Table 1. Owners' Group Changes Incorporated into ABWR Technical Specifications, Chapter 16, Amendment 33

OWners' Group Change Package No. and Item No.		Affected Changes	
BWR-01A	C1	3.3.1.1	SSLC Sensor Instrumentation
	C1	B 3.3.1.1	SSLC Sensor Instrumentation
	C1, C2	3.10.8	SDM Test-Refueling
	C1, C2	B 3.10.8	SDM Test-Refueling
BWR-02	C1, C2, C3, C4	1.1	Definition
	C5, C6	1.3	Completion Time
	C8, C8a	B 2.1.1	Reactor Core SLs
	C8, C9	B 2.2.4	RCS Pressure SL
	C11	3.0	LCO Applicability
BWR-03	C1	3.4.7	RHR-Hot Shutdown
	C1, C3	3.4.8	RHR-Cold Shutdown
	C4	3.9.7	RHR-High Water Level
	C4, C5	3.9.8	RHR-Low Water Level
	C1	B 3.4.7	RHR-Hot Shutdown
	C1, C3	B 3.4.8	RHR-Cold Shutdown
	C4	B 3.9.7	RHR-High Water Level
	C4, C5	B 3.9.8	RHR-Low Water Level
BWR-04	C1 (in P&R)	3.1.1	SDM
	C1 (in P&R)	B 3.1.1	SDM
	C4	3.1.4	Controol Rod Scram Times
BWR-04,	C3 Rev. 1	3.1.1	SDM and B 3.1.1, SR 3.1.1.1 Frequency
	C2 Rev. 1, C3	3.1.2	Reactivity Anomalies, and B 3.1.2, SR 3.1.2.1 Frequency
	C1 (in P&R)	B 3.1.1	SDM Required Actions C.1, D.1, D.2, D.3, and D.4
BWR-05	C1, C2, C3, C4, C5, C6, C8	1.1	Definitions
	C14	1.4	Frequency
	C7, C9	3.0	LCO Applicability
	C9	B 3.0	LCOs & SRs
	C12, C15, C11 Rev. 1	B 3.0	LCOs & SRs
BWR-06	C1, C2	1.1	Definitions
	C4	3.6.2.1	Suppression Pool Average Temperature
	C5	3.6.3.1	Primary Containment Hydrogen Recombiners
	C3 Rev. 1	1.3	Completion Times
	C3 Rev. 1	1.4	Frequency
	C9, Rev. 1	3.3.6.1	PAM Instrumentation

Table 1. Owners' Group Changes Incorporated into ABWR Technical Specifications, Chapter 16, Amendment 33

OWners' Group Change Package No. and Item No.		Affected Changes	
BWR-11,	C2	2.1.2	Reactor Coolant System Pressure SL
	C3, C4, C5, C6, C7, C8, C9	B2.1.1	Reactor Core SL
BWR-13,	C2	3.1.1	SDM, Required Action D.4, E.5
	C3	3.1.3	Control Rod Operability, Completion Time for Reqired Action B.1 and Bases
	C4	3.1.1	Control Rod Operability, Condition D, and Basis
	C5	3.1.4	Table 3.1.4-1 Note 2
	C7	3.1.5	Control Rod Scram Accumulator, Condition D
	C8, C9, C10	B 3.1.1	SDM
	C10, C12	B 3.1.3	Control Rod Operability
	C10, C11, C13, C14, C15	B 3.1.4	Bases
	C11, C17, C18	B 3.1.5	Bases
	C10, C11, C19		B 3.1.7 Bases
WOG-24	C1	B 3.7.5	Control Room Habitability Area (CRHA) - Air Conditioning (AC) System
	C9	4.0	Design Features
WOG-22	C6	B 3.4.4	RCS Pressure Isolation Valve (PIV) Leakage
WOG-32	C1	1.3	Completion Times
WOG-06	Items 1, 3, 5, 6, 7	5.0	Administrative Controls
BWOG-01	C1, C2	1.1	Definitions
	C5	1.2	Logical Connectors
	C7, C8, C9	1.3	Completion Times
	C11	3.0	SR Applicability
BWOG-02,	C3	LCO 3.4.6	RCS Specific Activity

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.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
BWR-02 C1 ~~AVERAGE PLANAR EXPOSURE~~ BWR-06 C1	~~The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.~~
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 heat generation rate per unit length of fuel rod 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 BWR-05 C1 Rev. 1 BWR-05 C8	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds within the necessary range and accuracy to specified 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 inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. ~~Whenever a sensing element is replaced, the next required inplace cross calibration consists of comparing the other sensing elements with the recently installed sensing element.~~ The CHANNEL CALIBRATION may be performed by means of any BWOG-01 C1

(continued)

1.1 Definitions

CHANNEL CALIBRATION (continued)

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.

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.

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c-2

~~b. Bistable channels (e.g., pressure switches and switch contacts) the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm and trip functions.~~

COMPREHENSIVE FUNCTIONAL TEST

A COMPREHENSIVE FUNCTIONAL TEST (CoFT) is a set of tests that exercises each RPS, ESF, and MSIV closure Function by simulating accident events that exercise the inputs and outputs of the SSLC, NMS, PRRM, RPS/MSIV actuation logic and ESF actuation logic. A CoFT also simulates power failures, measures CPU and network performance, runs microprocessor-specific and application-specific diagnostics. Test inputs include out-of-range conditions to verify OPERABILITY of the SSLC electronics, alarms and displays.

(continued)

1.1 Definitions

CORE ALTERATION

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C3

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C2

CORE ALTERATION shall be the movement of any fuel, sources, or other reactivity control components~~, or other components affecting reactivity~~ within the reactor vessel with the vessel head removed and fuel in the vessel. Movement of startup range neutron monitors, local power range monitors, traversing incore probes, or special movable detectors (including undervessel replacement) is not considered a CORE ALTERATION. In addition, control rod movement with other than the normal control rod drive is not considered a CORE ALTERATION 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 5.7.1.5 Plant operation within these limits is addressed in individual Specifications.

DIVISION FUNCTIONAL TEST

The injection of simulated or actual signals into a division as close to the sensors as practicable to verify OPERABILITY of SENSOR CHANNELS and LOGIC CHANNELS in that division. The DIVISION FUNCTIONAL TEST may be performed by means of a series of sequential or overlapping steps. The test shall comprise all the equipment from the DTM inputs to LOGIC CHANNEL outputs. This test shall also verify that the inputs to the DTMs are the same as the information presented at the control room indicators.

DOSE EQUIVALENT I-131

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C2

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 ~~Federal Guidance Report No 11~~Table III of TID-14844, AEC, 1962, "Calculation

(continued)

1.1 Definitions

DOSE EQUIVALENT I-131 (continued) BWOG-01 C.2	of Distance Factors for Power and Test Reactor Sites" or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977 or ICRP 30, Supplement to Part 1, pages 192-212, Table titled "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity.
$\bar{E}$ – AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.
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.
ISOLATION SYSTEM RESPONSE TIME BWR-02 C.2	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. 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.

(continued)

1.1 Definitions

LOGIC CHANNEL (continued)

actuation signals generated in the SLU out to the 2-out-of-2 voter.

LOGIC SYSTEM FUNCTIONAL TEST

BWR-02 C3

A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all required logic components (i.e., all required relays and contacts, trip functions, solid state logic elements, etc.) of a logic path, 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.

MINIMUM CRITICAL POWER RATIO (MCPR)

The MCPR shall be the smallest critical power ratio (CPR) that exists in the core. 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

BWR-05, C5

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, displays, 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). BWR-02 C4

OUTPUT CHANNEL

An OUTPUT CHANNEL is defined as a set of interconnected components that process outputs from associated LOGIC CHANNELS to produce an identifiable signal that deenergizes scram

(continued)

1.1 Definitions

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 consists of two components: a. The time from initial movement of the main turbine stop valve or control valve until 80% of the turbine bypass capacity is established; and b. The time from initial movement of the main turbine stop valve or control valve until initial movement of the turbine bypass valve. 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.

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c6

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE

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

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 AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

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

BWOG-01 C5

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

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

EXAMPLES

The following examples illustrate the use of logical connectors.

(continued)

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

BwR-02
C5

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

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.

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.

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C6

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.

(continued)

1.3 Completion Times

BWR-02, C6

DESCRIPTION (continued)

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

a. Must exist concurrent with the first inoperability; and

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

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

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

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

The above Completion Time extensions do not apply to those 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. BWR-02 C6

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

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. AND B.2 Be in MODE 4.	12 hours 36 hours

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C7

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, Condition 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)

1.3 Completion Times

EXAMPLES

<u>EXAMPLE 1.3-3</u> (continued)

BWR-02
C6

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)

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. AND B.2 Be in MODE 4.	12 hours 36 hours

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

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

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

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Rev. 1

(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. AND B.2 Be in MODE 4.	12 hours 36 hours

BWOG-01
C8

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 ~~A~~, the Note ~~may~~would appear in ~~the Condition column~~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 1.3-6 (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 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.

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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 AND Once per 8 hours thereafter
	AND	
	A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours
	AND	
	B.2 Be in MODE 4.	36 hours

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

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by 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

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(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-7 (continued)

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C9

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. ~~Since the second Completion Time of Required Action A.1 has a modified "time zero" (i.e., after the initial 1 hour, not from time of Condition entry), the allowance for a Completion Time extension does not apply.~~

IMMEDIATE COMPLETION TIME

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

1.4 Frequency

DESCRIPTION (continued)

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

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

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

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the 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

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(continued)

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-1 (continued)

(refer to Examples 1.4-3 and 1.4-4), then SR 3.0.3 becomes applicable.

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 1.4-2

SURVEILLANCE REQUIREMENTS

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

~~25%~~ BWR-06 C3, Rev. 1

(continued)

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 1.4-3

SURVEILLANCE REQUIREMENTS

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

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

As the Note modifies the required <u>performance</u> of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day interval (plus the extension allowed by 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 ≥ 25% RTP.

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(continued)

1.4 Frequency

EXAMPLES

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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~~; MODE changes then would be restricted in accordance with SR 3.0.4~~, and the provisions of SR 3.0.3 would apply.

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

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

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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 (plus the extension allowed by SR 3.0.2) interval, 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 < 55.2 Kg/cm^2g (785 psig) or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 55.2 Kg/cm^2g (785 psig) and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.07.

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

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2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ~~maintained~~ ≤ 93.1 Kg/cm^2g (1325 psig).

2.2 SL Violations

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

2.2.1 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

2.2.2 Within 2 hours:

2.2.2.1 Restore compliance with all SLs; and

2.2.2.2 Insert all insertable control rods.

2.2.3 Within 24 hours, notify the [General Manager—Nuclear Plant and Vice President—Nuclear Operations] and the [offsite reviewers specified in Specification 5.5.2, "[Offsite] Review and Audit"].

(continued)

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 (AOOs).

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

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

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

(continued)

BASES

BACKGROUND (continued)

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

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 an 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.

BWR-11 C5

The Reactor Protection System setpoints (LCO 3.3.1.1, "SSLC Sensor Instrumentation"), in combination with all the 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.

BWR-11 C4

2.1.1.1 Fuel Cladding Integrity (General Electric Company (GE) Fuel)

BWR-11 C6

GE critical power correlations are applicable for all critical power calculations at pressures $\geq$ 55.2 Kg/cm^{2}g (785 psig) or core flows $\geq$ 10% of rated flow. For operation at low pressures and low flows, another basis is used, as follows:

Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > .316 Kg/cm^2 (4.5 psi). Analyses (Ref. 2) show that with a bundle flow of 12.7 m^3/h (28 x 10^3 lb/hr), bundle pressure drop is nearly

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (General Electric Company (GE) Fuel) (continued)

independent of bundle power and has a value of .246 Kg/cm^2 (3.5 psi). Thus, the bundle flow with a .316 Kg/cm^2 (4.5 psi) driving head will be > 12.7 m^3/hr (28 x 10^3 lb/hr). Full scale ATLAS test data taken at pressures from 1 Kg/cm^2a (14.7 psia) to 56.2 Kg/cm^2a (800 psia) indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 50% RTP. Thus, a THERMAL POWER limit of 25% RTP for reactor pressure < 55.2 Kg/cm^2g (785 psig) is conservative.

BWR-11 C7

2.1.1.2 MCPR (GE Fuel)

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

The MCPR SL is determined using a statistical model that combines all the uncertainties in operating

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.2 MCPR (GE Fuel) (continued)

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

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2, the reactor vessel water level is required to be above the top of the active fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL 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.

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CB

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel, thus maintaining a coolable geometry.

(continued)

BASES

APPLICABILITY BWR-11 C9	SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES. ~~However, in MODES 3, 4, and 5, with the reactor shut down, it is unlikely that fuel cladding integrity SLs would be violated.~~
SAFETY LIMIT VIOLATIONS	2.2.1 If any SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 3). 2.2.2 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. 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 also ensures that the probability of an accident occurring during this period is minimal. 2.2.3 If any SL is violated, the appropriate senior management of the nuclear plant and the utility shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management. 2.2.4 If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 5). A copy of the report shall also be provided to the senior management of the nuclear plant, and the utility Vice
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(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

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2.2.4 (continued))

President - Nuclear Operations and the [offsite reviewers specified in Specification 5.5.2 ["Offsite Review and Audit"]].

2.2.5

If any SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. NEDE-24011-P-A, (latest approved revision).
3. 10 CFR 50.72.
4. 10 CFR 100.
5. 10 CFR 50.73.

BASES (continued)

APPLICABLE SAFETY ANALYSES

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

The RCS pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to ASME, Boiler and Pressure Vessel Code, Section III, [later Edition], including Addenda through the [later Edition] (Ref. 5), which permits a maximum pressure transient of 110%, 96.7 Kg/cm^2g (1375 psig), of design pressure 87.9 Kg/cm^2g (1250 psig). The SL of 93.1 Kg/cm^2g (1325 psig), as measured by the reactor steam dome pressure indicator, is equivalent to 96.7 Kg/cm^2g (1375 psig) at the lowest elevation of the RCS. The RCS is designed to ASME Code, Section III, [later Edition] (Ref. 6), for the reactor recirculation piping, which permits a maximum pressure transient of 110% of design pressures of 87.9 Kg/cm^2g (1250 psig) for suction piping and 116 Kg/cm^2g (1650 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 110% of design pressures of 87.9 Kg/cm^2g (1250 psig) for suction piping and 105.5 Kg/cm^2g (1500 psig) for discharge piping. The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is established at 96.7 Kg/cm^2g (1375 psig).

APPLICABILITY

SL 2.1.2 applies in all MODES~~; however, in MODE 5, because the reactor vessel head closure bolts are not fully tightened, it is unlikely the RCS would be pressurized~~.

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(continued)

BASES (continued)

SAFETY LIMIT VIOLATIONS

2.2.1

If any SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 7).

2.2.2

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). Therefore, it is required to 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.

2.2.3

If any SL is violated, the appropriate senior management of the nuclear plant and the utility shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

2.2.4

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If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 8). A copy of the report shall also be provided to the senior management of the nuclear plant, and the utility Vice President - Nuclear Operations, and the [offsite reviewers specified in Specification 5.5.2 ["Offsite Review and Audit"]].

2.2.5

If any SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

(continued)

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.

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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.

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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

(continued)

3.0 LCO APPLICABILITY

LCO 3.0.4 (continued)	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. 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 3.0.5 BWR-05 c9	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, the OPERABILITY of other equipment, or variables to be within limits. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the required testing.
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 5.8, "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 SR APPLICABILITY

SR 3.0.3 (continued)	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.
SR 3.0.4 BWOG-01 C11	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 ~~passage through or to~~entry into MODES or other specified conditions in ~~compliance with Required Actions~~the Applicability that are required to comply with ACTIONS.

BASES (continued)

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:

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

b. The OPERABILITY of other equipment; or

c. That variables are within limits.

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

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

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

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.0.1 (continued)

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.

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 3.1.3.4 is satisfied, the control rod can be considered OPERABLE. This allows startup to proceed to reach [800 psi] to perform other necessary testing.

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(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.0.2 (continued)

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.

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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

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SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met. This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

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

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (continued)	D.2	Initiate action to restore secondary containment to OPERABLE status.	1 hour
	AND		
	D.3	Initiate action to restore one standby gas treatment (SGT) subsystem to OPERABLE status.	1 hour
	AND		
	D.4	Initiate action to restore one isolation valve and associated instrumentation to OPERABLE status in each required secondary containment penetration flow path not isolated.	1 hour BWR-13 C2
E. SDM not within limits in MODE 5.	E.1	Suspend CORE ALTERATIONS except for control rod insertion and fuel assembly removal.	Immediately
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. (continued)	E.2 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
	AND	
	E.3 Initiate action to restore secondary containment to OPERABLE status.	1 hour
	AND	
	E.4 Initiate action to restore one SGT subsystem to OPERABLE status.	1 hour
	AND	
	E.5 Initiate action to restore one isolation valve and associated instrumentation to OPERABLE status in each required secondary containment penetration flow path not isolated.	1 hour BWR-13 C2

SURVEILLANCE REQUIREMENTS

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

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	Verify core reactivity difference between the monitored core k_{eff} and the predicted core k_{eff} is within ± 1% Δk/k ~~during operations in MODE 1~~. BWR-04 C2 Rev. 1	Once within 24 hours after reaching equilibrium conditions following startup after fuel movement within the reactor pressure vessel or control rod replacement BWR-04 C3 AND 1000 MWD/T thereafter during operations in MODE 1

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	---------NOTE--------- Not applicable when less than or equal to the low power setpoint (LPSP) of the Rod Control and Information System (RCIS). ----------------------	
		Perform SR 3.1.3.2 and SR 3.1.3.3 for each withdrawn OPERABLE control rod.	24 hours
	AND		
	A.3	Perform SR 3.1.1.1.	72 hours
B. Two or more withdrawn control rods stuck.	B.1	Disarm the associated CRD.	2 hour
	AND		
	B.2	Be in MODE 3.	12 hours

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(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more control rods inoperable for reasons other than Condition A or B.	---------NOTE--------- 1. Inoperable control rods may be bypassed in RAPI in accordance with SR 3.3.2.1.6, if required, to allow insertion of inoperable control rod and continued operation. 2. Inoperable control rods with failed motor drives can only be fully inserted by individual scram. ---------------------		
	C.1	Fully insert inoperable control rod	3 hours
	<u>AND</u>		
	C.2	Disarm the associated CRD.	4 hours
D. ---------NOTE--------- Not applicable when THERMAL POWER > 10% RTP. --------------------- Two or more inoperable control rods not in compliance with Ganged Withdrawal Sequence Restrictions (GWSR) and not separated by two or more OPERABLE control rods.	D.1	Restore compliance with GWSR.	4 hours
	<u>OR</u>		
	D.2	Restore control rod to OPERABLE status.	4 hours

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(continued)

Table 3.1.4-1
Control Rod Scram Times

-----NOTES-----

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

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2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod Operability," for control rods with scram times > [ ] seconds to 60% rod insertion position are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

ROD POSITION PERCENT INSERTION (%)	SCRAM TIMES(a) (seconds)		
	REACTOR STEAM DOME PRESSURE(b) 0 Kg/cm^2g (0 psig)	REACTOR STEAM DOME PRESSURE(b) 66.8 Kg/cm^2g (950 psig)	REACTOR STEAM DOME PRESSURE(b) 73.8 Kg/cm^2g (1050 psig)
10	(c)	[ ]	[ ]
40	(c)	[ ]	[ ]
60		[ ]	[ ]

(a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids as time zero.

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(b) For intermediate reactor steam dome pressures, the scram time criteria are determined by linear interpolation.

(c) For reactor steam dome pressure ≤ 66.8 Kg/cm^2g (950 psig), only 60% rod insertion position scram time limit applies.

2.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Required Action A.1 or B.1 not met. BWR-13 C1	C.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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.5.1 Verify each control rod scram accumulator pressure is ≥ 130 Kg/cm^2g.	7 days

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

SDM requirements are specified to ensure:

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

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

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

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

APPLICABLE SAFETY ANALYSES

The control rod removal error during refueling accident analysis (Ref. 2) assumes the core is subcritical with the highest worth control rod withdrawn. 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, or control rod pair, 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 or rod pair withdrawn, if adequate SDM has been demonstrated.

Prevention or mitigation of reactivity insertion events is necessary to limit energy deposition in the fuel to prevent significant fuel damage, which could result in undue release of radioactivity ~~(see Bases for LCO 3.1.7, "Standby Liquid~~

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C8

(continued)

BWR-13
C8

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~Control (SLC) System")~~. Adequate SDM ensures inadvertent criticalities will not cause significant fuel damage.

SDM satisfies Criterion 2 of the NRC Policy Statement.

LCO

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C9

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 or rod pair is determined analytically or by measurement. This is due to the reduced uncertainty in the SDM test when the highest worth control rod or rod pair is determined by measurement. When SDM is demonstrated by calculations not associated with a test (i.e., to confirm SDM during the fuel loading sequence), additional margin ~~must be added to the specified SDM limit~~ is included to account for uncertainties in the calculation. To ensure adequate SDM during the design process, a design margin is included to account for uncertainties in the design calculations (Ref. 3).

APPLICABILITY

In MODES 1 and 2, SDM must be provided because subcriticality with the highest worth control rod or rod pair withdrawn is assumed in the analysis (Ref. 4). In MODES 3 and 4, SDM is required to ensure the reactor will be held subcritical with margin for a single withdrawn control rod or rod pair. 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 or of a control rod pair from loaded core cells during scram time testing.

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.

(continued)

ACTIONS (continued)

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.

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. 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 (LCO 3.6.4.1, "Secondary Containment") is OPERABLE; at least one Standby Gas Treatment (SGT) subsystem (LCO 3.6.4.3, "Standby Gas Treatment (SGT) System") is OPERABLE; and at least one secondary containment isolation valve (LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)") and associated instrumentation (LCO 3.3.6.1, "Isolation Instrumentation") are OPERABLE in each associated penetration flow path not isolated. 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 SRs 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.

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C10

(continued)

ACTIONS
(continued)

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

BWR-13
C10

With SDM not within limits in MODE 5, the operator must immediately suspend CORE ALTERATIONS that could reduce SDM, e.g., ~~The suspensions are on~~ 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 (LCO 3.6.4.1) is OPERABLE; at least one SGT subsystem (LCO 3.6.4.3) is OPERABLE; and at least one secondary containment isolation valve (LCO 3.6.4.2) and associated instrumentation (LCO 3.3.6.1) are OPERABLE in each associated penetration flow path not isolated. 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 SRs 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

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

Adequate SDM must be demonstrated to ensure the reactor can be made subcritical from any initial operating condition. Adequate SDM is demonstrated by testing before or during the first startup after fuel movement, control rod replacement, or shuffling within the reactor pressure vessel. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from another core location. Since core reactivity will vary during the cycle as a function of fuel depletion and poison burnup, the beginning of cycle (BOC) test must also account for changes in core reactivity during the cycle. Therefore, to obtain the SDM, the initial measured value must be increased by an adder, "R", which is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated BOC core reactivity. If the value of R is negative (i.e., BOC is the most reactive point in the cycle), no correction to the BOC measured value is required (Ref. 4). For the SDM demonstrations that rely solely on calculation, additional margin (0.10% Δk/k) must be added to the SDM limit of 0.28% Δk/k to account for uncertainties in the calculation of the highest worth control rod or control rod pair.

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C9

The SDM may be demonstrated during an in sequence control rod pair withdrawal, in which the highest worth control rod pair is analytically determined, or during local criticals, where the highest worth control rod pair 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 MODE 5, adequate SDM is also required to ensure the reactor does not reach criticality during control rod withdrawals. An evaluation of each in vessel fuel movement during fuel loading (including shuffling fuel within the core) ~~shall be performed~~ is required to ensure adequate SDM

BwR-13
C10

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

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 SDM limit ot account for 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.

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c9

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. ABWR SSAR, Section 15.4.1.
3. ABWR SSAR, Section 4.3.2.
4. NDE-24011-P-A-9, "GE Standard Application for Reactor Fuel," Section 3.2.4.1, Sept. 1988.

BASES

ACTIONS

A.1 (continued)

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

B.1

If the core reactivity cannot be restored to within the 1% Δk/k limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Verifying the reactivity difference between the monitored and predicted 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. ~~This comparision requires the core to be operating at power levels which minimize the uncertainties and measurement errors, in order to obtain meaningful results. Therefore, the comparision is only done when in MODE 1.~~ 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

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C2 Rev.1

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1 (continued)

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 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 ≥ 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 comparision requires the core to be operating at power levels which minimize the uncertainties and measurement errors, in order to obtain meaningful results. Therefore, the comparision is only done when in MODE 1.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.
2. ABWR SSAR, Chapter 15.

BASES

ACTIONS

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

a reasonable time to test the control rods, considering the potential for a need to reduce power to perform the tests. Required Action A.2 is modified by a Note that states the requirement is not applicable when below the actual low power setpoint (LPSP) of the RC&IS, since the notch insertions may not be compatible with the requirements of rod pattern control (LCO 3.1.6) and the RCIS (LCO 3.3.1.2).

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 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 pair associated with the same HCU assumed to be fully withdrawn.

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 (and its associated control rod for the same HCU) to insert, sufficient reactivity control remains to reach and maintain MODE 3 conditions (Ref. 6). Required action A.2 performs a movement test on each remaining withdrawn control rod to ensure that no additional control rods are stuck. Therefore, 72 hours is allowed to perform the analysis to test in Required Action A.3.

BWR-13
C3

B.1 and B.2

With two or more withdrawn control rods stuck, the stuck control rods should be isolated from scram pressure within 2 hours and the plant brought to MODE 3 within 12 hours. Isolating the control rod from scram prevents damage to the CRD and surrounding fuel assemblies should a scram occur. The control rod can be isolated from scram by isolating its

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C12

(continued)

ACTIONS

B.1 and B.2 (continued)

associate hydraulic control unit. Two CRDs sharing an HCU can be individually isolated from scram. The occurrence of more than one control rod stuck at a withdrawn position increases the probability that the reactor cannot be shut down if required. Insertion of all insertable control rods eliminates the possibility of an additional failure of a control rod to insert. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

With one or more control rods inoperable for reasons other than being stuck in the withdrawn position, operation may continue, provided the control rods are fully inserted within 3 hours and disarmed (however, they do not need to be isolated from scram) 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 disarmed by disconnecting power to the motor drive or by placing the rod in RC&IS INOP Bypass. Required Action C.1 is modified by a Note that allows control rods to be bypassed in the RC&IS if required to allow insertion of the inoperable control rods and continued operation. Also, as noted, control rods declared inoperable with a failed motor drive can only be inserted by scram. Control rods with failed motor drives are not inoperable for this reason alone, but must be considered so upon failure of SR 3.1.3.2 or SR 3.1.3.3, or when not in compliance with GWSR (see LCO 3.1.6). This does not conflict with SR 3.0.1 since the ability to move the control rod via the FMCRD, as discussed in the bases for SR 3.1.3.2 and SR 3.1.3.3, is required to prove that the rod is not stuck. Likewise, loss of position indication, assuming no rod movement, would not result in control rod(s) inoperability until failure of SR 3.1.3.1. SR 3.3.2.1.6 provides additional requirements when the control rods are bypassed to ensure compliance with the RWE analysis.

The allowed Completion Times are reasonable, considering the small number of allowed inoperable control rods, and provide

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

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 control rod, or gang of control rods, during a RWE and therefore, the distribution of inoperable control rods must be controlled. At ≤ 10% RTP, the generic ganged withdrawal sequence restrictions (GWSR) (which is equivalent to previous banked position withdrawal sequence (BPWS) analysis (Ref. 6) requires inserted control rods not in compliance with GWSR to be separated by at least two OPERABLE control rods in all directions, including the diagonal. Therefore, if ~~one~~two or more inoperable control rods are not in compliance with GWSR and not separated by at least two OPERABLE control rods, action must be taken to restore compliance with GWSR 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 GWSR is not required to be followed under these conditions, as described in the Bases for LCO 3.1.6.

BWR-13 C10

BWR-13 C4

E.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

(continued)

BASES

ACTIONS

E.1 (continued)

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 CRD OPERABILITY and controlling rod patterns. Control rod position may be determined by the use of OPERABLE position indicators, by moving control rods to a position with an OPERABLE indicator, 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

BWR-13
c10

Control rod insertion capability is demonstrated by inserting each partially or fully withdrawn control rod at two notches 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 to ~~below~~ the actual LPSP of the RCIS since the notch insertions may not be compatible with the requirements of ~~rod pattern control~~the GWSR (LCO 3.1.6) and the RCIS (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 must be taken.

(continued)

BASES

LCO (continued)

indication. The reed switch closes ("pickup") when the hollow piston 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.

To ensure that local scram reactivity rates are maintained within acceptable limits, no more than two of the allowed "slow" control rods may occupy adjacent locations.

Table 3.1.4-1 is modified by two Notes, which state control rods with scram times not within the limits of the Table are considered "slow" and that control rods with scram times > [ ] seconds to 60% rod insertion position are considered inoperable as required by SR 3.1.3.4.

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C13

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" 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 ~~only allowed to be withdrawn under Special Operations LCO 3.10.3, "Single Control Rod Withdrawal — Hot Shutdown," and LCO 3.10.4, "Single Control Rod Withdrawal — Cold Shutdown," which~~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."

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C11

(continued)

BASES

ACTIONS

A.1

Bwr-13
C14

When the requirements of this LCO are not met, the rate of negative reactivity insertion during a scram may not be within the assumptions of the safety analyses. Therefore the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The four SRs of this LCO are modified by a Note stating that during a single or pair 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 or pair control rod scram times. During a full core scram, the CRD pump head would be seen by all control rods and would have a negligible effect on the scram insertion times.

SR 3.1.4.1

The scram reactivity used in DBA and transient analyses is based on assumed control rod scram time. Measurement of the scram times with reactor steam dome pressure ≥ 66.8 Kg/cm^2g (950 psig) demonstrates acceptable scram times for the transients analyzed in References 2 and 3.

Scram insertion times increase with increasing reactor pressure 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 greater than 66.8 Kg/cm^2g (950 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.1 (continued)

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before exceeding 40% RTP ~~following a shutdown~~. 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

BWR-13 C15

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~~, with~~. 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 (e.g., 20% of the entire sample size) is satisfied, ~~or Required Action A.1 must be taken.~~ or until the total number of "slow" control rods(throughout the core, from all Surveillances) exceeds the LCO limit. For planned testing, the control rods selected for the sample should be different for each test. Data from inadvertent scrams should be used whenever possible to avoid unnecessary testing at power, even if the control rods with data 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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Control rod scram accumulators satisfy Criterion 3 of the NRC Policy Statement.

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

BWR-13 C11

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 only allowed to be withdrawn ~~under Special Operations LCO 3.10.3, "Control Rod Withdrawal - Hot Shutdown," and LCO 3.10.4, "Control Rod Withdrawal - Cold Shutdown," which provide~~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

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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~~affected control rod. Complying with the Required Actions may allow for continued operation and subsequent ~~inoperable~~affected control rods governed by subsequent Condition entry and application of associated Required Actions.

A.1

With one control rod scram accumulator inoperable, the scram function could become severely degraded because the accumulator is the primary source of scram force for the associated control rod or rod pair at all reactor pressures.

(continued)

BASES

ACTIONS

C.1 (continued)

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 130 Kg/cm^2g is well below the expected pressure of 150 Kg/cm^2g (Ref. 2). 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 ~~other~~ indications available in the control room.

BWR-13 c18

REFERENCES

1. NEDE-24011-P-A, "General Electric Standard Application Fuel," September 1988.
2. ABWR SSAR, Section 4.6.1.
3. ABWR SSAR, Section 5.2.2.
4. ABWR SSAR, Section 15.4.1

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 motor operated injection 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 "B" high pressure core flooder (HPCF) subsystem sparger.

APPLICABLE SAFETY ANALYSES

The SLC System is automatically initiated. 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 a quantity of boron that produces a concentration of 850 ppm of natural boron in the reactor core at 20°C with the reactor at normal 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).~~Considering uncertainties of mixing and the dilution by RHR water, the borated water concentration will be 1070 ppm (Ref. 2) or higher for a mass of water equal to the sum of the mass of water in the RPV at normal water level and the RHR shutdown cooling piping. The temperature versus concentration limits in Figure 3.1.7-1~~(in the accompanying LCO)~~ are calculated such that the required concentration is achieved~~accounting for dilution in the RPV with normal water level and including the water volume in~~

BWR-13 C10

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~the residual heat removal shutdown cooling piping~~. This quantity of borated solution is the amount that is above the pump suction shutoff level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected.

The SLC System satisfies the requirements of the NRC Policy Statement because operating experience and probabilistic risk assessment have generally shown it to be important to public health and safety.

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. Because the minimum required boron solution concentration is the same for both ATWS mitigation and cold shutdown (unlike some previous reactor designs) then if the boron solution concentration is less than the required limit, both SLC subsystems shall be declared inoperable. Two SLC subsystems are required to be OPERABLE, each containing an OPERABLE pump, a motor operated injection valve, and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

APPLICABILITY

BWR-13 C11

In MODES 1 and 2, shutdown capability is required. In MODES 3 and 4, control rods are ~~only allowed to be withdrawn under Special Operations LCO 3.10.3, "Control Rod Withdrawal—Hot Shutdown," and LCO 3.10.4, "Control Rod Withdrawal—Cold Shutdown," which~~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 or control rod pair 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 or control rod pair can be withdrawn.

(continued)

BASES

ACTIONS

A.1

If the boron concentration is less than the required limits given in Figure 3.1.7-1, the concentration must be restored to within limits in 72 hours. For ATWS prevention/mitigation the ABWR features: an automatic rod insert (ARI) and an electrical insertion of FMCRDs, both of which utilize sensors and logic that are diverse and independent of the reactor protection system; an ATWS recirculation pump trip (RPT); and, automatic initiation of SLCS under ATWS conditions (Ref. 3). These features provide the ABWR an ATWS prevention and mitigation capability well beyond previous BWRs. Because of the low probability of an ATWS event, the ATWS prevention/mitigation features and the fact that SLC System capability still exists for vessel injection under these conditions, the allowed Completion Time of 72 hours is acceptable and provides adequate time to restore concentration to within limits. ~~The maximum Completion Time of 10 days is allowed for this LCO in the event of multiple Condition entry.~~The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of concentration out of limits or inoperable SLC subsystems during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, an SLC subsystem is inoperable and that subsystem is subsequently returned to OPERABLE, the LCO may already have been not met for up to 7 days. This situation could lead to a total duration of 10 days (7 days in Condition B, followed by 3 days in Condition A), since initial failure of the LCO, to restore the SLC System. Then an SLC subsystem could be found inooperable again, and concentration could be restored to within limits. This could continue indefinitely.

BWR-13
C19

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock," resulting in establishing the "time zero" at the time the LCO was initially not met instead of at the time Condition A was entered. The 10 day Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

(continued)

BASES

ACTIONS (continued)

B.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 shutdown function. However, the overall reliability is reduced because a single failure in the remaining OPERABLE subsystem could result in reduced SLC System shutdown capability. The 7 day Completion Time is based on the availability of an OPERABLE subsystem capable of performing the intended 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.~~an ATWS event. ~~The maximum Completion Time of 10 days is allowed for this LCO in the event of multiple Condition entry.~~The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of concentration out of limits or inoperable SLC subsystems during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, concentration is out of limits, and is subsequently returned to within limits, the LCO may already have been not met for up to 7 days. This situation could lead to a total duration of 10 days (3 days in Condition A, followed by 7 days in Condition B), since initial failure of the LCO, to restore the SLC System. Then concentration could be found out of limits again, and the SLC subsystem could be restored to OPERABLE. This could continue indefinitely.

BWR-13 C19

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock," resulting in establishing the "time zero" at the time the LCO was initially not met instead of at the time Condition B was entered. The 10 day Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

C.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~~

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Specific Activity

LCO 3.4.6 The specific activity of the reactor coolant shall be limited to:

a. DOSE EQUIVALENT I-131 specific activity ≤ 7400 Bq/gm (0.2 μCi/gm); and

b. Gross specific activity ≤ 3.7E + 6/$\bar{E}$/gm (100/$\bar{E}$ μCi/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 > 7400 Bq/gm (0.2 μCi/gm) and ≤ 148,000 Bq/gm (4.0 μCi/gm) DOSE EQUIVALENT I-131.	------------NOTE------------ LCO 3.0.4 is not applicable. ----------------------------	BWOG-02 C3
	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 > 148,000 Bq/gm (4.0 μCi/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)

3.4 REACTOR COOLANT SYSTEM (RCS)

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

LCO 3.4.7 Two RHR shutdown cooling subsystems shall be OPERABLE, and, with less than 5 reactor internal pumps (RIPs) in operation, at least one RHR shutdown cooling subsystem shall be in operation.

-----NOTES-----

BWR-03
C1

1. All RHR shutdown cooling subsystems and reactor internal pumps may be removed from operation for up to 2 hours per 8 hour period ~~provided one RHR shutdown cooling subsystem is OPERABLE~~.

2. One RHR shutdown cooling subsystem may be inoperable for up to 2 hours for performance of Surveillances provided one of the remaining RHR shutdown cooling subsystems is OPERABLE.

APPLICABILITY: MODE 3 with reactor steam dome pressure < 9.5 Kg/cm^2g (135 psig).

ACTIONS

-----NOTE-----

LCO 3.0.4 is not applicable.

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

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

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

LCO 3.4.8 ~~Two~~Three RHR shutdown cooling subsystems shall be OPERABLE, and, with less than 5 reactor internal pumps (RIPs) in operation, at least one RHR shutdown cooling subsystem shall be in operation.

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

BwR-03 C1

1. ALL RHR shutdown cooling subsystems and reactor internal pumps may be removed from operation for up to 2 hours per 8 hour period~~ provided one subsystem is OPERABLE~~.

BwR-03 C1, C3

2. One RHR shutdown cooling subsystem may be inoperable for up to 2 hours for the performance of Surveillances~~,~~ ~~provided one of the remaini[illegible] shutdown cooling subsystems is OPERABLE and [illegible] ation~~.

3. One RHR shutdown cooling subsystem may be inoperable after 30 hours from initial entry into MODE 4 from MODE 3.

APPLICABILITY: MODE 4.

ACTIONS

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

(continued)

BASES

ACTIONS

BWOG-02
C3

A Note to the ACTIONS excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the sigificant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

A.1 and A.2

When the reactor coolant specific activity exceeds the LCO DOSE EQUIVALENT I-131 limit, but is ≤ 148,000 Bq/gm (4.0 μCi/gm), samples must be analyzed for DOSE EQUIVALENT I-131 at least once every 4 hours. In addition, the specific activity must be restored to the LCO limit within 48 hours. The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 48 hour Completion Time to restore the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes or crud bursts) to be cleaned up with the normal processing systems.

B.1, B.2.1, B.2.2.1, and B.2.2.2

If the DOSE EQUIVALENT I-131 cannot be restored to ≤ 7400 Bq/gm (0.2 μCi/gm) within 48 hours, or if at any time it is > 148,000 Bq/gm (4.0 μCi/gm), it must be determined at least every 4 hours and all the main steam lines must be isolated within 12 hours. Isolating the main steam lines precludes the possibility of releasing radicactive material to the environment in an amount that is more than a small fraction of the requirements of 10 CFR 100 during a postulated MSLB accident.

Alternately, the plant can be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. This option is provided for those instances when isolation of main steam lines is not desired (e.g., due to the decay heat loads). In MODE 4, the requirements of the LCO are no longer applicable.

(continued)

BASES

LCO (continued)

cooling subsystem consists of one OPERABLE RHR pump, a heat exchanger, and the associated piping and valves. Each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. In MODE 3, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain or reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required.

BWR-03 c1

Note 1 permits all RHR shutdown cooling subsystems and reactor internal pumps to be shut down for a period of 2 hours in an 8 hour period~~, provided one subsystem is OPERABLE~~. Note 2 allows one RHR shutdown cooling subsystem to be inoperable ~~for up to 2 hours~~ for performance of surveillance tests ~~provided one of the remaining RHR shutdown cooling subsystems is OPERABLE~~. These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR system in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR subsystems or other operations requiring RHR flow interruption and loss of redundancy.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with reactor steam dome pressure above the RHR cut in permissive pressure, this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal at reactor pressures above the RHR cut in permissive pressure is typically accomplished by condensing the steam in the main condenser. Additionally, in MODE 2 below this pressure, the OPERABILITY requirements for the Emergency Core Cooling Systems (ECCS) (LCO 3.5.1, "ECCS — Operating") do not allow placing the low pressure RHR shutdown cooling subsystem into operation.

(continued)

BASES

LCO (continued)

cooling subsystem consists of one OPERABLE RHR pump, a heat exchanger, and the associated piping and valves.

Each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. In MODE 4, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain and reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required.

BWR-03
C1, C3

Note 1 permits all RHR shutdown cooling subsystems and RIPs to be shut down for a period of 2 hours in an 8 hour period. ~~provided one subsystem is OPERABLE.~~ Note 2 allows one RHR shutdown cooling subsystem to be inoperable [illegible] up to 2 hours for the performance of surveillance tests. ~~provided one of the remaining RHR shutdown cooling subsystems is OPERABLE.~~ These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR System in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR subsystems or other operations requiring RHR flow interruption and loss of redundancy.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with reactor steam dome pressure above the RHR cut in permissive pressure, this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal at reactor pressures above the RHR cut in permissive pressure is typically accomplished by condensing the steam in the main condenser. Additionally, in MODE 2 below this pressure, the OPERABILITY requirements for the Emergency Core Cooling Systems (ECCS) (LCO 3.5.1, "ECCS — Operating") do not allow placing the low pressure RHR shutdown cooling subsystem into operation.

APPLICABILITY

In MODE 4, the RHR System may be operated in the shutdown

(continued)

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 recombiner inoperable.	A.1	---------NOTE--------- LCO 3.0.4 is not applicable. ---------------------- Restore primary containment hydrogen recombiner to OPERABLE status.	30 days
B. Two primary containment hydrogen recombiners inoperable.	B.1	Verify by administrative means that the hydrogen control function is maintained.	1 hour __AND__ Once per 12 hours thereafter
	__AND__		
	B.2	Restore one primary containment hydrogen recombiner to OPERABLE status.	7 days
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	12 hours

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C5

BASES

ACTIONS

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

a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and handling of irradiated fuel in the primary or secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

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C1

This SR verifies that the heat removal capability of the system is sufficient to remove the MCAE heat load assumed in the safety analyses. The SR consists of a combination of testing and calculation. The 18 month Frequency is appropriate since significant degradation of the CRHA AC System is not expected over this time period.

REFERENCES

1. ABWR SSAR, Section 6.4.
2. ABWR SSAR, Section 9.4.1.

3.9 REFUELING OPERATIONS

3.9.7 Residual Heat Removal (RHR) — High Water Level

LCO 3.9.7 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 ≥ 7.0 m above the top of the RPV flange.

BWR-03
C4

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 AND 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.	Immediately
	AND		
	B.2	Initiate action to restore primary or secondary containment to OPERABLE status.	Immediately
	AND		
			(continued)

3.9 REFUELING OPERATIONS

3.9.8 Residual Heat Removal (RHR) — Low Water Level

LCO 3.9.8 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 < 7.0 m above the top of the RPV flange.

BWR-03
C4

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 AND Once per 24 hours thereafter
B. Required Action and associated Completion Time of Condition A not met.	B.1	Initiate action to restore primary or secondary containment to OPERABLE status.	Immediately
	AND		
BWR-03 C5	B.2	Initiate action to restore one standby gas treatment subsystem to OPERABLE status.	Immediately
	AND		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.3 Initiate action to restore one secondary containment isolation valve and associated instrumentation to OPERABLE status in each associated penetration flow path not isolated.	Immediately
C. No RHR shutdown cooling subsystem in operation.	C.1 Establish reactor coolant circulation by an alternate method.	1 hour from discovery of no reactor coolant circulation
	AND	
	C.2 Monitor reactor coolant temperature.	Once per hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.8.1	Verify one RHR shutdown cooling subsystem is operating and circulating reactor coolant.	12 hours

BASES

LCO (continued)

An OPERABLE RHR shutdown cooling subsystem consists of an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path.

Additionally, each RHR shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. Operation (either continuous or intermittent) of one subsystem can maintain and reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required. A Note is provided to allow a 2 hour exception to shut down the operating subsystem every 8 hours.

APPLICABILITY

BWR-03
C4

One RHR shutdown cooling subsystem is required to be OPERABLE in MODE 5, with irradiated fuel in the reactor pressure vessel and with the water level ≥ 7.0 m above the top of the RPV flange, to provide decay heat removal. RHR System requirements in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS); Section 3.5, Emergency Core Cooling Systems (ECCS); and Section 3.6, Containment Systems. RHR System requirements in MODE 5, with irradiated fuel in the reactor pressure vessel and with the water level < 7.0 m above the RPV flange, are given in LCO 3.9.8, "Residual Heat Removal (RHR) - Low Water Level."

ACTIONS

A.1

With no RHR shutdown cooling subsystem OPERABLE, an alternate method of decay heat removal must be established within 1 hour. In this condition, the volume of water above the RPV flange provides adequate capability to remove decay heat from the reactor core. However, the overall reliability is reduced because loss of water level could result in reduced decay heat removal capability. The 1 hour Completion Time is based on the decay heat removal function and the probability of a loss of the available decay heat removal capabilities. Furthermore, verification of the functional availability of these alternate method(s) must be

(continued)

BASES

LCO (continued)

Additionally, each RHR shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. Operation (either continuous or intermittent) of one subsystem can maintain and reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, continuous operation is required. A Note is provided to allow a 2 hour exception to shut down the operating subsystem every 8 hours.

APPLICABILITY

BWR-03
C4

Two RHR shutdown cooling subsystems are required to be OPERABLE in MODE 5, with irradiated fuel in the reactor pressure vessel and with the water level < 7.0 m above the top of the RPV flange, to provide decay heat removal. RHR System requirements in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS); Section 3.5, Emergency Core Cooling Systems (ECCS); and Section 3.6, Containment Systems. RHR System requirements in MODE 5, with irradiated fuel in the reactor pressure vessel and with the water level ≥ 7.0 m above the RPV flange, are given in LCO 3.9.7, "Residual Heat Removal (RHR)- High Water Level."

ACTIONS

A.1

With one of the two required RHR shutdown cooling subsystems inoperable, the remaining subsystem is capable of providing the required decay heat removal. However, the overall reliability is reduced. Therefore an alternate method of decay heat removal must be provided (such as the third RHR shutdown cooling subsystem). With both required RHR shutdown cooling subsystems inoperable, an alternate method of decay heat removal must be provided in addition to that provided for the initial RHR shutdown cooling subsystem inoperability. This re-establishes backup decay heat removal capabilities, similar to the requirements of the LCO. The 1 hour Completion Time is based on the decay heat removal function and the probability of a loss of the available decay heat removal capabilities. Furthermore, verification of the functional availability of these alternate method(s)

(continued)

BASES

ACTIONS

A.1 (continued)

must be reconfirmed every 24 hours thereafter. This will ensure continued heat removal capability.

Alternate decay heat removal methods are available to the operators for review and preplanning in the unit's Operating Procedures. For example, in addition to the third RHR shutdown cooling loop, this may include the use of the Reactor Water Cleanup System, operating with the regenerative heat exchanger bypassed. The method used to remove decay heat should be the most prudent choice based on unit conditions.

BWR-03
C5

B.1, B.2, B.3, C.1, and C.2

If no RHR shutdown cooling subsystem is in operation, an alternate method of coolant circulation is required to be established within 1 hour. The Completion Time is modified such that the 1 hour is applicable separately for each occurrence involving a loss of coolant circulation.

During the period when the reactor coolant is being circulated by an alternate method (other than by the required RHR Shutdown Cooling System), the reactor coolant temperature and level must be periodically monitored to ensure proper function of the alternate method. The once per hour Completion Time is deemed appropriate.

If at least one RHR subsystem is not restored to OPERABLE status immediately, additional actions are required to minimize any potential fission product release to the environment. This includes initiating immediate action to restore the following to OPERABLE status: secondary containment, one standby gas treatment subsystem, and one secondary containment isolation valve and associated instrumentation in each associated penetration not isolated. This may be performed as an administrative check, by examining logs or other information to determine whether 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, the

(continued)

BASES

ACTIONS (continued)

B.1, B.2, B.3, C.1, and C.2 (continued)

surveillance may need to be performed to restore the component to OPERABLE status. Actions must continue until all required components are OPERABLE.

SURVEILLANCE REQUIREMENTS

SR 3.9.8.1

This Surveillance demonstrates that one RHR subsystem is in operation and circulating reactor coolant. The required flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability. The Frequency of 12 hours is sufficient in view of other visual and audible indications available to the operator for monitoring the RHR subsystem in the control room.

REFERENCES

None.

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:

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C1

a. LCO 3.3.1.1, "SSLC Sensor Instrumentation," MODE 2 requirements for Functions 2.a and 2.d of Table 3.3.1.1-1;

b. 1. LCO 3.3.5.1, "Control Rod Block Instrumentation," MODE 2 requirements for Function 1.b of Table 3.3.5.1-1,

<u>OR</u>

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 CRD;

d. All control rod withdrawals [during out of sequence control rod moves] shall be made in notch out mode; and

e. No other CORE ALTERATIONS are in progress.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more of the above requirements not met, for reasons other than Condition B.	A.1 Place the reactor mode switch in the shutdown or refuel position.	Immediately
B. One control rod not coupled to its associated CRD.	B.1 Declare the affected control rod inoperable.	Immediately

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SURVEILLANCE REQUIREMENTS

BWR-01A C1

SURVEILLANCE		FREQUENCY
SR 3.10.8.1	Perform the applicable SRs for LCO 3.3.1.1, Functions 2.a and 2.d.	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 applicable SRs for LCO 3.3.5.1, Function 1.b.	According to the applicable SRs

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
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
SR 3.10.8.5	Verify each withdrawn 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 satisfying LCO 3.10.8.c requirement after work on control rod or CRD System that could affect coupling

BASES

APPLICABLE SAFETY ANALYSES (continued)

RWE analyses assume that the reactor operator follows prescribed withdrawal sequences. For SDM tests performed within these defined sequences, the analyses of References 1 and 2 are applicable. However, for some sequences developed for the SDM testing, the control rod patterns assumed in the safety analyses of References 1 and 2 may not be met. Therefore, special RWE analyses, performed in accordance with an NRC approved methodology, are required to demonstrate that the SDM test sequence will not result in unacceptable consequences should a RWE occur during the testing. For the purpose of this test, protection provided by the normally required MODE 5 applicable LCOs, in addition to the requirements of this LCO, will maintain normal test operations as well as postulated accidents within the bounds of the appropriate safety analyses (Refs. B3.10.8-1 and B3.10.8-2). In addition to the added requirements for the RWM, SRNM, APRM, and control rod coupling, the notch out mode is specified for out of sequence withdrawals. Requiring the notch out mode limits withdrawal steps to a single notch, which limits inserted reactivity, and allows adequate monitoring of changes in neutron flux, which may occur during the test.

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

LCO

As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. SDM tests may be performed while in MODE 2, in accordance with Table 1.1-1, without meeting this Special Operations LCO or its ACTIONS. For SDM tests performed while in MODE 5, additional requirements must be met to ensure that adequate protection against potential reactivity excursions is available. Because multiple control rods will be withdrawn and the reactor will potentially become critical, RPS MODE 2 requirements for Functions 2.a and 2.d of Table 3.3.1.1-1 must be enforced and the approved control rod withdrawal sequence must be enforced by the RWM (LCO 3.3.5.1, Function 1b, MODE 2), or must be verified by a second licensed operator or other

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C1

(continued)

BASES

LCO (continued)

qualified member of the technical staff. To provide additional protection against an inadvertent criticality, control rod withdrawals that do not conform to the ganged withdrawal sequence restrictions specified in LCO 3.1.6, "Rod Pattern Control" (i.e., out of sequence control rod withdrawals) must be made in the notched withdrawal mode to minimize the potential reactivity insertion associated with each movement. Coupling integrity of withdrawn control rods is required to minimize the probability of a RWE and ensure proper functioning of the withdrawn control rods, if they are required to scram. Because the reactor vessel head may be removed during these tests, no other CORE ALTERATIONS may be in progress. This Special Operations LCO then allows changing the Table 1.1-1 reactor mode switch position requirements to include the startup/hot standby position, such that the SDM tests may be performed while in MODE 5.

APPLICABILITY

These SDM test Special Operations requirements are only applicable if the SDM tests are to be performed while in MODE 5 with the reactor vessel head removed or the head bolts not fully tensioned. Additional requirements during these tests to enforce control rod withdrawal sequences and restrict other CORE ALTERATIONS provide protection against potential reactivity excursions. Operations in all other MODES are unaffected by this LCO.

ACTIONS

A.1

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

B.1

BWR-01A
C2

With the requirements of this LCO not met, the affected control rod shall be declared inoperable. This results in a condition that is consistent with the requirements for MODE

(continued)

BASES

ACTIONS (continued)

5 where the provisions of this Special Operations LCO are no longer required.

SURVEILLANCE REQUIREMENTS

BWR-01A
C1

SR 3.10.8.1

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

SR 3.10.8.2 and SR 3.10.8.3

The control rod withdrawal sequences during the SDM tests may be enforced by the RWM (LCO 3.3.5.1, Function 1b, MODE 2 requirements) or by a second licensed operator or other qualified member of the technical staff. As noted, either the applicable SRs for the RWM (LCO 3.3.5.1) must be satisfied according to the applicable Frequencies (SR 3.10.8.1 and SR 3.10.8.2), or the proper movement of control rods must be verified. This latter verification (i.e., SR 3.10.8.2) must be performed during control rod movement to prevent deviations from the specified sequence. These surveillances provide adequate assurance that the specified test sequence is being followed.

SR 3.10.8.4

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

SR 3.10.8.5

Coupling verification is performed to ensure the control rod is connected to the control rod drive mechanism and will perform its intended function when necessary. The verification is required to be performed any time a control rod is withdrawn to the "full out" notch position or prior

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.10.8.5 (continued)

to declaring the control rod OPERABLE after work on the control rod or CRD System that could affect coupling. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved as well as operating experience related to uncoupling events.

REFERENCES

1. NEDE-24011-P-A-US, General Electric Standard Application for Reactor Fuel, Supplement For United States (as amended).

2. ABWR SSAR, Section 15.4.1.

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. Fuel assemblies having a maximum k-infinity of 1.35 in the normal reactor core configuration at cold conditions;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the SSAR;

WOG-24
c9

~~c. A nominal fuel assembly center to center storage spacing of [ ] cm, within a neutron poison material between storage spaces, in the high density storage racks in the spent fuel storage pool.~~

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

a. Fuel assemblies having a maximum k-infinity of 1.35 in the normal reactor core configuration at 20 °C;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the SSAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the SSAR; and

d. A nominal [ ] cm center to center distance between fuel assemblies placed in 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 3.1 m (10 ft) above the top of the active fuel.

(continued)