control rod may be declared "slow" (after declaring the average scram time in all two-by-two arrays associated with the control rod with the inoperable accumulator not within the limits of Table 3.1.4-1), since the control rod will still scram at the reactor operating pressure but may not satisfy the required scram times in Table 3.1.4-1. Required Action A.1 is modified by a Note, which clarifies that declaring the control rod "slow" is only applicable if the average scram times of the two-by-two arrays associated with the control rod with the inoperable accumulator are within the limits of Table 3.1.4-1 during the last scram time Surveillance. Otherwise, the control rod may already be considered "slow" and the further degradation of scram performance with an

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

inoperable accumulator could result in excessive scram times. In this event, the associated control rod is declared inoperable (Required Action A.2) and LCO 3.1.3 entered. This would result in requiring the affected control rod to be fully inserted and disarmed, thereby satisfying its intended function in accordance with ACTIONS of LCO 3.1.3.

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

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

With two or more control rod scram accumulators inoperable and reactor steam dome pressure ≥ 900 psig, adequate pressure must be supplied to the charging water header. With inadequate charging water pressure, all of the accumulators could become inoperable, resulting in a potentially severe degradation of the scram performance. Therefore, within 20 minutes from discovery of charging water header pressure < 940 psig concurrent with Condition B, adequate charging water header pressure must be restored. The allowed Completion Time of 20 minutes is considered a reasonable time to place a CRD pump into service to restore the charging header pressure, if required. This Completion Time also recognizes the ability of the reactor pressure alone to fully insert all control rods.

The control rod may be declared "slow" (after declaring the average scram time in all two-by-two arrays associated with the control rod with the inoperable accumulator not within the limits of Table 3.1.4-1), since the control rod will still scram using only reactor pressure, but may not satisfy the times in Table 3.1.4-1. Required Action B.2.1 is modified by a Note indicating that declaring the control rod "slow" is only applicable if the average scram times of the two-by-two arrays associated with the control rod with the inoperable accumulator are within the limits of Table 3.1.4-1 during the last scram time Surveillance. Otherwise, the control rod may already be considered "slow"

(continued)

BASES

ACTIONS

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

and the further degradation of scram performance with an inoperable accumulator could result in excessive scram times. In this event, the associated control rod is declared inoperable (Required Action B.2.2) and LCO 3.1.3 entered. This would result in requiring the affected control rod to be fully inserted and disarmed, thereby satisfying its intended function in accordance with ACTIONS of LCO 3.1.3.

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

C.1 and C.2

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

D.1

The reactor mode switch must be immediately placed in the shutdown position if either Required Action and associated Completion Time associated with loss of the CRD charging pump (Required Actions B.1 and C.1) cannot be met. This

(continued)

BASES

ACTIONS

D.1 (continued)

ensures that all insertable control rods are inserted and that the reactor is in a condition that does not require the active function (i.e., scram) of the control rods. This Required Action is modified by a Note stating that the Required Action is not applicable if all control rods associated with the inoperable scram accumulators are fully inserted, since the function of the control rods has been performed.

SURVEILLANCE
REQUIREMENTS

SR 3.1.5.1

SR 3.1.5.1 requires that the accumulator pressure be checked every 7 days to ensure adequate accumulator pressure exists to provide sufficient scram force. The primary indicator of accumulator OPERABILITY is the accumulator pressure. A minimum accumulator pressure is specified, below which the capability of the accumulator to perform its intended function becomes degraded and the accumulator is considered inoperable. The minimum accumulator pressure of 940 psig is well below the expected pressure of 1400 psig to 1500 psig (Ref. 7). Declaring the accumulator inoperable when the minimum pressure is not maintained ensures that significant degradation in scram times does not occur. The 7 day Frequency has been shown to be acceptable through operating experience and takes into account indications available in the control room.

REFERENCES

1. FSAR, Section 4.3.2.5.
 2. FSAR, Section 4.6.1.1.2.5.3.
 3. FSAR, Section 5.2.2.2.3.
 4. FSAR, Section 15.4.1.1.
 5. FSAR, Section 15.F.4.3.
 6. 10 CFR 50.36(c)(2)(ii).
 7. FSAR, Section 4.6.1.1.2.4.1.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Rod Pattern Control

BASES

BACKGROUND

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

This Specification assures that the control rod patterns are consistent with the assumptions of the CRDA analyses of References 1, 2, and 3.

APPLICABLE
SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the CRDA are summarized in References 1, 2, 3, and 4. CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analysis. The RWM (LCO 3.3.2.1) provides backup to operator control of the withdrawal sequences to ensure that the initial conditions of the CRDA analysis are not violated.

Prevention or mitigation of positive reactivity insertion events is necessary to limit the energy deposition in the fuel, thereby preventing significant fuel damage, which could result in undue release of radioactivity. Since the failure consequences for UO_2 have been shown to be insignificant below fuel energy depositions of 300 cal/gm (Ref. 5), the fuel damage limit of 280 cal/gm provides a margin of safety from significant core damage, which would result in release of radioactivity (Refs. 6 and 7). Generic evaluations (Refs. 8 and 9) of a design basis CRDA (i.e., a CRDA resulting in a peak fuel energy deposition of 280 cal/gm) have shown that if the peak fuel enthalpy remains below 280 cal/gm, then the maximum reactor pressure will be less than the required ASME Code limits (Ref. 10) and the calculated offsite doses will be well within the required limits (Ref. 7).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

Control rod patterns analyzed in Reference 1 follow the banked position withdrawal sequence (BPWS) described in Reference 11. The BPWS is applicable from the condition of all control rods fully inserted to 10% RTP (Ref. 2). For the BPWS, the control rods are required to be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions (e.g., between notches 08 and 12). The banked positions are defined to minimize the maximum incremental control rod worths without being overly restrictive during normal plant operation. The generic BPWS analysis (Ref. 11) also evaluated the effect of fully inserted, inoperable control rods not in compliance with the sequence, to allow a limited number (i.e., eight) and distribution of fully inserted, inoperable control rods.

Rod pattern control satisfies the requirements of Criterion 3 of Reference 12.

LCO

Compliance with the prescribed control rod sequences minimizes the potential consequences of a CRDA by limiting the initial conditions to those consistent with the BPWS. This LCO only applies to OPERABLE control rods. For inoperable control rods required to be inserted, separate requirements are specified in LCO 3.1.3, "Control Rod OPERABILITY," consistent with the allowances for inoperable control rods in the BPWS.

APPLICABILITY

In MODES 1 and 2, when THERMAL POWER is $\leq 10\%$ RTP, the CRDA is a Design Basis Accident (DBA) and, therefore, compliance with the assumptions of the safety analysis is required. When THERMAL POWER is $> 10\%$ RTP, there is no credible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA (Ref. 2). In MODES 3, 4, and 5, since the reactor is shut down and only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will remain subcritical with a single control rod withdrawn.

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

With one or more OPERABLE control rods not in compliance with the prescribed control rod sequence, action may be taken to either correct the control rod pattern or declare the associated control rods inoperable within 8 hours. Noncompliance with the prescribed sequence may be the result of "double notching," drifting from a control rod drive cooling water transient, leaking scram valves, or a power reduction to $\leq 10\%$ RTP before establishing the correct control rod pattern. The number of OPERABLE control rods not in compliance with the prescribed sequence is limited to eight to prevent the operator from attempting to correct a control rod pattern that significantly deviates from the prescribed sequence.

Required Action A.1 is modified by a Note, which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.2.1 requires verification of control rod movement by a second licensed operator (Reactor Operator or Senior Reactor Operator) or by a qualified member of the technical staff (e.g., a qualified shift technical advisor or reactor engineer). This ensures that the control rods will be moved to the correct position. A control rod not in compliance with the prescribed sequence is not considered inoperable except as required by Required Action A.2. The allowed Completion Time of 8 hours is reasonable, considering the restrictions on the number of allowed out of sequence control rods and the low probability of a CRDA occurring during the time the control rods are out of sequence.

B.1 and B.2

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

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

a second licensed operator (Reactor Operator or Senior Reactor Operator) or by a qualified member of the technical staff (e.g., a qualified shift technical advisor or reactor engineer).

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

SURVEILLANCE
REQUIREMENTS

SR 3.1.6.1

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

REFERENCES

1. CE-NPSD-803-P, "WNP-2 Cycle 12 Reload Report," May 1996.
 2. Letter from T.A. Pickens (BWROG) to G.C. Laines (NRC), "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," BWROG-8644, August 15, 1988.
 3. FSAR, Section 15.F.4.3.
 4. CENPD-284-P-A, "Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors: Summary and Qualification," July 1996.
 5. NUREG-0979, "NRC Safety Evaluation Report for GESSAR II BWR/6 Nuclear Island Design, Docket No. 50-447," Section 4.2.1.3.2, April 1983.
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(continued)

BASES

REFERENCES
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6. NUREG-0800, "Standard Review Plan," Section 15.4.9, "Radiological Consequences of Control Rod Drop Accident (BWR)," Revision 2, July 1981.
 7. 10 CFR 100.11, "Determination of Exclusion Area Low Population Zone and Population Center Distance."
 8. NEDO-10527, "Rod Drop Accident Analysis for Large BWRs," (including Supplements 1 and 2), March 1972.
 9. NEDO-21778-A, "Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors," December 1978.
 10. ASME, Boiler and Pressure Vessel Code, Section III.
 11. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.
 12. 10 CFR 50.36(c)(2)(ii).
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND

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

The SLC System consists of a boron solution storage tank, two positive displacement pumps, two explosive valves, which are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged through the high pressure core spray system sparger.

APPLICABLE SAFETY ANALYSES

The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that not enough control rods can be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to compensate for all of the various reactivity effects that could occur during plant operation. To meet this objective, it is necessary to inject, using both SLC pumps, a quantity of boron that produces a concentration of 660 ppm of natural boron in the reactor core, including recirculation loops, at 70°F and normal reactor water level. To allow for potential leakage and imperfect mixing in the reactor system, an additional amount of boron equal to 25% of the amount cited above is added (Ref. 2). An additional 275 ppm is provided to accommodate dilution in the RPV by the residual heat removal shutdown cooling piping. The temperature versus concentration limits in Figure 3.1.7-1 are calculated such that the required concentration is achieved. This quantity of borated solution is the amount that is above the pump suction shutoff level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected.

(continued)



D BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The SLC System satisfies Criterion 4 of Reference 3.

LCO

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

APPLICABILITY

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

ACTIONS

A.1

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

(continued)

BASES

ACTIONS (continued)

B.1

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

C.1

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1 and SR 3.1.7.2

SR 3.1.7.1 and SR 3.1.7.2 are 24 hour Surveillances, verifying certain characteristics of the SLC System (e.g., the volume and temperature of the borated solution in the storage tank), thereby ensuring the SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure the proper borated solution and temperature are maintained. Maintaining a minimum specified borated solution temperature is important in ensuring that the boron remains in solution and does not precipitate out in the storage tank. The 24 hour Frequency of these SRs is based on operating experience that has shown there are relatively slow variations in the measured parameters of volume and temperature.

SR 3.1.7.3 and SR 3.1.7.5

SR 3.1.7.3 verifies the continuity of the explosive charges in the injection valves to ensure proper operation will occur if required. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on operating experience that has demonstrated the reliability of the explosive charge continuity.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.7.3 and SR 3.1.7.5 (continued)

SR 3.1.7.5 verifies each valve in the system is in its correct position, but does not apply to the squib (i.e., explosive) valves. Verifying the correct alignment for manual and power operated valves in the SLC System flow path ensures that the proper flow paths will exist for system operation. A valve is also allowed to be in the nonaccident position, provided it can be aligned to the accident position from the control room, or locally by a dedicated operator at the valve control. This is acceptable since the SLC System is a manually initiated system. This Surveillance does not apply to valves that are locked, sealed, or otherwise secured in position, since they were verified to be in the correct position prior to locking, sealing, or securing. This verification of valve alignment does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct positions. The 31 day Frequency is based on engineering judgment and is consistent with the procedural controls governing valve operation that ensure correct valve positions.

SR 3.1.7.4

This Surveillance requires an examination of the sodium pentaborate solution by using chemical analysis to ensure the proper concentration of boron (measured in weight % sodium pentaborate decahydrate) exists in the storage tank. SR 3.1.7.4 must be performed anytime boron or water is added to the storage tank solution to establish that the boron solution concentration is within the specified limits. This Surveillance must be performed anytime the temperature is restored to within the limits of Figure 3.1.7-1, to ensure no significant boron precipitation occurred. The 31 day Frequency of this Surveillance is appropriate because of the relatively slow variation of boron concentration between surveillances.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.1.7.6

Demonstrating each SLC System pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ 1220 psig ensures that pump performance has not degraded during the fuel cycle. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms one point on the pump design curve, and is indicative of overall performance. Such inservice tests confirm component OPERABILITY and detect incipient failures by indicating abnormal performance. The Frequency of this Surveillance is in accordance with the Inservice Testing Program.

SR 3.1.7.7 and SR 3.1.7.8

These Surveillances ensure that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. The pump and explosive valve tested should be alternated such that both complete flow paths are tested every 48 months, at alternating 24 month intervals. The Surveillance may be performed in separate steps to prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance test when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Demonstrating that all heat traced piping between the boron solution storage tank and the suction valve to the injection pumps is unblocked ensures that there is a functioning flow

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.7.7 and SR 3.1.7.8 (continued)

path for injecting the sodium pentaborate solution. An acceptable method for verifying that the suction piping up to the suction valve is unblocked is to pump from the storage tank to the test tank. - Upon completion of this verification, the pump suction piping must be drained and flushed with demineralized water since the suction piping between the pump suction valve and pump suction is not heat traced. The 24 month Frequency is acceptable since there is a low probability that the subject piping will be blocked due to precipitation of the boron from solution in the heat traced piping. However, if, in performing SR 3.1.7.1, it is determined that the temperature of the solution in the storage tank has fallen below the specified minimum, SR 3.1.7.8 must be performed once within 24 hours after the solution temperature is restored within the limits of Figure 3.1.7-1.

REFERENCES

1. 10 CFR 50.62.
 2. FSAR, Section 9.3.5.3.
 3. 10 CFR 50.36(c)(2)(ii).
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B 3.1 REACTIVITY CONTROL SYSTEMS

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

BASES

BACKGROUND

The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. The SDV consists of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two headers and two instrument volumes, each receiving approximately one half of the control rod drive (CRD) discharges. The two instrument volumes are connected to a common drain line with two valves in series. Each header is connected to a common vent line with two valves in series. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram. The design and functions of the SDV are described in Reference 1.

APPLICABLE
SAFETY ANALYSES

The Design Basis Accident and transient analyses assume all the control rods are capable of scramming. The primary function of the SDV is to limit the amount of reactor coolant discharged during a scram. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to:

- a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR 100 (Ref. 2); and
- b. Open on scram reset to maintain the SDV vent and drain path open so there is sufficient volume to accept the reactor coolant discharged during a scram.

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. For a bounding leakage case, the offsite doses are well within the limits of 10 CFR 100 (Ref. 2) and adequate core cooling is maintained (Ref. 3). The SDV vent and drain valves also allow continuous drainage of the SDV during normal plant

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

operation to ensure the SDV has sufficient capacity to contain the reactor coolant discharge during a full core scram. To automatically ensure this capacity, a reactor scram (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation") is initiated if the SDV water level exceeds a specified setpoint. The setpoint is chosen such that all control rods are inserted before the SDV has insufficient volume to accept a full scram.

SDV vent and drain valves satisfy Criterion 3 of Reference 4.

LCO

The OPERABILITY of all SDV vent and drain valves ensures that, during a scram, the SDV vent and drain valves will close to contain reactor water discharged to the SDV piping. Since the vent and drain lines are provided with two valves in series, the single failure of one valve in the open position will not impair the isolation function of the system. Additionally, the valves are required to be open to ensure that a path is available for the SDV piping to drain freely at other times.

APPLICABILITY

In MODES 1 and 2, scram may be required, and therefore, the SDV vent and drain valves must be OPERABLE. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. During MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Therefore, the SDV vent and drain valves are not required to be OPERABLE in these MODES since the reactor is subcritical and only one rod may be withdrawn and subject to scram.

ACTIONS

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

(continued)

BASES

ACTIONS
(continued)

The ACTIONS Table is modified by a second Note stating that an isolated line may be unisolated under administrative control to allow draining and venting of the SDV. When a line is isolated, the potential for an inadvertent scram due to high SDV level is increased. During these periods, the line may be unisolated under administrative control. This allows any accumulated water in the line to be drained, to preclude a reactor scram on SDV high level. This is acceptable, since administrative controls ensure the valve can be closed quickly, by a dedicated operator, if a scram occurs with the valve open.

A.1

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

B.1

If both valves in a line are inoperable, the line must be isolated to contain the reactor coolant during a scram. The 8 hour Completion Time to isolate the line is based on the low probability of a scram occurring while the line is not isolated and unlikelihood of significant CRD seal leakage.

C.1

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.1.8.1

During normal operation, the SDV vent and drain valves should be in the open position (except when performing SR 3.1.8.2) to allow for drainage of the SDV piping. Verifying that each valve is in the open position ensures that the SDV vent and drain valves will perform their intended function during normal operation. This SR does not require any testing or valve manipulation; rather, it involves verification that the valves are in the correct position. The 31 day Frequency is based on engineering judgment and is consistent with the procedural controls governing valve operation, which ensure correct valve positions. Improper valve position (closed) would not affect the isolation function.

SR 3.1.8.2

During a scram, the SDV vent and drain valves should close to contain the reactor water discharged to the SDV piping. Cycling each valve through its complete range of motion (closed and open) ensures that the valve will function properly during a scram. The 92 day Frequency is based on operating experience and takes into account the level of redundancy in the system design.

SR 3.1.8.3

SR 3.1.8.3 is an integrated test of the SDV vent and drain valves to verify total system performance. After receipt of a simulated or actual scram signal, the closure of the SDV vent and drain valves is verified. The closure time of 30 seconds after a receipt of a scram signal is based on the bounding leakage case evaluated in the accident analysis. Similarly, after receipt of a simulated or actual scram reset signal, the opening of the SDV vent and drain valves is verified. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1 and the scram time testing of control rods in LCO 3.1.3, "Control Rod—OPERABILITY," overlap this Surveillance to provide complete testing of the assumed safety function. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.8.3 (continued)

unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 4.6.1.1.2.4.2.6.
 2. 10 CFR 100.
 3. NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," August 1981.
 4. 10 CFR 50.36(c)(2)(ii).
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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that the fuel design limits identified in References 1, 2, and 3 are not exceeded and that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46. As a result, core geometry will be maintained by minimizing gross fuel cladding failure due to heatup following a design basis LOCA.

APPLICABLE
SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 1 and 2. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs) and normal operations that determine APLHGR limits are presented in FSAR, Chapters 4, 6, 15, and 15.F and in References 1, 2, 3, 4, and 5.

LOCA analyses are performed to ensure that the specified APLHGR limits are adequate to meet the PCT and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. A complete discussion of the analysis code is provided in References 1 and 5. The PCT following a postulated LOCA is a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within an assembly. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by its local peaking factor. A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR. For single recirculation loop operation, References 1 and 2 show that no APLHGR reduction is required.

The APLHGR satisfies Criterion 2 of Reference 6.

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BASES (continued)

LCO The APLHGR limits specified in the COLR are the result of the fuel design and design basis accident analyses. Limits have been provided in the COLR for two recirculation loop operation and single recirculation loop operation. The limits on single recirculation loop operation are provided to allow operation in this condition in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating."

APPLICABILITY The APLHGR limits are primarily derived from fuel design evaluations and LOCA analyses that are assumed to occur at high power levels. Studies and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels \leq 25% RTP, the reactor operates with substantial margin to the APLHGR limits; thus, this LCO is not required.

ACTIONS

A.1

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

B.1

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

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is $\geq 25\%$ RTP and then every 24 hours thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution under normal conditions. The 12 hour allowance after THERMAL POWER $\geq 25\%$ RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. NEDC-32115P, "SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," Revision 2, June 1993.
 2. CE-NPSD-803-P, "WNP-2 Cycle 12 Reload Report," May 1996.
 3. XN-NF-80-19(A), "Exxon Nuclear Methodology for Boiling Water Reactors," Volumes 2, 2A, 2B, and 2C, September 1982.
 4. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel," July 1996.
 5. CE-NPSD-801-P, "WNP-2 LOCA Analysis Report," May 1996.
 6. 10 CFR 50.36(c)(2)(ii).
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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). The operating limit MCPR is established to ensure that no fuel damage results during anticipated operational occurrences (AOOs). Although fuel damage does not necessarily occur if a fuel rod actually experiences boiling transition (Refs. 1 and 2), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

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

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented in the FSAR, Chapters 4, 6, and 15, and References 2, 3, and 4. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (Δ CPR). When the largest Δ CPR is added to the MCPR SL, the required operating limit MCPR is obtained.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state (MCPR_i and MCPR_p, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency as identified in FSAR, Chapters 15 and 15.F.

Flow dependent MCPR limits are determined by steady state methods using the three dimensional BWR simulator code (Ref. 2). MCPR_i curves are provided based on the maximum credible flow runout transient for ASD operation (i.e., runout of both loops).

Power dependent MCPR limits (MCPR_p) are determined by the three dimensional BWR simulator code and the one dimensional transient code (Ref. 2). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scram trips are bypassed, high and low flow MCPR_p operating limits are provided for operating between 25% RTP and the previously mentioned bypass power level.

The MCPR satisfies Criterion 2 of Reference 5.

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The MCPR operating limits are determined by the larger of the MCPR_i and MCPR_p limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a slow recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs.

Statistical analyses indicate that the nominal value of the initial MCPR at 25% RTP is expected to be very large. Studies of the variation of limiting transient behavior have

(continued)

BASES

APPLICABILITY (continued)

been performed over the range of power and flow conditions. These studies encompass the range of key actual plant parameter values important to typically limiting transients. The results of these studies demonstrate that a margin is expected between performance and the MCPR requirements, and that margins increase as power is reduced to 25% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor (IRM) provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 25% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.

ACTIONS

A.1

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

B.1

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

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is \geq 25% RTP and then every 24 hours thereafter. It is compared to the specified limits

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.2.2.1 (continued)

in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER reaches $\geq 25\%$ RTP is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. XN-NF-524(A), "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors," Revision 1, November 1983.
 2. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel," July 1996.
 3. CE-NPSD-802-P, "WNP-2 Cycle 12 Transient Analysis Report," May 1996.
 4. CE-NPSD-803-P, "WNP-2 Cycle 12 Reload Report," May 1996.
 5. 10 CFR 50.36(c)(2)(ii).
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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on the LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs). Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure or inability to cool the fuel does not occur during the anticipated operating conditions identified in References 1 and 2.

APPLICABLE
SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design are presented in References 1, 2, 3, 4, 5, 6, and 7. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20, 50, and 100. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations are:

- a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet; and
- b. Severe overheating of the fuel rod cladding caused by inadequate cooling.

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 8).

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. The analysis also includes allowances for short term transient operation above the operating limit to account for AOOs.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The LHGR satisfies Criterion 2 of Reference 9.

LCO

The LHGR is a basic assumption in the fuel design analysis. The fuel has been designed to operate at rated core power with sufficient design margin to the LHGR calculated to cause a 1% fuel cladding plastic strain. The operating limit to accomplish this objective is specified in the COLR.

APPLICABILITY

The LHGR limits are derived from fuel design analysis that is limiting at high power level conditions. At core thermal power levels < 25% RTP, the reactor is operating with a substantial margin to the LHGR limits and, therefore, the Specification is only required when the reactor is operating at $\geq 25\%$ RTP.

ACTIONS

A.1

If any LHGR exceeds its required limit, an assumption regarding an initial condition of the fuel design analysis is not met. Therefore, prompt action should be taken to restore the LHGR(s) to within its required limits such that the plant is operating within analyzed conditions. The 2 hour Completion Time is normally sufficient to restore the LHGR(s) to within its limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LHGR out of specification.

B.1

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

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.2.3.1

The LHGRs are required to be initially calculated within 12 hours after THERMAL POWER is $\geq 25\%$ RTP and then every 24 hours thereafter. They are compared with the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution under normal conditions. The 12 hour allowance after THERMAL POWER $\geq 25\%$ RTP is achieved is acceptable given the large inherent margin to operating limits at lower power levels.

REFERENCES

1. XN-NF-85-67(A), "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload," September 1986.
2. CENPD-287-P-A, "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors," July 1996.
3. XN-NF-81-21(A), "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Revision 1, January 1982.
4. ANF-89-014(P)(A), Revision 1 and Supplements 1 and 2, "Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels 9x9-IX and 9x9-9X BWR Reload Fuel," October 1991.
5. EMF-95-006, "WNP-2 Cycle 11 Plant Transient Analysis," March 1995.
6. EMF-95-007, "WNP-2 Cycle 11 Reload Analysis," March 1995.
7. FSAR, Chapter 4.
8. NUREG-0800, Section II A.2(g), Revision 2, July 1981.
9. 10 CFR 50.36(c)(2)(ii).

B 3.2 POWER DISTRIBUTION LIMITS

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

BASES

BACKGROUND

The OPERABILITY of the APRMs and their setpoints is an initial condition of all safety analyses that assume rod insertion upon reactor scram. Applicable GDCs are GDC 10, "Reactor Design"; GDC 13, "Instrumentation and Control"; GDC 20, "Protection System Functions"; and GDC 29, "Protection against Anticipated Operation Occurrences" (Ref. 1). This LCO is provided to require the APRM gain or APRM flow biased scram setpoints to be adjusted when operating under conditions of excessive power peaking to maintain acceptable margin to the fuel cladding integrity Safety Limit (SL) and the fuel cladding 1% plastic strain limit.

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

$$\frac{\text{MFLPD}}{\text{F RTP}} > 1,$$

indicating that MFLPD is not decreasing proportionately to the overall power reduction, or conversely, that power peaking is increasing. To maintain margins similar to those at RTP conditions, the excessive power peaking is compensated by gain adjustment on the APRMs or adjustment of the APRM Flow Biased Simulated Thermal Power—High Function Allowable Value (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b). Either of these adjustments has effectively the same result as maintaining MFLPD less than or equal to F RTP and thus maintains RTP margins for APLHGR, MCPR, and LHGR.

The normally selected APRM Flow Biased Simulated Thermal Power—High Function Allowable Value positions the scram above the upper bound of the normal power/flow operating region that has been considered in the design of the fuel rods. The Allowable Value is flow biased with a slope that approximates the upper flow control line. The normally selected APRM Allowable Value is supported by the analyses presented in References 1 and 2 that concentrate on events

(continued)

BASES

BACKGROUND
(continued)

initiated from rated conditions. Design experience has shown that minimum deviations occur within expected margins to operating limits (APLHGR, MCPR, and LHGR), at rated conditions for normal power distributions. However, at other than rated conditions, control rod patterns can be established that significantly reduce the margin to thermal limits. Therefore, the APRM Flow Biased Simulated Thermal Power—High Function Allowable Value may be reduced during operation when the combination of THERMAL POWER and MFLPD indicates an excessive power peaking distribution.

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

APPLICABLE
SAFETY ANALYSES

The acceptance criteria for the APRM gain or setpoint adjustments are that acceptable margins (to APLHGR, MCPR, and LHGR) be maintained to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

FSAR safety analyses (Ref. 2) concentrate on the rated power condition for which the minimum expected margin to the operating limits (APLHGR, MCPR, and LHGR) occurs. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR) limit the initial margins to these operating limits at rated

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

conditions so that specified acceptable fuel design limits are met during transients initiated from rated conditions. At initial power levels less than rated levels, the margin degradation of either the APLHGR, the MCPR, or the LHGR during a transient can be greater than at the rated condition event. This greater margin degradation during the transient is primarily offset by the larger initial margin to limits at the lower than rated power levels. However, power distributions can be hypothesized that would result in reduced margins to the pretransient operating limit. When combined with the increased severity of certain transients at other than rated conditions, the SLs could be approached. At substantially reduced power levels, highly peaked power distributions could be obtained that could reduce thermal margins to the minimum levels required for transient events. To prevent or mitigate such situations, either the APRM gain is adjusted upward by the ratio of the core limiting MFLPD to the FRTP, or the APRM Flow Biased Simulated Thermal Power—High Function Allowable Value is required to be reduced by the ratio of FRTP to the core limiting MFLPD. Either of these adjustments effectively counters the increased severity of some events at other than rated conditions by proportionally increasing the APRM gain or proportionally lowering the APRM Flow Biased Simulated Thermal Power—High Function Allowable Value dependent on the increased peaking that may be encountered.

The APRM gain and setpoint satisfy Criteria 2 and 3 of Reference 3.

LCO

Meeting any one of the following conditions ensures acceptable operating margins for events described above:

- a. Limiting excess power peaking;
- b. Reducing the APRM Flow Biased Simulated Thermal Power—High Function Allowable Value by multiplying the APRM Flow Biased Simulated Thermal Power—High Function Allowable Value by the ratio of FRTP and the core limiting value of MFLPD; or
- c. Increasing the APRM gains to cause the APRM to read greater than 100(%) times MFLPD. This condition is to account for the reduction in margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

(continued)

BASES

LCO
(continued)

MFLPD is the ratio of the limiting LHGR to the LHGR limit for the specific bundle type. For Siemens fuel, MFDLRX is the equivalent of MFLPD. As power is reduced, if the design power distribution is maintained, MFLPD is reduced in proportion to the reduction in power. However, if power peaking increases above the design value, the MFLPD is not reduced in proportion to the reduction in power. Under these conditions, the APRM gain is adjusted upward or the APRM Flow Biased Simulated Thermal Power—High Function Allowable Value is reduced accordingly. When the reactor is operating with peaking less than the design value, it is not necessary to modify the APRM Flow Biased Simulated Thermal Power—High Function Allowable Value. Adjusting the APRM gain or modifying the Flow Biased Simulated Thermal Power—High Function Allowable Value is equivalent to maintaining MFLPD less than or equal to FRTP, as stated in the LCO.

For compliance with LCO Item b (APRM Flow Biased Simulated Thermal Power—High Function Allowable Value modification) or Item c (APRM gain adjustment), only APRMs required to be OPERABLE per LCO 3.3.1.1, Function 2.b, are required to be modified or adjusted. In addition, each APRM may be allowed to have its gain or Allowable Value adjusted or modified independently of other APRMs that are having their gain or Allowable Value adjusted or modified.

APPLICABILITY

The MFLPD limit, APRM gain adjustment, or APRM Flow Biased Simulated Thermal Power—High Function Allowable Value modification is provided to ensure that the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit are not violated during design basis transients. As discussed in the Bases for LCO 3.2.1, LCO 3.2.2, and LCO 3.2.3, sufficient margin to these limits exists below 25% RTP and, therefore, these requirements are only necessary when the plant is operating at $\geq 25\%$ RTP.

ACTIONS

A.1

If the APRM gain or Flow Biased Simulated Thermal Power—High Function Allowable Value is not within limits while the MFLPD has exceeded FRTP, the margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit may be reduced. Therefore, prompt action should be taken to

(continued)

BASES

ACTIONS

A.1 (continued)

restore the MFLPD to within its required limit or make acceptable APRM adjustments such that the plant is operating within the assumed margin of the safety analyses.

The 6 hour Completion Time is normally sufficient to restore either the MFLPD to within limits or the APRM gain or Flow Biased Simulated Thermal Power—High Function Allowable Value to within limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LCO not met.

B.1

If the APRM gain or Flow Biased Simulated Thermal Power—High Function Allowable Value cannot be restored to within their required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2

The MFLPD is required to be calculated and compared to F RTP or APRM gain or Flow Biased Simulated Thermal Power—High Function Allowable Value to ensure that the reactor is operating within the assumptions of the safety analysis. These SRs are required only to determine the MFLPD and, assuming MFLPD is greater than F RTP, the appropriate APRM gain or Flow Biased Simulated Thermal Power—High Function Allowable Value, and is not intended to be a CHANNEL FUNCTIONAL TEST for the APRM gain or APRM Flow Biased Simulated Thermal Power—High Function circuitry. The 24 hour Frequency of SR 3.2.4.1 is chosen to coincide with the determination of other thermal limits, specifically those for the APLHGR and LHGR (LCO 3.2.1 and LCO 3.2.3, respectively). The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2 (continued)

12 hour allowance after THERMAL POWER \geq 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

The 12 hour Frequency of SR 3.2.4.2 is required when MFLPD is greater than FRTP, because more rapid changes in power distribution are typically expected.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 13, GDC 20, and GDC 29.
 2. FSAR, Chapters 15 and 15.F.
 3. 10 CFR 50.36(c)(2)(ii).
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B 3.3 INSTRUMENTATION

B 3.3.1.1 Reactor Protection System (RPS) Instrumentation

BASES

BACKGROUND

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

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

The RPS, as described in the FSAR, Section 7.2 (Ref. 1), includes sensors, relays, bypass circuits, and switches that are necessary to cause initiation of a reactor scram. Functional diversity is provided by monitoring a wide range of dependent and independent parameters. The input parameters to the scram logic are from instrumentation that monitors reactor vessel water level; reactor vessel pressure; neutron flux; main steam line isolation valve position; turbine governor valve (TGV) fast closure, trip oil pressure low; turbine throttle valve (TTV) position; primary containment pressure and scram discharge volume (SDV) water level; as well as reactor mode switch in shutdown position and manual scram signals. There are at least four redundant sensor input signals from each of these parameters. Most channels include equipment (e.g., pressure switches) that compares measured input signals with pre-established setpoints. When a setpoint is exceeded, the channel outputs an RPS trip signal to the trip logic.

The RPS is comprised of two independent trip systems (A and B), with two logic channels in each trip system (logic channels A1 and A2, B1 and B2), as shown in

(continued)

BASES

BACKGROUND
(continued)

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

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

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

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

The actions of the RPS are assumed in the safety analyses of References 2, 3, 4, 5, 6, and 7. The RPS initiates a reactor scram when monitored parameter values exceed the Allowable Values specified by the setpoint methodology and listed in Table 3.3.1.1-1 to preserve the integrity of the fuel cladding, the RCPB, and the containment by minimizing the energy that must be absorbed following a LOCA.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

RPS instrumentation satisfies Criterion 3 of Reference 8. Functions not specifically credited in the accident analysis are retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

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

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

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., differential pressure switch) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for process and all instrument uncertainties, except drift and calibration. The trip setpoints are derived from the analytic limits, corrected for process and all instrument uncertainties, including drift and calibration. The trip setpoints derived in this manner provide adequate protection because all instrumentation uncertainties and process effects are taken into account.

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

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

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

The only MODES specified in Table 3.3.1.1-1 are MODES 1 and 2, and MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies. No RPS Function is required in MODES 3 and 4 since all control rods are fully inserted and the Reactor Mode Switch Shutdown Position control rod withdrawal block (LCO 3.3.2.1) does not allow any control rod to be withdrawn. In MODE 5, control rods withdrawn from a core cell containing no fuel assemblies do not affect the reactivity of the core and, therefore, are not required to have the capability to scram. Provided all other control rods remain inserted, no RPS Function is required. In this condition, the required SDM (LCO 3.1.1) and refuel position one-rod-out interlock (LCO 3.9.2) ensure that no event requiring RPS will occur.

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

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

The IRMs monitor neutron flux levels from the upper range of the source range monitors (SRMs) to the lower range of the average power range monitors (APRMs). The IRMs are capable of generating trip signals that can be used to prevent fuel damage resulting from abnormal operating transients in the intermediate power range. In this power range, the most significant source of reactivity change is due to control rod withdrawal. The IRM provides diverse protection from the rod worth minimizer (RWM), which monitors and controls the movement of control rods at low power. The RWM prevents the withdrawal of an out of sequence control rod during startup that could result in an unacceptable neutron flux excursion (Ref. 9). The IRM provides mitigation of the neutron flux excursion. To demonstrate the capability of the IRM System to mitigate control rod withdrawal events, a generic analysis has been performed (Ref. 10) to evaluate

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

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

the consequences of control rod withdrawal during startup that are mitigated only by the IRM. This analysis, which assumes that one IRM channel in each trip system is bypassed, demonstrates that the IRMs provide protection against local control rod withdrawal errors and results in peak fuel enthalpy below the 170 cal/gm fuel failure threshold criterion.

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

The IRM System is divided into two groups of IRM channels, with four IRM channels inputting to each trip system. The analysis of Reference 9 assumes that one channel in each trip system is bypassed. Therefore, six channels with three channels in each trip system are required for IRM OPERABILITY to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This trip is active in each of the 10 ranges of the IRM, which must be selected by the operator to maintain the neutron flux within the monitored level of an IRM range.

The analysis of Reference 9 has adequate conservatism to permit the IRM Allowable Value specified in the Table.

The Intermediate Range Monitor Neutron Flux—High Function must be OPERABLE during MODE 2 when control rods may be withdrawn and the potential for criticality exists. In MODE 5, when a cell with fuel has its control rod withdrawn, the IRMs provide monitoring for and protection against unexpected reactivity excursions. In MODE 1, the APRM System, the RWM and Rod Block Monitor provide protection against control rod withdrawal error events and the IRMs are not required. The IRMs are automatically bypassed when the mode switch is in the run position.

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BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

1.b. Intermediate Range Monitor—Inop

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

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

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

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

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

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

The APRM channels receive input signals from the local power range monitors (LPRM) within the reactor core, which provide an indication of the power distribution and local power changes. The APRM channels average these LPRM signals to provide a continuous indication of average reactor power from a few percent to greater than RTP. For operation at low power (i.e., MODE 2), the Average Power Range Monitor Neutron Flux—High, Setdown Function is capable of generating a trip signal that prevents fuel damage resulting from abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux—High, Setdown Function will provide a secondary scram to the Intermediate Range Monitor Neutron Flux—High Function because of the relative setpoints. With

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

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

the IRMs at Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux—High, Setdown Function will provide the primary trip signal for a core-wide increase in power.

No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux—High, Setdown Function. However, this Function indirectly ensures that, before the reactor mode switch is placed in the run position, reactor power does not exceed 25% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < 25% RTP.

The APRM System is divided into two groups of channels with three APRM channel inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Neutron Flux—High, Setdown, with two channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 14 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.

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

The Average Power Range Monitor Neutron Flux—High, Setdown Function must be OPERABLE during MODE 2 when control rods may be withdrawn. In MODE 1, the Average Power Range Monitor Neutron Flux—High Function provides protection against reactivity transients and the RWM and Rod Block Monitor protect against control rod withdrawal error events.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
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(continued)

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

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

The APRM System is divided into two groups of channels with three APRM inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one Average Power Range Monitor channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Flow Biased Simulated Thermal Power—High, with two channels in each trip system arranged in one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 14 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

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

Each APRM channel receives two independent, redundant flow signals representative of total recirculation driving flow. The total recirculation driving flow signals are generated by four flow units, two of which supply signals to the trip system A APRMs, while the other two supply signals to the trip system B APRMs. Each flow unit signal is provided by summing the flow signals from the two recirculation loops. These redundant flow signals are sensed from four pairs of elbow taps, two in each recirculation loop. To obtain the most conservative reference signals under single failure conditions, the total flow signals from the two flow units (associated with a trip system as described above) are routed to a low auction circuit associated with each APRM. Each APRM's circuit selects the lower of the two flow unit signals for use as the reference for that particular APRM. Each required Average Power Range Monitor Flow Biased Simulated Thermal Power—High channel only requires an input from one OPERABLE flow unit, since the individual APRM channel will perform the intended function with only one OPERABLE flow unit input. However, in order to maintain single failure criteria as described above for the Function, at least one required Average Power Range Monitor Flow Biased Simulated Thermal Power—High channel in each trip system must be capable of maintaining an OPERABLE flow unit signal in the event of a failure of an auction circuit, or a flow unit, in the associated trip system (e.g., if a flow unit is inoperable, one of the two required Average Power Range Monitor Flow Biased Simulated Thermal Power—High channels in the associated trip system must be considered inoperable).

No specific safety analyses take direct credit for the Average Power Range Monitor Flow Biased Simulated Thermal Power—High Function. Originally, the clamped Allowable Value was based on analyses that took credit for the Average Power Range Monitor Flow Biased Simulated Thermal Power—High Function for the mitigation of the loss of feedwater heater event. However, the current methodology for this event is based on a steady state analysis that allows power to increase beyond the clamped Allowable Value. Therefore, applying a clamp is conservative. The THERMAL POWER time constant of ≤ 7 seconds is based on the fuel heat transfer dynamics and provides a signal that is proportional to the THERMAL POWER.

(continued)

BASES

APPLICABLE
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APPLICABILITY

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

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

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

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases. The Average Power Range Monitor Fixed Neutron Flux—High Function is capable of generating a trip signal to prevent fuel damage or excessive Reactor Coolant System (RCS) pressure. For the overpressurization protection analyses of References 2, 3, and 6, the Average Power Range Monitor Fixed Neutron Flux—High Function is assumed to terminate the main steam isolation valve (MSIV) closure event and, along with the safety/relief valves (SRVs), limits the peak reactor pressure vessel (RPV) pressure to less than the ASME Code limits. The control rod drop accident (CRDA) analysis (Ref. 11) takes credit for the Average Power Range Monitor Fixed Neutron Flux—High Function to terminate the CRDA.

The APRM System is divided into two groups of channels with three APRM channels inputting to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Fixed Neutron Flux—High with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 14 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.

(continued)

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

The Allowable Value is based on the Analytical Limit assumed in the CRDA analyses.

The Average Power Range Monitor Fixed Neutron Flux—High Function is required to be OPERABLE in MODE 1 where the potential consequences of the analyzed transients could result in the SLs (e.g., MCPR and RCS pressure) being exceeded. The Average Power Range Monitor Fixed Neutron Flux—High Function is assumed in the CRDA analysis (Ref. 11) that is applicable in MODE 2. However, the Average Power Range Monitor Neutron Flux—High, Setdown Function conservatively bounds the assumed trip and, together with the assumed IRM trips, provides adequate protection. Therefore, the Average Power Monitor Fixed Neutron Flux—High Function is not required in MODE 2.

2.d. Average Power Range Monitor—Inop

This signal provides assurance that a minimum number of APRMs are OPERABLE. Anytime an APRM mode switch is moved to any position other than "Operate," an APRM module is unplugged, or the APRM has too few LPRM inputs (< 14), an inoperative trip signal will be received by the RPS, unless the APRM is bypassed. Since only one APRM in each trip system may be bypassed, only one APRM in each trip system may be inoperable without resulting in an RPS trip signal. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

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

There is no Allowable Value for this Function.

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

(continued)

BASES

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(continued)

3. Reactor Vessel Steam Dome Pressure—High

An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes the neutron flux and THERMAL POWER transferred to the reactor coolant to increase, which could challenge the integrity of the fuel cladding and the RCPB. No specific safety analysis takes direct credit for this Function. However, the Reactor Vessel Steam Dome Pressure—High Function initiates a scram for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power. For the overpressurization protection analyses of References 2, 3, and 6, the reactor scram (the analyses conservatively assume scram on the Average Power Range Monitor Fixed Neutron Flux—High signal, not the Reactor Vessel Steam Dome Pressure—High signal), along with the SRVs, limits the peak RPV pressure to less than the ASME Section III Code limits.

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

Four channels of Reactor Vessel Steam Dome Pressure—High Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. The Function is required to be OPERABLE in MODES 1 and 2 since the RCS is pressurized and the potential for pressure increase exists.

4. Reactor Vessel Water Level—Low, Level 3

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, a reactor scram is initiated at Level 3 to substantially reduce the heat generated in the fuel from fission. The Reactor Vessel Water Level—Low, Level 3 Function is assumed in the analysis of the recirculation line break (Ref. 4). The reactor scram reduces the amount of energy required to be

(continued)

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

absorbed and, along with the actions of the Emergency Core Cooling Systems (ECCS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

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

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

The Reactor Vessel Water Level—Low, Level 3 Allowable Value is selected to ensure that, for transients involving loss of all normal feedwater flow, initiation of the low pressure ECCS at RPV Water Level 1 will not be required.

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

5. Main Steam Isolation Valve—Closure

MSIV closure results in loss of the main turbine and the condenser as a heat sink for the Nuclear Steam Supply System and indicates a need to shut down the reactor to reduce heat generation. Therefore, a reactor scram is initiated on a Main Steam Isolation Valve—Closure signal before the MSIVs are completely closed in anticipation of the complete loss of the normal heat sink and subsequent overpressurization transient. However, for the overpressurization protection analyses of References 2, 3, and 6, the Average Power Range Monitor Fixed Neutron Flux—High Function, along with the SRVs, limits the peak RPV pressure to less than the ASME Code limits. That is, the direct scram on position switches for MSIV closure events is not assumed in the overpressurization analysis. Additionally, MSIV closure is

(continued)



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5. Main Steam Isolation Valve—Closure (continued)

assumed in the transients analyzed in Reference 5 (e.g., low steam line pressure, manual closure of MSIVs, high steam line flow). The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the ECCS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

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

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

Sixteen channels of the Main Steam Isolation Valve—Closure Function with eight channels in each trip system are required to be OPERABLE to ensure that no single instrument failure will preclude the scram from this Function on a valid signal. This Function is only required in MODE 1 since, with the MSIVs open and the heat generation rate high, a pressurization transient can occur if the MSIVs close. In MODE 2, the heat generation rate is low enough so that the other diverse RPS functions provide sufficient protection.

6. Primary Containment Pressure—High

High pressure in the drywell could indicate a break in the RCPB. A reactor scram is initiated to minimize the possibility of fuel damage and to reduce the amount of energy being added to the coolant and the drywell. The Primary Containment Pressure—High Function is a secondary scram signal to Reactor Vessel Water Level—Low, Level 3 for

(continued)

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6. Primary Containment Pressure—High (continued)

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

The reactor scram reduces the amount of energy required to be absorbed and along with the actions of the ECCS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

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

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

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

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

SDV water level is measured by two diverse methods. The level in each of the two SDVs is measured by two float type level switches and two transmitters and trip units for a total of eight level signals. The outputs of these devices are arranged so that there is a signal from a level switch

(continued)

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

and a transmitter and trip unit to each RPS logic channel. The level measurement instrumentation satisfies the recommendations of Reference 12.

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

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

8. Turbine Throttle Valve—Closure

Closure of the TTVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated at the start of TTV closure in anticipation of the transients that would result from the closure of these valves. The Turbine Throttle Valve—Closure Function is the primary scram signal for the turbine trip event analyzed in Reference 5. For this event, the reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the End of Cycle Recirculation Pump Trip (EOC-RPT) System, ensures that the MCPR SL is not exceeded.

Turbine Throttle Valve—Closure signals are initiated by valve stem position switches at each throttle valve. Two switches are associated with each throttle valve. One of the two provides input to RPS trip system A; the other, to RPS trip system B. Thus, each RPS trip system receives an input from four Turbine Throttle Valve—Closure channels, each consisting of one valve stem position switch. The

(continued)

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8. Turbine Throttle Valve—Closure (continued)

Logic for the Turbine Throttle Valve—Closure Function is such that three or more TTVs must close to produce a scram. In addition, certain combinations of two valves closed will result in a half-scram.

This Function must be enabled at THERMAL POWER \geq 30% RTP. This is accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this Function.

The Turbine Throttle Valve—Closure Allowable Value is selected to detect imminent TTV closure thereby reducing the severity of the subsequent pressure transient.

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

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

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

Turbine Governor Valve Fast Closure, Trip Oil Pressure—Low signals are initiated by the digital-electro hydraulic fluid pressure at each governor valve. There is one pressure

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9. Turbine Governor Valve Fast Closure, Trip Oil
Pressure—Low (continued)

switch associated with each governor valve, the signal from each switch being assigned to a separate RPS logic channel. This Function must be enabled at THERMAL POWER \geq 30% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this Function. The basis for the setpoint of this automatic bypass is identical to that described for the Turbine Throttle Valve—Closure Function.

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

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

10. Reactor Mode Switch—Shutdown Position

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

The reactor mode switch is a single switch with four channels (one from each of the four independent banks of contacts), each of which inputs into one of the RPS logic channels.

(continued)

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APPLICABLE
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LCO, and
APPLICABILITY

10. Reactor Mode Switch—Shutdown Position (continued)

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

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

11. Manual Scram

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

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

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

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

ACTIONS

A Note has been provided to modify the ACTIONS related to RPS instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered,

(continued)



BASES

ACTIONS
(continued)

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

A.1 and A.2

Because of the diversity of sensors available to provide trip signals and the redundancy of the RPS design, an allowable out of service time of 12 hours has been shown to be acceptable (Ref. 13) to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided the associated Functions inoperable channel is in one trip system and the Function still maintains RPS trip capability (refer to Required Actions B.1, B.2, and C.1 Bases.) If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel or the associated trip system must be placed in the tripped condition per Required Actions A.1 and A.2. Placing the inoperable channel in trip (or the associated trip system in trip) would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel (or trip system) in trip (e.g., as in the case where placing the inoperable channel in trip would result in a full scram), Condition D must be entered and its Required Action taken.

B.1 and B.2

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

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

Required Actions B.1 and B.2 limit the time the RPS scram logic for any Function would not accommodate single failure in both trip systems (e.g., one-out-of-one and one-out-of-one arrangement for a typical four channel Function). The reduced reliability of this logic arrangement was not evaluated in Reference 13 for the 12 hour Completion Time. Within the 6 hour allowance, the associated Function will have all required channels either OPERABLE or in trip (or in any combination) in one trip system.

Completing one of these Required Actions restores RPS to an equivalent reliability level as that evaluated in Reference 13, which justified a 12 hour allowable out of service time as presented in Condition A. The trip system in the more degraded state should be placed in trip or, alternatively, all the inoperable channels in that trip system should be placed in trip (e.g., a trip system with two inoperable channels could be in a more degraded state than a trip system with four inoperable channels, if the two inoperable channels are in the same Function while the four inoperable channels are all in different Functions). The decision as to which trip system is in the more degraded state should be based on prudent judgment and current plant conditions (i.e., what MODE the plant is in). If this action would result in a scram or recirculation pump trip, it is permissible to place the other trip system or its inoperable channels in trip.

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

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

(continued)

BASES

ACTIONS
(continued)

C.1

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

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

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

D.1

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

(continued)



BASES

ACTIONS (continued)

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

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

H.1

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

SURVEILLANCE REQUIREMENTS

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

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

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

the RPS reliability analysis (Ref. 13) assumption of the average time required to perform channel surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the RPS will trip when necessary.

SR 3.3.1.1.1

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

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

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

SR 3.3.1.1.2

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoint," allows the APRMs to be reading greater than actual THERMAL POWER to compensate for localized power peaking. When this adjustment is made, the requirement for the APRMs to

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.1.2 (continued)

indicate within 2% RTP of calculated power is modified to require the APRMs to indicate within 2% RTP of calculated MFLPD. The Frequency of once per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading between performances of SR 3.3.1.1.7.

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

SR 3.3.1.1.3

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

As noted, SR 3.3.1.1.3 is not required to be performed when entering MODE 2 from MODE 1 since testing of the MODE 2 required IRM and APRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This allows entry into MODE 2 if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after entering MODE 2 from MODE 1. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

A Frequency of 7 days provides an acceptable level of system average unavailability over the Frequency interval and is based on reliability analysis (Ref. 13).

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.1.1.4

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended Function. A Frequency of 7 days provides an acceptable level of system average availability over the Frequency and is based on the reliability analysis of Reference 13. (The Manual Scram Functions CHANNEL FUNCTIONAL TEST Frequency was credited in the analysis to extend many automatic scram Functions Frequencies.)

SR 3.3.1.1.5 and SR 3.3.1.1.6

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

The overlap between SRMs and IRMs is required to be demonstrated to ensure that reactor power will not be increased into a region without adequate neutron flux indication. This is required prior to withdrawing SRMs from the fully inserted position since indication is being transitioned from the SRMs to the IRMs.

The overlap between IRMs and APRMs is of concern when reducing power into the IRM range. On power increases, the system design will prevent further increases (initiate a rod block) if adequate overlap is not maintained. Overlap between IRMs and APRMs exists when sufficient IRMs and APRMs concurrently have onscale readings such that the transition between MODE 1 and MODE 2 can be made without either APRM downscale rod block, or IRM upscale rod block. Overlap between SRMs and IRMs similarly exists when, prior to withdrawing the SRMs from the fully inserted position, IRMs are above mid-scale on range 1 before SRMs have reached the upscale rod block. The IRM/APRM and SRM/IRM overlaps are also acceptable if a 1/2 decade overlap exists.

As noted, SR 3.3.1.1.6 is only required to be met during entry into MODE 2 from MODE 1. That is, after the overlap requirement has been met and indication has transitioned to the IRMs, maintaining overlap is not required (APRMs may be reading downscale once in MODE 2).

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BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.1.5 and SR 3.3.1.1.6 (continued)

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

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

SR 3.3.1.1.7

LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP) System. This establishes the relative local flux profile for appropriate representative input to the APRM System. The 1130 MWD/T Frequency is based on operating experience with LPRM sensitivity changes.

SR 3.3.1.1.8 and SR 3.3.1.1.13

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

For Function 2.b, the CHANNEL FUNCTIONAL TEST includes the adjustment of the APRM channel to conform to a calibrated flow signal. This ensures that the total loop drive flow signals from the flow unit used to vary the setpoint are appropriately compared to an injection test flow signal to verify the flow signal trip setpoint and, therefore, the APRM Function accurately reflects the required setpoint as a function of flow. If the flow signal trip setpoint is not within the appropriate limit, the APRMs that receive an input from the inoperable flow unit must be declared inoperable.

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BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.1.8 and SR 3.3.1.1.13 (continued)

The 92 day Frequency of SR 3.3.1.1.8 is based on the reliability analysis of Reference 13. The 24 month Frequency of SR 3.3.1.1.13 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.

SR 3.3.1.1.9 and SR 3.3.1.1.10

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

Note 1 states that neutron detectors are excluded from CHANNEL CALIBRATION because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the 1130 MWD/T LPRM calibration against the TIPS (SR 3.3.1.1.7). A second Note is provided that requires the APRM and IRM SRs to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM and IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or moveable links. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR. The Frequency of SR 3.3.1.1.9 is based upon the assumption of a 184 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.10 is based on the assumption of a 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

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BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.1.1.11

The Average Power Range Monitor Flow Biased Simulated Thermal Power—High Function uses an electronic filter circuit to generate a signal proportional to the core THERMAL POWER from the APRM neutron flux signal. This filter circuit is representative of the fuel heat transfer dynamics that produce the relationship between the neutron flux and the core THERMAL POWER. The filter time constant must be verified to ensure that the channel is accurately reflecting the desired parameter.

The Frequency of 18 months is based on engineering judgment and reliability of the components.

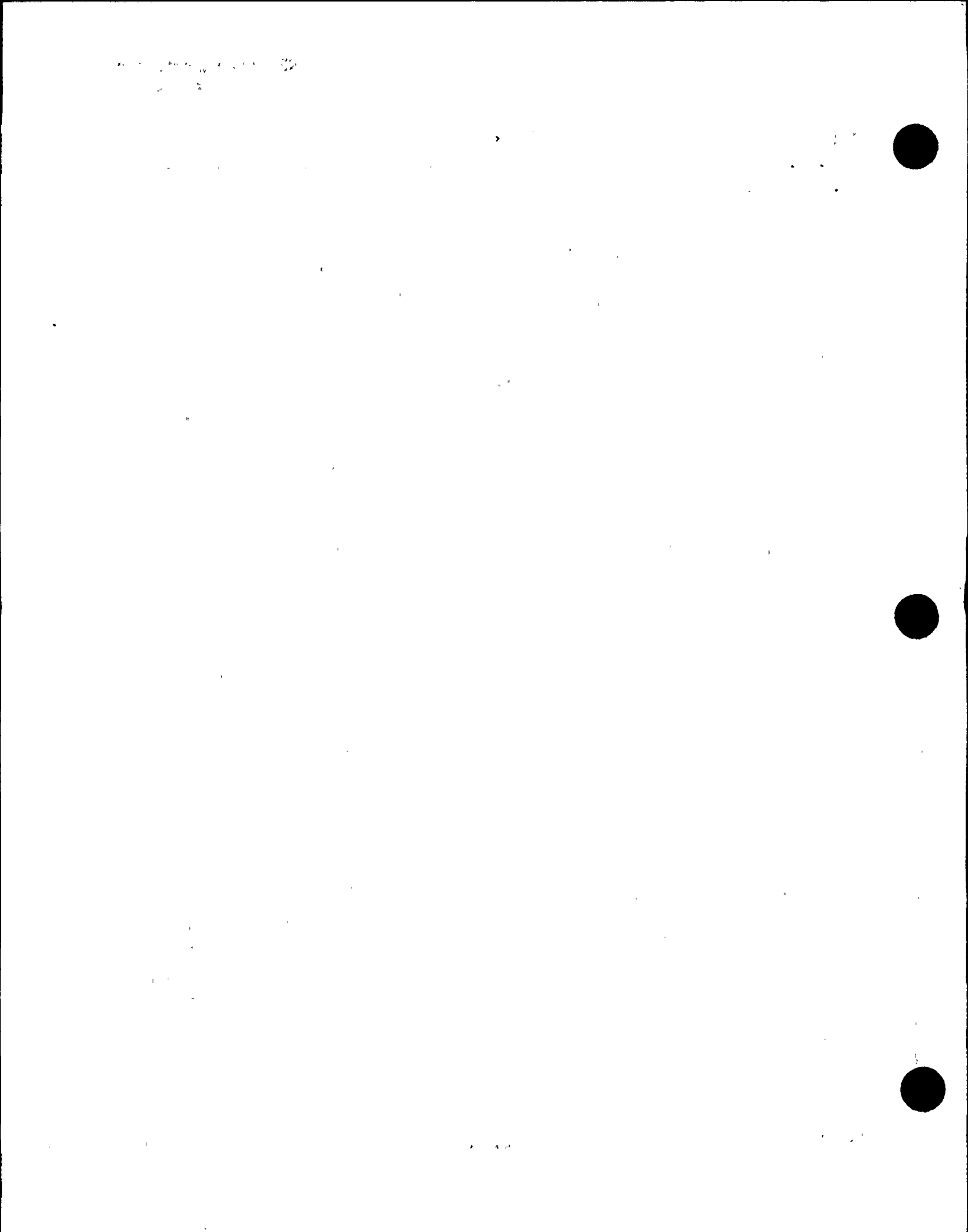
SR 3.3.1.1.12

This SR ensures that scrams initiated from the Turbine Throttle Valve—Closure and Turbine Governor Valve Fast Closure, Trip Oil Pressure—Low Functions will not be inadvertently bypassed when THERMAL POWER is $\geq 30\%$ RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodology are incorporated into the Allowable Value and the actual setpoint. Because main turbine bypass flow can affect this setpoint nonconservatively (THERMAL POWER is derived from turbine first stage pressure), the main turbine bypass valves must remain closed during an in-service calibration at THERMAL POWER $\geq 30\%$ RTP to ensure that the calibration is valid.

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

The Frequency of 18 months is based on engineering judgment and reliability of the components.

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BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.1.1.14

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The functional testing of control rods, in LCO 3.1.3, "Control Rod OPERABILITY," and SDV vent and drain valves, in LCO 3.1.8, "Scram Discharge Volume (SDV) Vent and Drain Valves," overlaps this Surveillance to provide complete testing of the assumed safety function.

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

SR 3.3.1.1.15

This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. The RPS RESPONSE TIME acceptance criteria are included in Reference 14.

As noted (Note 1), neutron detectors for Function 2 are excluded from RPS RESPONSE TIME testing because the principles of detector operation virtually ensure an instantaneous response time.

RPS RESPONSE TIME tests are conducted on a 24 month STAGGERED TEST BASIS. Note 2 requires STAGGERED TEST BASIS Frequency to be determined based on 4 channels per trip system, in lieu of the 8 channels specified in Table 3.3.1.1-1 for the MSIV Closure Function. This Frequency is based on the logic interrelationships of the various channels required to produce an RPS scram signal. Therefore, staggered testing results in response time verification of these devices every 24 months. The 24 month Frequency is consistent with the typical industry refueling cycle and is based upon plant operating experience, which shows that random failures of instrumentation components causing serious time degradation, but not channel failure, are infrequent.

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BASES (continued)

- REFERENCES
1. FSAR, Section 7.2.
 2. FSAR, Section 15.2.4.
 3. WNP-2 Calculation NE-02-94-66, Revision 0, November 13, 1995.
 4. FSAR, Section 6.3.3.
 5. FSAR, Chapters 15 and 15.F.
 6. CE-NPSD-802-P, "WNP-2 Cycle 12 Transient Analysis Report," May 1996.
 7. CE-NPSD-803-P, "WNP-2 Cycle 12 Reload Report," May 1996.
 8. 10 CFR 50.36(c)(2)(ii).
 9. FSAR, Section 15.4.1.
 10. NEDO-23842, "Continuous Control Rod Withdrawal in the Startup Range," April 18, 1978.
 11. FSAR, Section 15.F.4.3.
 12. Letter, P. Check (NRC) to G. Lainas (NRC), "BWR Scram Discharge System Safety Evaluation," December 1, 1980.
 13. NEDO-30851-P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988.
 14. Licensee Controlled Specifications Manual.
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B 3.3 INSTRUMENTATION

B 3.3.1.2 Source Range Monitor (SRM) Instrumentation

BASES

BACKGROUND

The SRMs provide the operator with information relative to the neutron level at very low flux levels in the core. As such, the SRM indication is used by the operator to monitor the approach to criticality and to determine when criticality is achieved. The SRMs are maintained fully inserted until the count rate is greater than a minimum allowed count rate (a control rod block is set at this condition). After SRM to intermediate range monitor (IRM) overlap is demonstrated (as required by SR 3.3.1.1.5) and the IRMs are on Range 3, the SRMs are normally fully withdrawn from the core.

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

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

APPLICABLE SAFETY ANALYSES

Prevention and mitigation of prompt reactivity excursions during refueling and low power operation are provided by LCO 3.9.1, "Refueling Equipment Interlocks"; LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"; LCO 3.3.1.1, "Reactor Protection

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

System (RPS) Instrumentation," Intermediate Range Monitor (IRM) Neutron Flux High and Average Power Range Monitor (APRM) Neutron Flux—High, Setdown Functions; and LCO 3.3.2.1, "Control Rod Block Instrumentation."

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

LCO

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

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

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

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

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1. 2. 3. 4. 5. 6. 7. 8. 9. 10. 11. 12. 13. 14. 15. 16. 17. 18. 19. 20. 21. 22. 23. 24. 25. 26. 27. 28. 29. 30. 31. 32. 33. 34. 35. 36. 37. 38. 39. 40. 41. 42. 43. 44. 45. 46. 47. 48. 49. 50. 51. 52. 53. 54. 55. 56. 57. 58. 59. 60. 61. 62. 63. 64. 65. 66. 67. 68. 69. 70. 71. 72. 73. 74. 75. 76. 77. 78. 79. 80. 81. 82. 83. 84. 85. 86. 87. 88. 89. 90. 91. 92. 93. 94. 95. 96. 97. 98. 99. 100. 101. 102. 103. 104. 105. 106. 107. 108. 109. 110. 111. 112. 113. 114. 115. 116. 117. 118. 119. 120. 121. 122. 123. 124. 125. 126. 127. 128. 129. 130. 131. 132. 133. 134. 135. 136. 137. 138. 139. 140. 141. 142. 143. 144. 145. 146. 147. 148. 149. 150. 151. 152. 153. 154. 155. 156. 157. 158. 159. 160. 161. 162. 163. 164. 165. 166. 167. 168. 169. 170. 171. 172. 173. 174. 175. 176. 177. 178. 179. 180. 181. 182. 183. 184. 185. 186. 187. 188. 189. 190. 191. 192. 193. 194. 195. 196. 197. 198. 199. 200. 201. 202. 203. 204. 205. 206. 207. 208. 209. 210. 211. 212. 213. 214. 215. 216. 217. 218. 219. 220. 221. 222. 223. 224. 225. 226. 227. 228. 229. 230. 231. 232. 233. 234. 235. 236. 237. 238. 239. 240. 241. 242. 243. 244. 245. 246. 247. 248. 249. 250. 251. 252. 253. 254. 255. 256. 257. 258. 259. 260. 261. 262. 263. 264. 265. 266. 267. 268. 269. 270. 271. 272. 273. 274. 275. 276. 277. 278. 279. 280. 281. 282. 283. 284. 285. 286. 287. 288. 289. 290. 291. 292. 293. 294. 295. 296. 297. 298. 299. 300. 301. 302. 303. 304. 305. 306. 307. 308. 309. 310. 311. 312. 313. 314. 315. 316. 317. 318. 319. 320. 321. 322. 323. 324. 325. 326. 327. 328. 329. 330. 331. 332. 333. 334. 335. 336. 337. 338. 339. 340. 341. 342. 343. 344. 345. 346. 347. 348. 349. 350. 351. 352. 353. 354. 355. 356. 357. 358. 359. 360. 361. 362. 363. 364. 365. 366. 367. 368. 369. 370. 371. 372. 373. 374. 375. 376. 377. 378. 379. 380. 381. 382. 383. 384. 385. 386. 387. 388. 389. 390. 391. 392. 393. 394. 395. 396. 397. 398. 399. 400. 401. 402. 403. 404. 405. 406. 407. 408. 409. 410. 411. 412. 413. 414. 415. 416. 417. 418. 419. 420. 421. 422. 423. 424. 425. 426. 427. 428. 429. 430. 431. 432. 433. 434. 435. 436. 437. 438. 439. 440. 441. 442. 443. 444. 445. 446. 447. 448. 449. 450. 451. 452. 453. 454. 455. 456. 457. 458. 459. 460. 461. 462. 463. 464. 465. 466. 467. 468. 469. 470. 471. 472. 473. 474. 475. 476. 477. 478. 479. 480. 481. 482. 483. 484. 485. 486. 487. 488. 489. 490. 491. 492. 493. 494. 495. 496. 497. 498. 499. 500. 501. 502. 503. 504. 505. 506. 507. 508. 509. 510. 511. 512. 513. 514. 515. 516. 517. 518. 519. 520. 521. 522. 523. 524. 525. 526. 527. 528. 529. 530. 531. 532. 533. 534. 535. 536. 537. 538. 539. 540. 541. 542. 543. 544. 545. 546. 547. 548. 549. 550. 551. 552. 553. 554. 555. 556. 557. 558. 559. 560. 561. 562. 563. 564. 565. 566. 567. 568. 569. 570. 571. 572. 573. 574. 575. 576. 577. 578. 579. 580. 581. 582. 583. 584. 585. 586. 587. 588. 589. 590. 591. 592. 593. 594. 595. 596. 597. 598. 599. 600. 601. 602. 603. 604. 605. 606. 607. 608. 609. 610. 611. 612. 613. 614. 615. 616. 617. 618. 619. 620. 621. 622. 623. 624. 625. 626. 627. 628. 629. 630. 631. 632. 633. 634. 635. 636. 637. 638. 639. 640. 641. 642. 643. 644. 645. 646. 647. 648. 649. 650. 651. 652. 653. 654. 655. 656. 657. 658. 659. 660. 661. 662. 663. 664. 665. 666. 667. 668. 669. 670. 671. 672. 673. 674. 675. 676. 677. 678. 679. 680. 681. 682. 683. 684. 685. 686. 687. 688. 689. 690. 691. 692. 693. 694. 695. 696. 697. 698. 699. 700. 701. 702. 703. 704. 705. 706. 707. 708. 709. 710. 711. 712. 713. 714. 715. 716. 717. 718. 719. 720. 721. 722. 723. 724. 725. 726. 727. 728. 729. 730. 731. 732. 733. 734. 735. 736. 737. 738. 739. 740. 741. 742. 743. 744. 745. 746. 747. 748. 749. 750. 751. 752. 753. 754. 755. 756. 757. 758. 759. 760. 761. 762. 763. 764. 765. 766. 767. 768. 769. 770. 771. 772. 773. 774. 775. 776. 777. 778. 779. 780. 781. 782. 783. 784. 785. 786. 787. 788. 789. 790. 791. 792. 793. 794. 795. 796. 797. 798. 799. 800. 801. 802. 803. 804. 805. 806. 807. 808. 809. 810. 811. 812. 813. 814. 815. 816. 817. 818. 819. 820. 821. 822. 823. 824. 825. 826. 827. 828. 829. 830. 831. 832. 833. 834. 835. 836. 837. 838. 839. 840. 84

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D BASES

LCO
(continued)

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

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

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

APPLICABILITY

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

ACTIONS

A.1 and B.1

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

Providing that at least one SRM remains OPERABLE, Required Action A.1 allows 4 hours to restore the required SRMs to OPERABLE status. This is a reasonable time since there is adequate capability remaining to monitor the core, limited risk of an event during this time, and sufficient time to take corrective actions to restore the required SRMs to

(continued)

D BASES

ACTIONS

A.1 and B.1 (continued)

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

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

C.1

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

D.1 and D.2

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

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BASES

ACTIONS

D.1 and D.2 (continued)

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

E.1 and E.2

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

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

SURVEILLANCE
REQUIREMENTS

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

SR 3.3.1.2.1 and SR 3.3.1.2.3

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to the same parameter indicated on other similar channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or

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D BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.2.1 and SR 3.3.1.2.3 (continued)

something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

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

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

SR 3.3.1.2.2

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

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1. 2. 3. 4. 5. 6. 7. 8. 9. 10.

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1. 2. 3. 4. 5. 6. 7. 8. 9. 10.

BASES

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(continued)

SR 3.3.1.2.4

This Surveillance consists of a verification of the SRM instrument readout to ensure that the SRM reading is greater than a specified minimum count rate with the detector full in. This ensures that the detectors are indicating count rates indicative of neutron flux levels within the core. With few fuel assemblies loaded, the SRMs will not have a high enough count rate to satisfy the SR. Therefore, allowances are made for loading sufficient "source" material, in the form of irradiated fuel assemblies, to establish the minimum count rate.

To accomplish this, the SR is modified by a Note that states that the count rate is not required to be met on an SRM that has less than or equal to four fuel assemblies adjacent to the SRM and no other fuel assemblies are in the associated core quadrant. With four or less fuel assemblies loaded around each SRM and no other fuel assemblies in the associated quadrant, even with a control rod withdrawn the configuration will not be critical.

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

SR 3.3.1.2.5 and SR 3.3.1.2.6

Performance of a CHANNEL FUNCTIONAL TEST demonstrates the associated channel will function properly. SR 3.3.1.2.5 is required in MODE 5, and the 7 day Frequency ensures that the channels are OPERABLE while core reactivity changes could be in progress. This 7 day Frequency is reasonable, based on operating experience and on other Surveillances (such as a CHANNEL CHECK) that ensure proper functioning between CHANNEL FUNCTIONAL TESTS.

SR 3.3.1.2.6 is required in MODE 2 with IRMs on Range 2 or below and in MODES 3 and 4. Since core reactivity changes do not normally take place in MODES 3 and 4 and core reactivity changes are due only to control rod movement in MODE 2, the Frequency has been extended from 7 days to

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.2.5 and SR 3.3.1.2.6 (continued)

31 days. The 31 day Frequency is based on operating experience and on other Surveillances (such as CHANNEL CHECK) that ensure proper functioning between CHANNEL FUNCTIONAL TESTS.

Verification of the signal to noise ratio also ensures that the detectors are inserted to a normal operating level. In a fully withdrawn condition, the detectors are sufficiently removed from the fueled region of the core to essentially eliminate neutrons from reaching the detector. Any count rate obtained while fully withdrawn is assumed to be "noise" only. With few fuel assemblies loaded, the SRMs will not have a high enough count rate to determine the signal to noise ratio. Therefore, allowances are made for loading sufficient "source" material, in the form of irradiated fuel assemblies, to establish the conditions necessary to determine the signal to noise ratio. To accomplish this, SR 3.3.1.2.5 is modified by a Note that states that the determination of signal to noise ratio is not required to be met on an SRM that has less than or equal to four fuel assemblies adjacent to the SRM and no other fuel assemblies are in the associated core quadrant. With four or less fuel assemblies loaded around each SRM, even with a control rod withdrawn the configuration will not be critical.

The Note to the SR 3.3.1.2.6 allows the Surveillance to be delayed until entry into the specified condition of the Applicability. The SR must be performed in MODE 2 within 12 hours of entering MODE 2 with IRMs on Range 2 or below. The allowance to enter the Applicability with the 31 day Frequency not met is reasonable, based on the limited time of 12 hours allowed after entering the Applicability and the inability to perform the Surveillance while at higher power levels. Although the Surveillance could be performed while on IRM Range 3, the plant would not be expected to maintain steady state operation at this power level. In this event, the 12 hour Frequency is reasonable, based on the SRMs being otherwise verified to be OPERABLE (i.e., satisfactorily performing the CHANNEL CHECK) and the time required to perform the Surveillances.

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100-100000



BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.1.2.7

Performance of a CHANNEL CALIBRATION verifies the performance of the SRM detectors and associated circuitry. The Frequency considers the plant conditions required to perform the test, the ease of performing the test, and the likelihood of a change in the system or component status. The neutron detectors are excluded from the CHANNEL CALIBRATION (Note 1) because they cannot readily be adjusted. The detectors are fission chambers that are designed to have a relatively constant sensitivity over the range, and with an accuracy specified for a fixed useful life.

Note 2 to the Surveillance allows the Surveillance to be delayed until entry into the specified condition of the Applicability. The SR must be performed in MODE 2 within 12 hours of entering MODE 2 with IRMs on Range 2 or below. The allowance to enter the Applicability with the 18 month Frequency not met is reasonable, based on the limited time of 12 hours allowed after entering the Applicability and the inability to perform the Surveillance while at higher power levels. Although the Surveillance could be performed while on IRM Range 3, the plant would not be expected to maintain steady state operation at this power level. In this event, the 12 hour Frequency is reasonable, based on the SRMs being otherwise verified to be OPERABLE (i.e., satisfactorily performing the CHANNEL CHECK) and the time required to perform the Surveillances.

REFERENCES

None.

