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.
ACTUATION LOGIC TEST	An ACTUATION LOGIC TEST shall be the application of various simulated or actual input combinations in conjunction with each possible interlock logic state and the verification of the required logic output. The ACTUATION LOGIC TEST, as a minimum, shall include a continuity check of output devices.
AXIAL FLUX DIFFERENCE (AFD)	AFD shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel so that it responds within the required range and accuracy to known input. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, interlock, display, and trip functions. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may ~~shall~~ consist of an inplace cross calibration of the sensing elements 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 series of sequential, overlapping calibrations or total channel steps so that the entire channel is calibrated.

A

qualitative assessment of channel behavior

(continued)

1.1 Definitions (continued)

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 OPERATIONAL TEST (COT)

A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, display, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.

CORE ALTERATION

CORE ALTERATION shall be the movement of any fuel, sources, reactivity control components, or other ~~components affecting reactivity~~ within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position. A

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 parameter limits shall be determined for each reload cycle in accordance with Specification 5.9.1.6. Plant operation within these limits is addressed in individual Specifications.

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977.

(continued)

1.1 Definitions (continued)

$\bar{E}$ - 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. B
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF 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.
LEAKAGE	LEAKAGE shall be:

a. <u>Identified LEAKAGE</u>

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System;

(continued)

1.1 Definitions

LEAKAGE (continued)

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

MASTER RELAY TEST

A MASTER RELAY TEST shall consist of energizing each master relay and verifying the OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

MODE

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

OPERABLE – OPERABILITY

A

or have OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:

a. Described in Chapter 14, Initial Test Program of the FSAR;

b. Authorized under the provisions of 10 CFR 50.59; or

(continued)

1.1 Definitions

SHUTDOWN MARGIN (SDM) (continued)	b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level. With any RCCAs not capable of being fully inserted, the reactivity worth of the RCCAs must be accounted for in the determination of SDM. (A)
SLAVE RELAY TEST	A SLAVE RELAY TEST shall consist of energizing each slave relay and verifying the OPERABILITY of each slave relay. The SLAVE RELAY TEST shall include, as a minimum, a continuity check of associated testable actuation devices.
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.
TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of required alarm, interlock, display, and trip functions. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the required accuracy.

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 <u>AND</u> and <u>OR</u>. 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 indentations of the logical connectors.

A

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.

Condition, A

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 Conditions, Completion Time, Surveillance, or Frequency.

A

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

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). (A)

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.

Once a Condition has been entered, subsequent trains, 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

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 5.	6 hours 36 hours

When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is 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, the Required Actions of Condition B may be terminated. A

Condition A and B are exited, and therefore,

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.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for

(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.	6 hours 12 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 (including ~~any~~ the extension~~s~~) expires while one or more valves are still inoperable, Condition B is entered. A

(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.	6 hours 12 hours

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

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 1.3-7 (continued)

Condition A was initially entered. If Required Action A.1 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.

A

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

EXAMPLES (continued)

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.

A

1.25 times the stated Frequency

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 25% extension allowed by SR 3.0.2. "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.

(continued)

1.4 Frequency

EXAMPLES (continued)

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 performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day (plus ~~25% per SR 3.0.2~~ the 1.25 times the Stated Frequency extension allowed by SR 3.0.2) interval, 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.

A

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.

A

JUSTIFICATION FOR CHANGES TO SECTION 2.0

GENERIC JUSTIFICATIONS

A. This change reflects a change to NUREG-1431 submitted to the NRC by the Industry Owners groups via the generic STS change process.

B. Removal of brackets and/or addition of plant specific parameter or information.

C. Correction of typographical error in the Proof and Review TS.

SPECIFIC JUSTIFICATIONS

1. This information was added to address NRC's question 2 on SER section 4.2.3 in a letter dated March 15, 1993 from Peter S. Tam to Mark O. Medford.

2. The NUREG references the non-proprietary version of the WCAP. The applicable reference for Watts Bar is the proprietary version of the referenced WCAP.

3. References to CFRs do not normally include dates.

4. Watts Bar's RCS piping and valves are also designed in accordance with ASME III, not USAS B31.1. The reference to B31.1 is not needed and should be deleted and other references renumbered accordingly.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the SLs specified in Figure 2.1.1-1.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained $\leq$ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

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

2.2.4 Within 24 hours, notify the Plant Manager and Site Vice President and the Nuclear Safety Review Board (NSRB). C

2.2.5 Within 30 days a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC, the NSRB, and the Plant Manager and Site Vice President. C

2.2.6 Operation of the unit shall not be resumed until authorized by the NRC.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB 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.

INSERT 1

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(continued)

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INSERT 1

DNB is not a directly measurable parameter during operation; therefore, THERMAL POWER, reactor coolant temperature, and pressure are related to DNB through critical heat flux (CHF) correlations. The WRB-1 CHF correlation (Ref. 7) is the primary DNB correlation and takes credit for significant improvement in the accuracy of the CHF predictions. The W-3 CHF correlation (Ref. 8 and 9) is used for conditions outside the range of the WRB-1 correlation.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in the FSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

The curves provided in Figure B 2.1.1-1 show the loci of points of THERMAL POWER, RCS pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the exit quality is within the limits defined by the DNBR correlation.

The curves are based on enthalpy hot channel factor limits provided in the COLR. The dashed line of Figure B 2.1.1-1 shows an example of a limit curve at 2235 psig. In addition, it illustrates the various RPS functions that are designed to prevent the unit from reaching the limit.

The SL is higher than the limit calculated when the AFD is within the limits of the $F_1(\Delta I)$ function of the overtemperature ΔT reactor trip. When the AFD is not within the tolerance, the AFD effect on the overtemperature ΔT reactor trips will reduce the setpoints to provide protection consistent with the reactor core SLs (Refs. 3 and 4).

INSERT 2

1

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4,

(continued)

INSERT 2

SL 2.1.1 reflects the use of the WRB-1 CHF correlation DNBR limit of 1.17 and a safety analysis DNBR limit of 1.31. Comparison of these DNBR limits results in a 10.7% DNBR margin which is more than sufficient to offset the DNBR penalty due to rod bow.

BASES

APPLICABILITY (continued)	5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs.

2.2.1

If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

2.2.3

If SL 2.1.1 is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 5).

2.2.4

If SL 2.1.1 is violated, the Plant Manager, Site Vice President, and Nuclear Safety Review Board (NSRB) shall be notified within 24 hours. This 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

2.2.5

If SL 2.1.1 is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC, the Plant Manager, Site Vice President, and NSRB. A

~~This requirement is~~ in accordance with 10 CFR 50.73 (Ref. 6). A copy of the report shall also be provided to

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.6

If SL 2.1.1 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. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 10, "Reactor Design," ~~1988~~. (3)

2. Watts Bar FSAR, Section 7.2, "Reactor Trip System".

3. WCAP-8746-A, "Design Bases for the Overtemperature ΔT and the Overpower ΔT Trips," March 1977.

4. WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985. (2)

5. Title 10, Code of Federal Regulations, Part 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors."

6. Title 10, Code of Federal Regulations, Part 50.73, "Licensee Event Report System."

INSERT 3 (1)

INSERT 3

7. WCAP-8762-P-A, "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," July 1984.

8. Tong, L. S., "Boiling Crisis and Critical Heat Flux," AEC Critical Review Series, TID-25887, 1972.

9. Tong, L. S., "Critical Heat Fluxes on Rod Bundles," in "Two-Phase Flow and Heat Transfer in Rod Bundles," pp. 31-41, American Society of Mechanical Engineers, New York, 1969.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the transient, no control actions are assumed, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained.

The Reactor Trip System setpoints 8 (Ref. 5), together with the settings of the MSSVs (Ref. ~~8~~), provide pressure protection for normal operation and AOOs. The reactor high pressure trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices. 4

More specifically, no credit is taken for operation of the following:

a. Pressurizer power operated relief valves (PORVs);

b. Steam line power operated relief valve (PORV);

c. Steam Dump System;

d. Reactor Control System;

e. Pressurizer Level Control System; or

f. Pressurizer spray valve.

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. ~~The maximum transient pressure allowed in the RCS piping, valves, and fittings under [USAS, Section B31.1 (Ref. 6)] is 120% of design pressure. The most limiting of these two allowances is the 110% of design pressure;~~ therefore, the SL on maximum allowable RCS pressure is 2735 psig. 4

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.3

If the RCS pressure SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. ~~7~~ 6). (4)

2.2.4

If the RCS pressure SL is violated, the Plant Manager, Site Vice President, and Nuclear Safety Review Board (NSRB) shall be notified within 24 hours. The 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to senior management.

2.2.5

If the RCS pressure SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC. the Plant Manager, Site Vice President, and NSRB. ~~This requirement is~~ in accordance with 10 CFR 50.73 (Ref. ~~8~~ 7). A copy of the report shall also be provided to

(4) (A)

2.2.6

If the RCS pressure 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. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 14, "Reactor Coolant Pressure Boundary"; General Design Criterion 15, "Reactor Coolant System Design"; and General Design Criterion 28, "Reactivity Limits."

2. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure."

(continued)

BASES

REFERENCES (continued)

3. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI, IWX-5000, "System Pressure Tests."

4. Title 10, Code of Federal Regulations, Part 100, "Reactor Site Criteria."

5. Watts Bar FSAR, Section 7.2, "Reactor Trip System."

6. USAS B31.1, "Standard Code for Pressure Piping," American Society of Mechanical Engineers, 1967. e

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6 ~~7~~. Title 10, Code of Federal Regulations, Part 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors."

7 ~~8~~. Title 10, Code of Federal Regulations, Part 50.73, "Licensee Event Report System."

8 ~~9~~. Watts Bar FSAR, Section 10.3, "Main Steam Supply System."

JUSTIFICATION FOR CHANGES TO SECTION 3.0

GENERIC JUSTIFICATIONS

A. This change reflects a change to NUREG-1431 submitted to the NRC by the Industry Owners groups via the generic STS change process.

SPECIFIC JUSTIFICATIONS

None

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.

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. (A)

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.

or if directed by the associated ACTIONS (A)

LCO 3.0.3

When an LCO is not met and the associated ACTIONS are not met, ~~or~~ an associated ACTION is not provided, 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 3 within 7 hours;

b. MODE 4 within 13 hours; and

c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

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

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4. (A)

LCO 3.0.4

When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This

(continued)

3.0 LCO APPLICABILITY

LCO 3.0.4 (continued)

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

A

) or variables to be within limits.

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

LCO 3.0.6

When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, additional evaluations and limitations may be required in accordance with Specification 5.8, "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.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1

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

SR 3.0.2

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

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

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

Exceptions to this Specification are stated in the individual Specifications.

SR 3.0.3

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

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. The Completion Times of the Required Actions begin immediately upon expiration of the delay period.

A

(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. The Completion Times of the Required Actions begin immediately upon failure to meet the Surveillance.

A

SR 3.0.4

Entry into a MODE or other specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent passage through or to MODES or other specified conditions in compliance with Required Actions.

entry into

A

the Applicability that are required to comply with ACTIONS.

BASES

LCO 3.0.3 (continued)

assemblies in the fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.13 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.13 of "Suspend movement of irradiated fuel assemblies in the fuel storage pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It precludes placing the unit in a different MODE or other specified condition when the following exist:

a. The requirements of an LCO, in the MODE or other specified condition to be entered, are not met; and

b. Continued noncompliance with these LCO requirements would result in the unit being required to be placed in a MODE or other specified condition ~~in which the LCO does not apply to comply with the Required Actions.~~ other than that desired to be entered. (A)

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

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

(continued)

BASES

LCO 3.0.4 (continued)

or other specified conditions in the Applicability that result from a normal shutdown.

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

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

LCO 3.0.5

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

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

b. The OPERABILITY of other equipment~~.~~; or

C. That variables are within limits.

A

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

(continued)

BASES

SR 3.0.2 (continued)

Therefore, there is a Note in the Frequency stating, "SR 3.0.2 is not applicable."

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

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

SR 3.0.3

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

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

(continued)

BASES

SR 3.0.4 (continued)

INSERT

A

safe operation of the unit. This Specification applies to changes in MODES or other specified conditions in the Applicability associated with unit shutdown as well as startup. ¶ The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS.

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

INSERT

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

JUSTIFICATION FOR CHANGES TO SECTION 3.1

GENERIC JUSTIFICATIONS

A. This change reflects a change to NUREG-1431 submitted to the NRC by the Industry Owners groups via the generic STS change process.

B. Removal of brackets and/or addition of plant specific parameter or information.

C. Correction of typographical error in the Proof and Review TS.

SPECIFIC JUSTIFICATIONS

1. Statement as worded is irrelevant - an idle RCP does nothing. The statement should be revised to indicate that "Startup of an RCP" cannot produce a return to critical.

2. Changes to match LCO applicabilities.

3. The NUREG references the non-proprietary version of the WCAP. The applicable reference for Watts Bar is the proprietary version of the referenced WCAP.

4. GDC 28, not GDC 26, is entitled "Reactivity Limits." GDC 26 refers to "Reactivity Control System Redundancy and Protection."

5. The Required Action is to "initiate" not to "restore" within the 1 hour completion time.

6. The Bases, not the SR, describes the effects to consider when calculating SDM.

7. Change first paragraph of B 3.1.7, Background, to be consistent with Bases 3.1.6 which has the same discussion.

8. Change to make Bases consistent with LCO Required Action.

9. These two paragraphs are difficult to understand as written. The intent of the first paragraph is to show that if the 12 step agreement between ARPIs and demand indicators are met, the demand indicators can be used for position indication. In the second paragraph, LCO 3.1.5 provides the 12 step limit, not the COLR.

10. References to the CFR do not normally include a reference.

11. Consistency with similar presentations throughout the TS.

12. Samarium is included in the Bases for SRs 3.1.1.1 and 3.1.2.1, it is not clear why it is not included here also.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.5.3	Verify rod drop time of each rod, from the fully withdrawn position, is $\leq$ ~~2.4~~ 2.7 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. $T_{avg} \geq 551°F$; and b. All reactor coolant pumps operating.	Prior to reactor criticality after each removal of the reactor head B

BASES

APPLICABLE SAFETY ANALYSES (continued)

power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting MSLB transient, the SDM requirement must also protect against:

a. Inadvertent boron dilution;

b. An uncontrolled rod withdrawal from subcritical or low power condition;

c. Startup of an inactive reactor coolant pump (RCP); and

d. Rod ejection.

Each of these events is discussed below.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life, when critical boron concentrations are highest.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled rod withdrawal transient is terminated by either a high power level trip or a high pressurizer pressure trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

The startup of an inactive RCP will not result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core. The maximum positive reactivity addition that can occur due to an inadvertent RCP start is less than half the minimum required SDM. ~~An~~ idle RCP cannot, therefore, produce a return to power from the hot standby condition.

1 Startup of a

The ejection of a control rod rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a

(continued)

BASES

ACTIONS

A.1 (continued)

boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid storage tank, or the ~~borated~~ refueling water storage tank. The operator should borate with the best source available for the plant conditions. (B)

In determining the boration flow rate, the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of 1% Δk/k must be recovered and a boration flow rate of ~~[10]~~ 35 gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 35 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1% Δk/k. These boration parameters of ~~[10]~~ 35 gpm and ~~[20,000]~~ 6120 ppm represent typical values and are provided for the purpose of offering a specific example. (B)

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

In MODES 1 and 2, SDM is verified by observing that the requirements of LCO 3.1.6 and LCO 3.1.7 are met. In the event that a rod is known to be untrippable, however, SDM verification must account for the worth of the untrippable rod as well as another rod of maximum worth.

In MODES 3 and 4, the SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects:

a. RCS boron concentration;

b. Control bank position;

(continued)

BASES (continued)

LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

The boron dilution accident (Ref. 2) is the most limiting analysis that establishes the SDM value of the LCO. For the boron dilution accident, if the LCO is violated, then the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

APPLICABILITY

In MODE 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 2 with $K_{eff} < 1.0$ and MODES 3 and 4, the SDM requirements are given in LCO 3.1.1, "SHUTDOWN MARGIN (SDM) – $T_{avg} > 200°F$." In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2, MODE 2 with $K_{eff} \geq 1.0$, SDM is ensured by complying with LCO 3.1.6 and LCO 3.1.7. 2

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a ~~highly~~ concentrated solution, such as that normally found in the boric acid storage tank or the ~~borated~~ refueling water storage tank. The operator should borate with the best source available for the plant conditions. B

In determining the boration flow rate the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle, when the boron concentration may

(continued)

BASES

ACTIONS

A.1 (continued)

approach or exceed 2000 ppm. Assuming that a value of 1% Δk/k must be recovered and a boration flow rate of ~~[10]~~ 35 gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 35 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1% Δk/k. These boration parameters of ~~[10]~~ gpm and ~~[20,000]~~ 6120 ppm represent typical values and are provided for the purpose of offering a specific example.

B

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

In MODE 5, the SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Design isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM. This allows time enough for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.2 and SR 3.1.4.3

In similar fashion, the LCO demands that the MTC be less negative than the specified value for EOC full power conditions. This measurement may be performed at any THERMAL POWER, but its results must be extrapolated to the conditions of RTP and all banks withdrawn in order to make a proper comparison with the LCO value. Because the RTP MTC value will gradually become more negative with further core depletion and boron concentration reduction, a 300 ppm SR value of MTC should necessarily be less negative than the EOC LCO limit. The 300 ppm SR value is sufficiently less negative than the EOC LCO limit value to ensure that the LCO limit will be met when the 300 ppm Surveillance criterion is met.

SR 3.1.4.3 is modified by a Note that includes the following requirements:

a. If the 300 ppm Surveillance limit is exceeded, it is possible that the EOC limit on MTC could be reached before the planned EOC. Because the MTC changes slowly with core depletion, the Frequency of 14 effective full power days is sufficient to avoid exceeding the EOC limit.

b. The Surveillance limit for RTP boron concentration of 60 ppm is conservative. If the measured MTC at 60 ppm is more positive than the 60 ppm Surveillance limit, the EOC limit will not be exceeded because of the gradual manner in which MTC changes with core burnup.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 11, "Reactor Inherent Protection."

2. Watts Bar FSAR, Section 15.0, "Accident Analyses."

3. WCAP 9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985. 3

4. Watts Bar FSAR, Section 15.2.1, "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition."

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," and GDC ~~26~~ 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power ~~Plants~~ Reactors" (Ref. 2). 4

C

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately ⅝ inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups

(continued)

BASES

LCO (continued)

some cases a total misalignment from fully withdrawn to fully inserted is assumed.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control rods are bottomed and the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM) − T_{avg} > 200°F," for SDM in MODES 3 and 4, LCO 3.1.2, "Shutdown Margin (SDM)-T_{avg} ≤ 200°F" for SDM in MODE 5, and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

ACTIONS

A.1.1 and A.1.2

When one or more rods are untrippable, there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating boration, ~~and restoring~~ to restore SDM. 5

In this situation, SDM verification must include the worth of the untrippable rod, as well as a rod of maximum worth.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.5.~~3~~2 (Continued)

trippability (OPERABILITY) of the control rod(s) must be made, and appropriate action taken.

C

SR 3.1.5.3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality, after reactor vessel head removal, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature ≥ 551°F to simulate a reactor trip under actual conditions.

This Surveillance is performed during a plant outage, due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 10, "Reactor Design," General Design Criterion ~~26~~ 28, "Reactivity Limits." 4

2. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

3. Watts Bar FSAR, Section 15.0, "Accident Analyses."

4. Watts Bar FSAR, Section 15.2.3, "Rod Cluster Control Assembly Misalignment."

5. Watts Bar FSAR, Section 15.4.2, "Major Secondary System Pipe Rupture."

6. Watts Bar FSAR, Section 15.3.6, "Single ~~RCCA~~ Rod Cluster Control Assembly Withdrawal at Full Power." C

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Shutdown Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth, SDM and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," and GDC ~~26~~ 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved. 4

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. There are four control banks and four shutdown banks. See LCO 3.1.5, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.8, "Rod Position Indication," for position indication requirements.

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally automatically controlled by the Rod Control System, but they can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating). The control banks must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal full power operations. Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor

(continued)

BASES (continued)

ACTIONS

A.1.1, A.1.2 and A.2

When one or more shutdown banks is not within insertion limits, 2 hours is allowed to restore the shutdown banks to within the insertion limits. This is necessary because the available SDM may be significantly reduced, with one or more of the shutdown banks not within their insertion limits. Also, verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by adhering to the control and shutdown bank insertion limits ~~(see LCO 3.1.1).~~ If shutdown banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the Bases for SR 3.1.1.1. 6

The allowed Completion Time of 2 hours provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

B.1

If the shutdown banks cannot be restored to within their insertion limits within 2 hours, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

Verification that the shutdown banks are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown banks will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1 (continued)

shutdown banks are withdrawn before the control banks are withdrawn during a unit startup.

Since the shutdown banks are positioned manually by the control room operator, a verification of shutdown bank position at a Frequency of 12 hours, after the reactor is taken critical, is adequate to ensure that they are within their insertion limits. Also, the 12 hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of shutdown rods.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 10, "Reactor Design," General Design Criterion ~~26~~ 28, "Reactivity Limits." 4

2. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

3. Watts Bar FSAR, Section 15.0, "Accident Analyses."

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Control Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions, assumptions of available and ejected rod worth, SDM, and initial reactivity insertion rate. The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," and GDC ~~26~~, 28 "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

7 4

Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. There are four control banks and four shutdown banks. See LCO 3.1.5, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.8, "Rod Position Indication," for position indication requirements.

The control bank insertion limits are specified in the COLR. An example is provided for information only in Figure B 3.1.7-1. The control banks are required to be at or above the insertion limit lines.

Figure B 3.1.7-1 also indicates how the control banks are moved in an overlap pattern. Overlap is the distance travelled together by two control banks. The predetermined

(continued)

BASES

ACTIONS

A.1.1, A.1.2, A.2, B.1.1, B.1.2, and B.2 (continued)

MARGIN (SDM) – T_{avg} > 200°F") has been upset. If control banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the Bases for SR 3.1.1.1. (6)

Similarly, if the control banks are ~~found to be~~ out of sequence or in the wrong overlap configuration, they must be restored to meet the limits.

Operation beyond the LCO limits is allowed for a short time period in order to take conservative action because the simultaneous occurrence of either a LOCA, loss of flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability.

The allowed Completion Time of 2 hours for restoring the banks to within the insertion, sequence, and overlap limits provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time. (8)

C.1

If Required Actions A.1 and A.2, or B.1 and B.2 cannot be completed within the associated Completion Times, the plant must be brought to MODE 3, where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

(continued)

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BASES (continued)

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 10, "Reactor Design," General Design Criterion ~~26~~ 28, "Reactivity Limits." 4

2. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

3. Watts Bar FSAR, Section 15.2.1, "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition."

4. Watts Bar FSAR, Section 15.2.2, "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal At Power."

5. Watts Bar FSAR, Section 15.2.3, "Rod Cluster Control Assembly Misalignment."

6. Watts Bar FSAR, Section 15.2.4, "Uncontrolled Boron Dilution."

7. Watts Bar FSAR, Section 15.2.5, "Partial Loss of Forced Reactor Coolant Flow."

8. Watts Bar FSAR, Section 15.2.13, "Accidental Depressurization of the Main Steam System."

9. Watts Bar FSAR, Section 15.3.4, "Complete Loss of Forced Reactor Coolant Flow."

10. Watts Bar FSAR, Section 15.3.6, "Single Rod Cluster Control Assembly Withdrawal At Full Power."

11. Watts Bar FSAR, Section 15.4.2.1, "Major Rupture of Main Steam Line."

12. Watts Bar FSAR, Section 15.4.4, "Single Reactor Coolant Pump Locked Rotor."

13. Watts Bar FSAR, Section 15.4.6, "Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)."

BASES (continued)

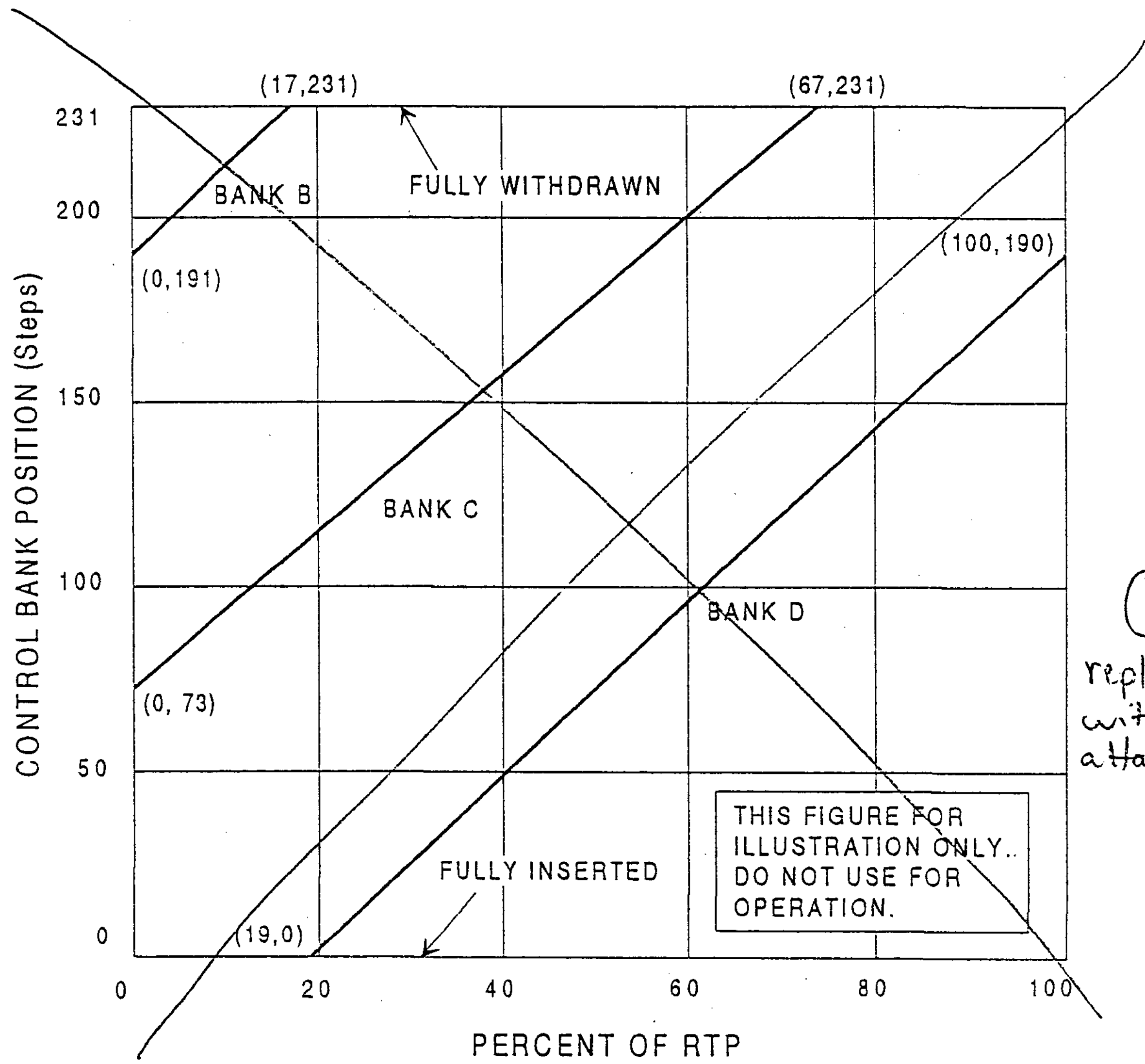

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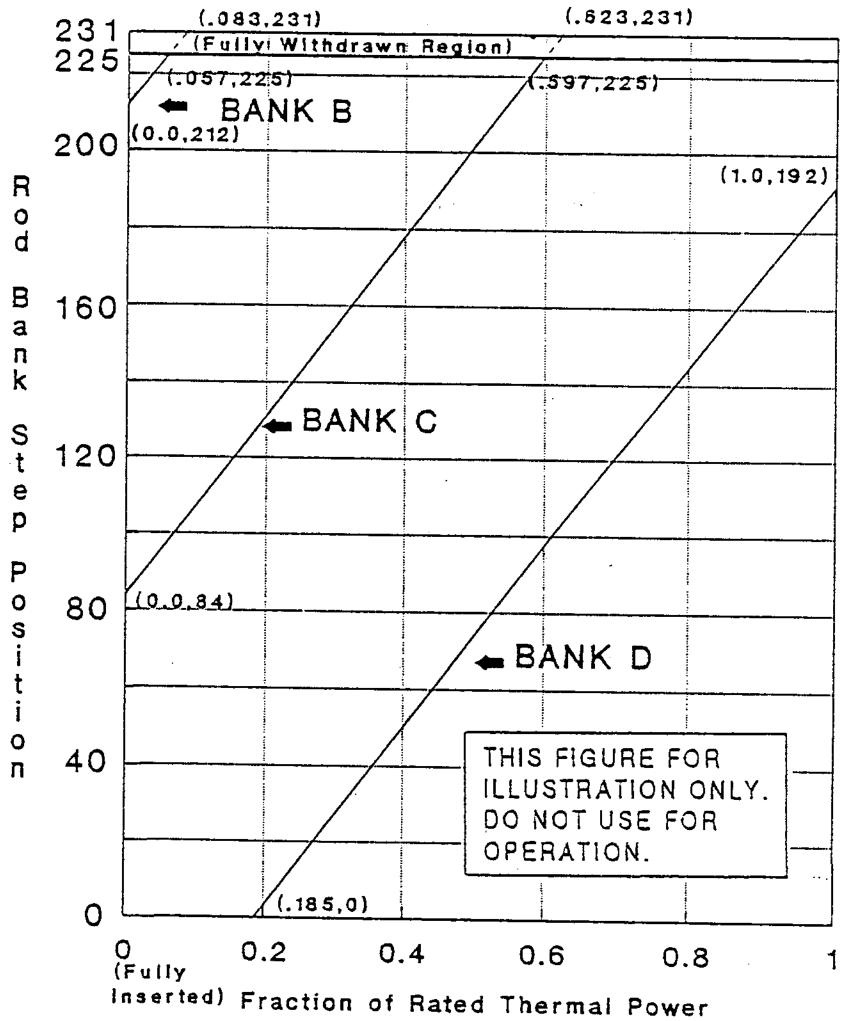
Figure B 3.1.7-1 (page 1 of 1)
Control Bank Insertion vs. Percent RTP

BASES (continued)



* Fully withdrawn region shall be the condition where shutdown and control banks are at a position within the interval of ≥ 225 and ≤ 231 steps withdrawn, inclusive.

Figure B 3.1.7-1 (page 1 of 1)
Control Bank Insertion vs. Percent RTP

BASES

APPLICABLE SAFETY ANALYSES (continued)

monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions.

The control rod position indicator channels satisfy Criterion 2 of the NRC Policy Statement. The control rod position indicators monitor control rod position, which is an initial condition of the accident.

LCO

LCO 3.1.8 specifies that the ARPI System and the Bank Demand Position Indication System be OPERABLE for all control rods. For the control rod position indicators to be OPERABLE requires meeting the SR of the LCO and the following:

a. The ARPI System indicates within 12 steps of the group step counter demand position as required by LCO 3.1.5, "Rod Group Alignment Limits;"

b. For the ARPI System there are no failed coils; and

c. The Bank Demand Indication System has been calibrated either in the fully inserted position or to the ARPI System.

The 12 step agreement limit between the Bank Demand Position Indication System and the ARPI System ~~is within the limit, indicating~~ indicates that the Bank Demand Position Indication System is adequately calibrated, ~~for measurement of~~ and can be used for indication of the control rod bank position. 9

A deviation of less than the allowable limit, given in ~~the COLR~~ LCO 3.1.5, in position indication for a single control rod, ensures high confidence that the position uncertainty of the corresponding control rod group is within the assumed values used in the analysis (that specified control rod group insertion limits). 9

These requirements ensure that control rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged. OPERABILITY of the position indicator channels ensures that

(continued)

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.9.2

Verification of the Power Range Neutron Flux — High trip setpoints within 8 hours prior to initiation of the PHYSICS TESTS will ensure that the RTS is properly set to perform PHYSICS TESTS.

SR 3.1.9.3

The performance of SR 3.2.1.1 and SR 3.2.2.1 measures the core $F_Q(Z)$ and the $F^N_{\Delta H}$, respectively. If the requirements of these LCOs are met, the core has adequate protection from exceeding its design limits, while other LCO requirements are suspended. The Frequency of 12 hours is based on operating experience and the practical amount of time that it may take to run an incore flux map and calculate the hot channel factors.

SR 3.1.9.4

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. Reactor Coolant System (RCS) boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration; and
f. Samarium Concentration;
g ~~f~~. Design isothermal temperature coefficient (ITC). 12

Using the ITC accounts for Doppler reactivity in the calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident without the required SDM.

(continued)

BASES (continued)

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," ~~1988.~~ 9 10

2. Title 10, Code of Federal Regulations, Part 50.59, "Changes, Tests, and Experiments."

3. Regulatory Guide 1.68, Revision 2, "Initial Test Programs for Water-Cooled Nuclear Power Plants," ~~U.S. Nuclear Regulatory Commission,~~ August 1978. 11

4. ANSI/ANS-19.6.1-1985, "Reload Startup PHYSICS TESTS for Pressurized Water Reactors," American National Standards Institute, December 13, 1985.

5. WCAP-927~~3~~2-N~~P~~-A, "Westinghouse Reload Safety Evaluation Methodology Report," July 1985. 3

6. Watts Bar FSAR, Section 14.2, "Test Program."

7. WCAP-11618, "MERITS Program - Phase II, Task 5, Criteria Application," dated November 1987, including Addendum 1, April 1989.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.10.3

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration; and 12

f. Samarium Concentration;

g ~~f~~. Design isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

2. Title 10, Code of Federal Regulations, Part 50.59, "Changes, Tests and Experiments."

3. Regulatory Guide 1.68, Revision 2, "Initial Test Programs for Water-Cooled Nuclear Power Plants," August 1978.

4. ANSI/ANS-19.6.1-1985, "Reload Startup Physics Tests for Pressurized Water Reactors," American National Standards Institute, December 13, 1985.

(continued)

BASES

REFERENCES (continued)

5. WCAP-927~~3~~2-~~N~~P-A, "Westinghouse Reload Safety Evaluation Methodology Report," July 1985. 3

6. WCAP-11618, "MERITS Program - Phase II, Task 5, Criteria Application," dated November 1987, including Addendum 1, April 1989.

JUSTIFICATION FOR CHANGES TO SECTION 3.2

GENERIC JUSTIFICATIONS

A. This change reflects a change to NUREG-1431 submitted to the NRC by the Industry Owners groups via the generic STS change process.

B. Removal of brackets and/or addition of plant specific parameter or information.

C. Correction of typographical error in the Proof and Review TS.

SPECIFIC JUSTIFICATIONS

1. Although this information is stated in the Bases, Watts Bar prefers to have a note added to the surveillance requirement to clearly specify this required adjustment to the measured value before comparison with the calculated limit.

2. Definition as written is not completely correct as can be verified by the discussion on page B 3.2-3 of the Bases, second to last paragraph.

3. The statement of "fission energy" is not correct. The 280 cal/gm limit of RG 1.77 also includes thermal energy. The appropriate discussion is "energy deposition to the fuel" and is similar to the discussion in Bases 3.1.1.

4. The discussion is not complete since for Watts Bar, two correlations are actually used for different core regions resulting in two DNBR values. The discussion of critical heat flux correlations and the resulting DNBR values has been added to Bases Safety Analysis Section and it is not considered necessary to repeat that discussion in the Background section also.

5. The NUREG references the non-proprietary version of the WCAP. The applicable reference for Watts Bar is the proprietary version of the referenced WCAP.

6. This change was necessary for the Bases to match the applicability requirements of the LCO.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify $F^N_{\Delta H}$ is within limits specified in the COLR. 1 ------ NOTE ------ The measured $F^N_{\Delta H}$ shall be increased by 4% to account for measurement uncertainty. ------------	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> 31 EFPD thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4 The QPTR shall be ≤ 1.02.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. QPTR not within limit.	A.1	Reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	2 hours
	<u>AND</u>		
	A.2	Perform SR 3.2.4.1	12 hours <u>AND</u> Once per 12 hours thereafter
	<u>AND</u>		
	A.3	Perform SR 3.2.1.1 and SR 3.2.2.1.	24 hours <u>AND</u> Once per 7 days thereafter
C	<u>AND</u>		
	A.4	Reevaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	<u>AND</u>		

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$)

BASES

BACKGROUND

The purpose of the limits on the values of $F_Q(Z)$ is to limit the local (i.e., pellet) peak power density. The value of $F_Q(Z)$ varies along the axial height (Z) of the core.

2

adjusted for uncertainty

$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_Q(Z)$ is a measure of the peak fuel pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT TILT POWER RATIO (QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.7, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

$F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

$F_Q(Z)$ is measured periodically using the incore detector system. These measurements are generally taken with the core at or near steady state conditions.

Using the measured three dimensional power distributions, it is possible to derive a measured value for $F_Q(Z)$. However, because this value represents a steady state condition, it does not include the variations in the value of $F_Q(Z)$ that are present during nonequilibrium situations, such as load following.

To account for these possible variations, the steady state value of $F_Q(Z)$ is adjusted by an elevation dependent factor that accounts for the calculated worst case transient conditions.

Core monitoring and control under nonsteady state conditions are accomplished by operating the core within the limits of

(continued)

BASES

BACKGROUND (continued)

the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition;

c. During an ejected rod accident, the fission energy deposition input to the fuel must not exceed 280 cal/gm (Ref. 2); and 3

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(Z)$ satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

The Heat Flux Hot Channel Factor, $F_Q(Z)$, shall be limited by the following relationships:

$$F_Q(Z) \le \frac{CFQ}{P} K(Z) \qquad \text{for } P > 0.5$$

$$F_Q(Z) \le \frac{CFQ}{0.5} K(Z) \qquad \text{for } P \le 0.5$$

where: CFQ is the $F_Q(Z)$ limit at RTP provided in the COLR,

K(Z) is the normalized $F_Q(Z)$ as a function of core height provided in the COLR, and

$$P = \frac{\text{THERMAL POWER}}{\text{RTP}}$$

For this facility, the actual values of CFQ and K(Z) are given in the COLR; however, CFQ is normally a number on the order of 2.4, and K(Z) is a function that looks like the one provided in Figure B 3.2.1-1.

For Relaxed Axial Offset Control operation, $F_Q(Z)$ is approximated by $F^C_Q(Z)$ and $F^W_Q(Z)$. Thus, both $F^C_Q(Z)$ and $F^W_Q(Z)$ must meet the preceding limits on $F_Q(Z)$.

An $F^C_Q(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the measured value $(F^M_Q(Z))$ of $F_Q(Z)$. Then,

$$F^C_Q(Z) = F^M_Q(Z)\ [1.0815]$$

where [1.0815] is a factor that accounts for fuel manufacturing tolerances and flux map measurement uncertainty.

B

$F^C_Q(Z)$ is an excellent approximation for $F_Q(Z)$ when the reactor is at the steady state power at which the incore flux map was taken.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

for extended operation) is achieved. In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days without verification of $F^C_Q(Z)$ and $F^W_Q(Z)$. The Frequency condition is not intended to require verification of these parameters after every 10% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which F_Q was last measured.

SR 3.2.1.1

Verification that $F^C_Q(Z)$ is within its specified limits involves increasing $F^M_Q(Z)$ to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F^C_Q(Z)$. Specifically, $F^M_Q(Z)$ is the measured value of $F_Q(Z)$ obtained from incore flux map results and $F^C_Q(Z) = F^M_Q(Z)\ (1.0815)$ (Ref. 4). $F^C_Q(Z)$ is then compared to its specified limits.

B

The limit with which $F^C_Q(Z)$ is compared varies inversely with power above 50% RTP and directly with a function called K(Z) provided in the COLR.

Performing this Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F^C_Q(Z)$ limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.

If THERMAL POWER has been increased by $\geq$ 10% RTP since the last determination of $F^C_Q(Z)$, another evaluation of this factor is required 12 hours after achieving equilibrium conditions at this higher power level (to ensure that $F^C_Q(Z)$ values are being reduced sufficiently with power increase to stay within the LCO limits).

The Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the plant is operated in accordance with the Technical Specifications (TS).

(continued)

BASES

REFERENCES (continued)

3. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," GDC 26, "Reactivity Control System Redundancy and Capability."

4. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.

B

BASES

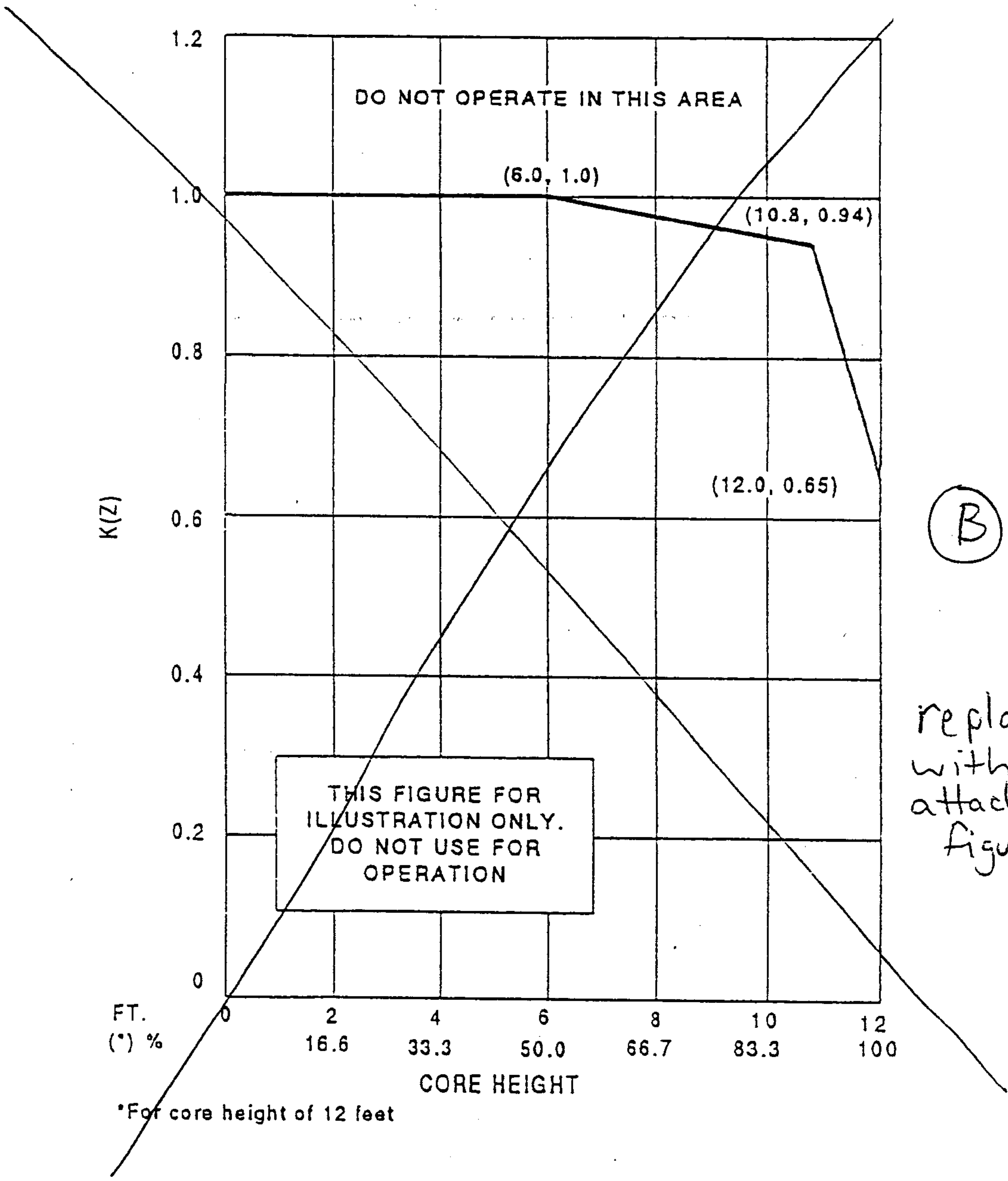

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Figure B 3.2.1-1 (page 1 of 1)
K(Z) - Normalized $F_Q(Z)$ as a Function of Core Height

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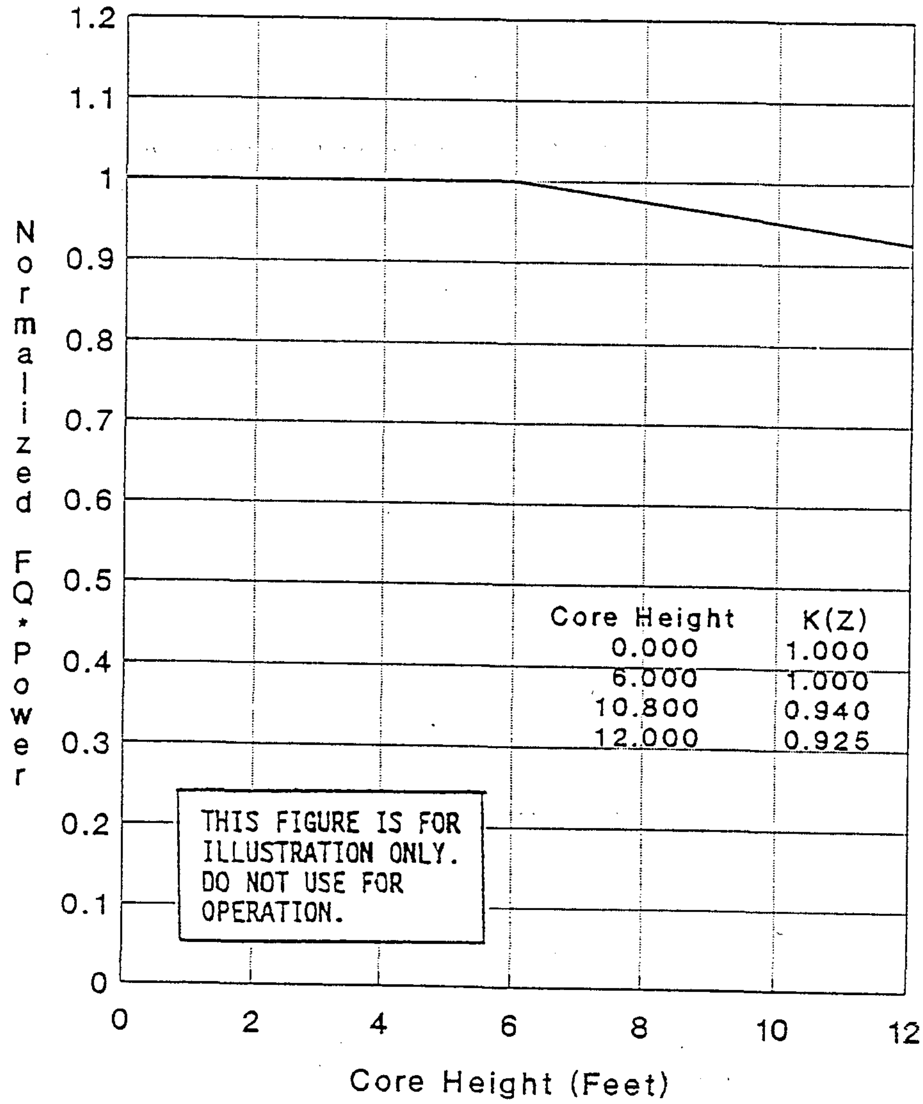


Figure B 3.2.1-1 (page 1 of 1)
K(Z) - Normalized $F_Q(Z)$ as a Function of Core Height

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

$F^N_{\Delta H}$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod.

$F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. $F^N_{\Delta H}$ typically increases with control bank insertion and typically decreases with fuel burnup.

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine $F^N_{\Delta H}$. This factor is calculated at least every 31 EFPD. However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which address directly and continuously measured process variables.

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB ~~and is met by limiting the minimum local DNB heat flux ratio to [1.3] using the WRB1 CHF correlation~~. All DNB limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements. 4

for the hottest fuel rod in the core

(continued)

BASES

BACKGROUND (continued)

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition;

b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F;

3 c. During an ejected rod accident, the ~~fission~~ energy deposition ~~input~~ to the fuel must not exceed 280 cal/gm (Ref. 1); and C

9 ( )

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, the Reactor Coolant System flow and $F^N_{\Delta H}$ are the core parameters of most importance. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum DNBR to the 95/95 DNB criterion of [1.3] using the WRB1 CHF correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB. 4

INSERT 1

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use

(continued)

INSERT 1

...local DNB heat flux ratio to a value which satisfies the 95/95 criterion for the DNB correlation used. In the heated rod spans above the first mixing vane grid, the WRB-1 correlation with a DNBR criterion of 1.17 is applied. The W-3 correlation with a DNBR criterion of 1.3 is applied in the heated region below the first mixing vane grid. Application of these criteria provides assurance...

BASES

ACTIONS

A.1.1 (continued)

However, if power is reduced below 50% RTP, Required Action A.3 requires that another determination of $F^N_{\Delta H}$ must be done prior to exceeding 50% RTP, prior to exceeding 75% RTP, and within 24 hours after reaching or exceeding 95% RTP. In addition, Required Action A.2 is performed if power ascension is delayed past 24 hours.

A.1.2.1 and A.1.2.2

If the value of $F^N_{\Delta H}$ is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce THERMAL POWER to < 50% RTP in accordance with Required Action A.1.2.1 and reduce the Power Range Neutron Flux – High to ≤ 55% RTP in accordance with Required Action A.1.2.2. Reducing RTP to < 50% RTP increases the DNB margin and does not likely cause the DNBR limit to be violated in steady state operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 4 hours for Required Action A.1.2.1 is consistent with those allowed for in Required Action A.1.1 and provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time. The Completion Times of 4 hours for Required Actions A.1.1 and A.~~2~~1.2.1 are not additive. C

The allowed Completion Time of 8 hours to reset the trip setpoints per Required Action A.1.2.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may inadvertently trip the Reactor Protection System.

A.2

Once the power level has been reduced to < 50% RTP per Required Action ~~A.1.1~~ A.1.2.1, an incore flux map (SR 3.2.2.1) must be obtained and the measured value of $F^N_{\Delta H}$ verified not to C

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

This SR is modified by a Note which states that

The value of $F^N_{\Delta H}$ is determined by using the movable incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions. The measured value of $F^N_{\Delta H}$ must be multiplied by 1.04 to account for measurement uncertainty before making comparisons to the $F^N_{\Delta H}$ limit. 1

After each refueling, $F^N_{\Delta H}$ must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that $F^N_{\Delta H}$ limits are met at the beginning of each fuel cycle.

The 31 EFPD Frequency is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this Frequency is short enough that the $F^N_{\Delta H}$ limit cannot be exceeded for any significant period of operation.

REFERENCES

1. Regulatory Guide 1.77, Rev. 0, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized water Reactors," May 1974.

2. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," GDC 26, "Reactivity Control System Redundancy and Capability."

3. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

BASES

LCO (continued)

of the control banks is in response to temperature deviations resulting from manual operation of the Chemical and Volume Control System to change boron concentration or from power level changes.

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors (Ref. 3). Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detectors in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as %Δ flux.

The AFD limits are provided in the COLR. Figure B 3.2.3-1 shows typical RAOC AFD limits. The AFD limits for RAOC do not depend on the target flux difference. However, the target flux difference may be used to minimize changes in the axial power distribution.

Violating this LCO on the AFD could produce unacceptable consequences if a Condition 2, 3, or 4 event occurs while the AFD is outside its specified limits.

APPLICABILITY

The AFD requirements are applicable in MODE 1 at or above 50% RTP when the combination of THERMAL POWER and core peaking factors are of primary importance in safety analysis. 6

For AFD limits developed using RAOC methodology, the value of the AFD does not affect the limiting accident consequences with THERMAL POWER < 50% RTP and for lower operating power MODES.

ACTIONS

A.1

As an alternative to restoring the AFD to within its specified limits, Required Action A.1 requires a THERMAL POWER reduction to < 50% RTP. This places the core in a condition for which the value of the AFD is not important in the applicable safety analyses. A Completion Time of 30 minutes is reasonable, based on operating experience, to reach 50% RTP without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The AFD is monitored on an automatic basis using the unit process computer, which has an AFD monitor alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels is outside its specified limits.

This Surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits and is consistent with the status of the AFD monitor alarm. With the AFD monitor alarm inoperable, the AFD is monitored every hour to detect operation outside its limit. The Frequency of 1 hour is based on operating experience regarding the amount of time required to vary the AFD, and the fact that the AFD is closely monitored. With the AFD monitor alarm OPERABLE, the Surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed.

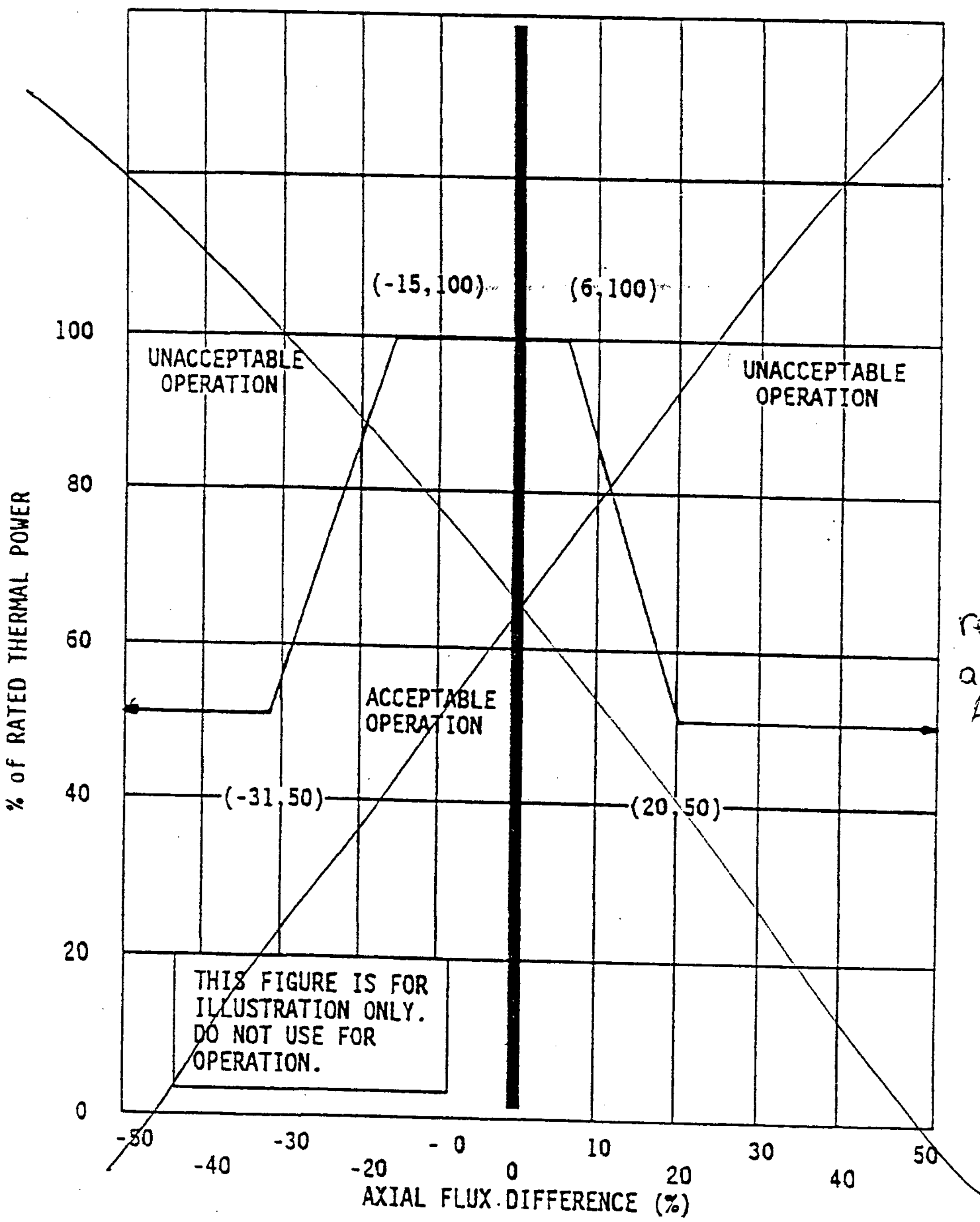
REFERENCES

1. WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974.

2. R. W. Miller et al., "Relaxation of Constant Axial Offset Control: F_Q Surveillance Technical Specification," WCAP-~~10217(NP)~~ 10216-P-A, June 1983. (5)

3. Watts Bar FSAR, Section 7.7, "Control Systems Not Required for Safety."

BASES



(B)

replace with attached figure

Figure B 3.2.3-1 (page 1 of 1)
AXIAL FLUX DIFFERENCE Acceptable Operation Limits
as a Function of RATED THERMAL POWER

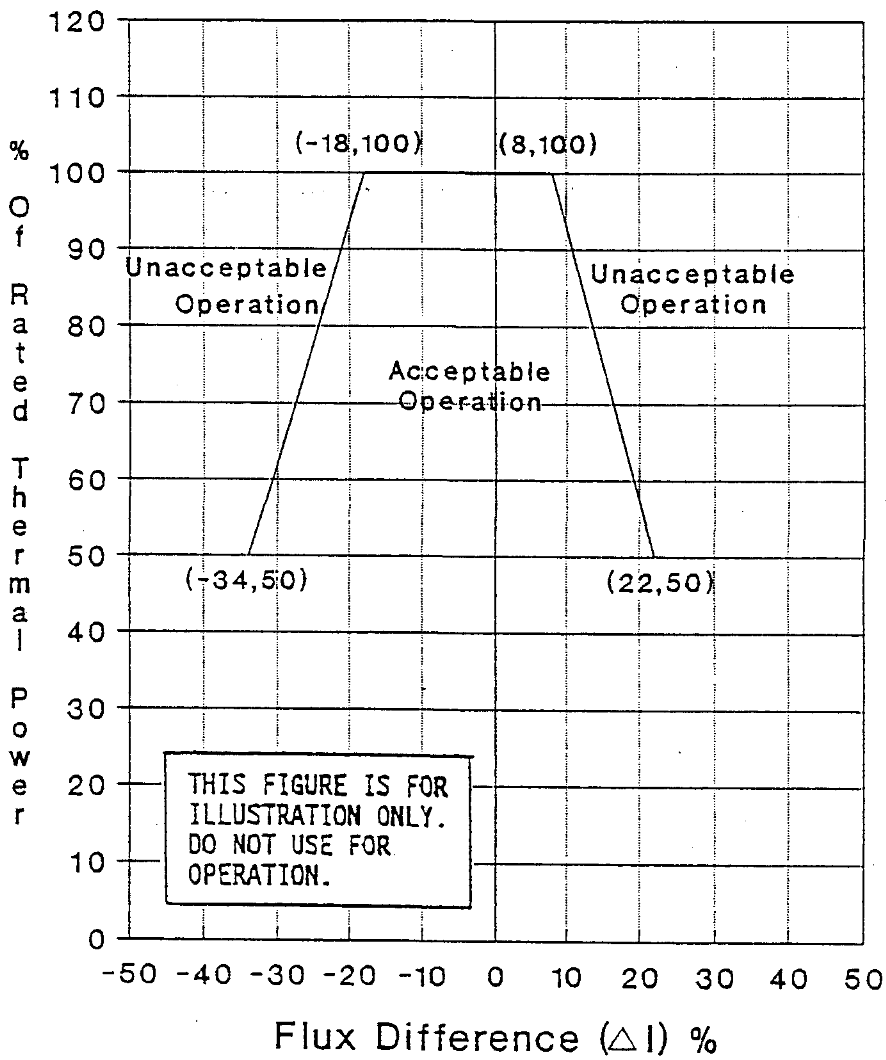


Figure B 3.2.3-1 (page 1 of 1)
AXIAL FLUX DIFFERENCE Acceptable Operation Limits
as a Function of RATED THERMAL POWER

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.7, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the ~~fission~~ energy deposition ~~input~~ to the fuel must not exceed 280 cal/gm (Ref. 2); and 3

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot

(continued)

JUSTIFICATION FOR CHANGES TO SECTION 3.4

GENERIC JUSTIFICATIONS

A. This change reflects a change to NUREG-1431 submitted to the NRC by the Industry Owners groups via the generic STS change process.

B. Removal of brackets and/or addition of plant specific parameter or information.

C. Correction of typographical error in the Proof and Review TS.

SPECIFIC JUSTIFICATIONS

1. The discussion is not complete since for Watts Bar, two correlations are actually used for different core regions resulting in two DNBR values. The discussion of critical heat flux correlations and the resulting DNBR values has been added to Bases Safety Analysis Section for B 3.2.2, and it is not necessary to repeat that discussion here.

2. Since the calculation revision may change, it not appropriate to provided it here.

3. The August 27, 1992 submittal added additional sections to Reference 1, but did not add any new reference numbers in the text. As presented, there are no References 2 and 3.

4. Change to be consistent with LCO.

5. Terminology is RCP, not RCS pumps at Watts Bar.

6. Change first paragraph to be consistent with B 3.1.6 and to make it technically correct.

7. The RHR system is connected to the RCS piping, not the reactor vessel.

8. During a planned heatup, the function is circulation, not heat removal.

9. The sentence wording is grammatically incorrect.

10. The setpoint for Watts Bar is $\pm$ 1%, not 3% which is already discussed in the LCO section of the Bases. Since this paragraph does not apply, it should be deleted.

11. These statements are incorrect. The ASME code allows a 2 year frequency for seat leakage testing. Additionally, the CFR invokes the ASME code, but does not specify a frequency. The 18 month frequency is based on the refueling cycle, not a specified frequency in the CFR or ASME code.

12. Consistency with similar presentations.

13. The ASME code now references the OM standard, not IWV, in the 1989 edition.

14. The LTOP analysis does not permit any Safety Injection Pump (SIP) to be OPERABLE during the COMS applicability, therefore, this requirement has been added to the LCO and Bases allowing no SIPs to be capable of RCS injection consistent with the similar generic application for the HPI pump in the NUREG.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure ≥ [2216] psig;

b. RCS average temperature ≤ [592.9] °F; and

c. RCS total flow rate ≥ 390,000 gpm.

B

APPLICABILITY: MODE 1.

-----------------------------NOTE-----------------------------
Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
SR 3.4.1.1	Verify pressurizer pressure is ≥ [2216] psig.	12 hours	B
SR 3.4.1.2	Verify RCS average temperature is ≤ [592.9]°F.	12 hours	B
SR 3.4.1.3	Verify RCS total flow rate is ≥ 390,000 gpm.	12 hours	
SR 3.4.1.4	-------------------NOTE------------------- Not required to be performed until 24 hours after ≥ [90]% RTP. -- Verify by precision heat balance that RCS total flow rate is ≥ 390,000 gpm.	18 months	B

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Three pressurizer safety valves shall be OPERABLE with lift settings ≥ 2460 psig and ≤ 2510 psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures > 310°F.

---------------------------NOTE---------------------------
The lift settings are not required to be within the LCO limits during MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 54 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. OR Two or more pressurizer safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 with any ~~all~~ RCS cold leg temperatures ≤ 310°F.	6 hours 12 hours

A

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Cold Overpressure Mitigation System (COMS)

(14)

LCO 3.4.12 A COMS System shall be OPERABLE with a maximum of one charging pump and no Safety Injection pump capable of injecting into the RCS and the accumulators isolated and either a or b below.

a. Two RCS relief valves, as follows:

1. Two power operated relief valves (PORVs) with lift settings within the limits specified in the PTLR, or

2. One PORV with a lift setting within the limits specified in the PTLR and the RHR suction relief valve with a setpoint ≥ 436.5 psig and ≤ 463.5 psig.

b. The RCS depressurized and an RCS vent of capable of relieving > 475 gpm water flow.

APPLICABILITY: MODE 4 when any RCS cold leg temperature is ≤ 310°F,
MODE 5,
MODE 6 when the reactor vessel head is on.

---------------------------NOTE---------------------------
Accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.

14

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Safety Injection pumps capable of injecting into the RCS.	A.1 Initiate action to verify no Safety Injection pumps are capable of injecting into the RCS.	Immediately
B ~~A~~. Two or more charging pumps capable of injecting into the RCS.	B ~~A~~.1 --------NOTE--------- Two charging pumps may be capable of injecting into the RCS during pump swap operation for ≤ 15 minutes. --------------------- Initiate action to verify a maximum of one charging pump is capable of injecting into the RCS.	 Immediately
C ~~B~~. An accumulator not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	C ~~B~~.1 Isolate affected accumulator.	1 hour
D ~~C~~. Required Action and associated Completion Time of Condition ~~B~~ C not met.	D ~~C~~.1 Increase RCS cold leg temperature to > 310°F. <u>OR</u> D ~~C~~.2 Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	12 hours 12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
E ~~D~~. One required RCS relief valve inoperable in MODE 4.	E ~~D~~.1 Restore required RCS relief valve to OPERABLE status.	7 days	14
F ~~E~~. One required RCS relief valve inoperable in MODE 5 or 6.	F ~~E~~.1 Restore required RCS relief valve to OPERABLE status.	24 hours	14
G ~~F~~. Two required RCS relief valves inoperable. OR Required Action and associated Completion Time of Condition A, B ~~C~~, D, E, or F ~~E~~ not met. OR COMS ~~LTOP~~ System inoperable for any reason other than Condition A, B, C, D, E, or F ~~E~~.	G ~~F~~.1 Depressurize RCS and establish RCS vent.	8 hours	14 14 C 14

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.12.~~1~~2 Verify a maximum of one charging pump is capable of injecting into the RCS.	12 hours
SR 3.4.12.1 Verify no Safety Injection pumps are capable of injecting into the RCS.	12 hours

14

(continued)

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.4.12.~~2~~3	Verify each accumulator is isolated.	12 hours	14
SR 3.4.12.~~3~~4	-------------------NOTE-------------------- Only required to be performed when complying with LCO 3.4.12.b. --- Verify RCS vent open.	12 hours for unlocked open vent paths <u>AND</u> 31 days for locked open vent paths	14
SR 3.4.12.~~4~~5	Verify PORV block valve is open for each required PORV.	72 hours	14
SR 3.4.12.~~5~~6	Verify both RHR suction isolation valves are locked open with operator power removed for the required RHR suction relief valve.	31 days	14

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.12.~~6~~ 7	-------------------NOTE------------------- Not required to be met until 12 hours after decreasing RCS cold leg temperature to ≤ 310°F. -- Perform a COT on each required PORV, excluding actuation.	31 days
SR 3.4.12.~~7~~ 8	Perform CHANNEL CALIBRATION for each required PORV actuation channel.	18 months

14

14

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.15.1	Perform CHANNEL CHECK of the required containment atmosphere radioactivity monitor.	12 hours
SR 3.4.15.2	Perform COT of the required containment atmosphere radioactivity monitor.	92 days
SR 3.4.15.3	Perform CHANNEL CALIBRATION of the required containment pocket sump level monitor.	18 months
SR 3.4.15.4	Perform CHANNEL CALIBRATION of the required containment atmosphere radioactivity monitor.	18 months

B

BASES

APPLICABLE SAFETY ANALYSES (continued)

result in meeting the DNBR criterion ~~of ≥ [1.3]~~. This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed for include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.7, "Control Bank Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)." 1

The pressurizer pressure limit of 2216 psig and the RCS average temperature limit of 592.2°F correspond to analytical limits of 2204 psig and 594.7°F used in the safety analyses, with allowance for measurement uncertainty. B

The RCS DNB parameters satisfy Criterion 2 of the NRC Policy Statement.

LCO

This LCO specifies limits on the monitored process variables – pressurizer pressure, RCS average temperature, and RCS total flow rate – to ensure the core operates within the limits assumed in the safety analyses. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

RCS total flow rate contains a measurement error of 2.5% based on performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi, which might not be detected, could bias the result from the precision heat balance in a nonconservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi raises the nominal flow measurement allowance to 2.6% for no fouling. B

Any fouling that might bias the flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling. B

(continued)

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.2 (continued)

potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.3

The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed flow instrumentation. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions.

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a precision calorimetric heat balance once every 18 months allows the installed RCS flow instrumentation to be calibrated and verifies the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate.

The Frequency of 18 months reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance.

B

This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the unit in the best condition for performing the SR. The Note states that the SR is not required to be performed until 24 hours after ≥ 90% RTP. This exception is appropriate since the heat balance requires the plant to be at a minimum of 90% RTP to obtain the stated RCS flow accuracies. The Surveillance shall be performed within 24 hours after reaching 90% RTP.

REFERENCES

1. Watts Bar FSAR, Section 15.0, "Accident Analysis,"
2. Watts Bar FSAR, Section 15.2, "Normal Operation and Anticipated Transients," and
3. Watts Bar FSAR, Section 15.3.4, "Complete Loss Of Forced Reactor Coolant Flow."

3

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND

This LCO is based upon meeting several major considerations before the reactor can be made critical and while the reactor is critical.

The first consideration is moderator temperature coefficient (MTC), LCO 3.1.4, "Moderator Temperature Coefficient (MTC)." In the transient and accident analyses, the MTC is assumed to be in a range from slightly positive to negative and the operating temperature is assumed to be within the nominal operating envelope while the reactor is critical. The LCO on minimum temperature for criticality helps ensure the plant is operated consistent with these assumptions.

The second consideration is the protective instrumentation. Because certain protective instrumentation (e.g., excore neutron detectors) can be affected by moderator temperature, a temperature value within the nominal operating envelope is chosen to ensure proper indication and response while the reactor is critical.

The third consideration is the pressurizer operating characteristics. The transients and accident analyses assume that the pressurizer is within its normal startup and operating range (i.e., saturated conditions and steam bubble present). It is also assumed that the RCS temperature is within its normal expected range for startup and power operation. Since the density of the water, and hence the response of the pressurizer to transients, depends upon the initial temperature of the moderator, a minimum value for moderator temperature within the nominal operating envelope is chosen. C

The fourth consideration is that the reactor vessel is above its minimum nil ductility reference temperature when the reactor is critical.

APPLICABLE SAFETY ANALYSES

Although the RCS minimum temperature for criticality is not itself an initial condition assumed in Design Basis Accidents (DBAs), the closely aligned temperature for hot

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

zero power (HZP) is a process variable that is an initial condition of DBAs, such as the rod cluster control assembly (RCCA) withdrawal, RCCA ejection, and main steam line break accidents performed at zero power that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

All low power safety analyses assume initial RCS loop temperatures ≥ the HZP temperature of 557°F (Ref. 1). The minimum temperature for criticality limitation provides a small band, 6°F, for critical operation below HZP. This band allows critical operation below HZP during plant startup and does not adversely affect any safety analyses since the MTC is not significantly affected by the small temperature difference between HZP and the minimum temperature for criticality.

The RCS minimum temperature for criticality satisfies Criterion 2 of the NRC Policy Statement.

LCO

Compliance with the LCO ensures that the reactor will not be made or maintained critical ($k_{eff} \geq 1.0$) at a temperature less than a small band below the HZP temperature, which is assumed in the safety analysis. Failure to meet the requirements of this LCO may produce initial conditions inconsistent with the initial conditions assumed in the safety analysis.

APPLICABILITY

MODE

In MODE~~S~~ 1 and 2, with $k_{eff} \geq 1.0$, LCO 3.4.2 is applicable since the reactor can only be critical ($k_{eff} \geq 1.0$) in these MODES. 4

The special test exception of LCO 3.1.10, "MODE 2 PHYSICS TESTS Exceptions," permits PHYSICS TESTS to be performed at ≤ 5% RTP with RCS loop average temperatures slightly lower than normally allowed so that fundamental nuclear characteristics of the core can be verified. In order for nuclear characteristics to be accurately measured, it may be necessary to operate outside the normal restrictions of this LCO. For example, to measure the MTC at beginning of cycle, it is necessary to allow RCS loop average temperatures to fall below $T_{no\ load}$, which may cause RCS loop average

(continued)

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BASES

APPLICABILITY (continued)	temperatures to fall below the temperature limit of this LCO.
ACTIONS	A.1 If the parameters that are outside the limits cannot be restored, 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 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period. The allowed time is reasonable, based on operating experience, to reach MODE 3 in an orderly manner and without challenging plant systems.
SURVEILLANCE REQUIREMENTS	SR 3.4.2.1 RCS loop average temperature is required to be verified at or above 551°F within 15 minutes prior to achieving criticality and every 30 minutes thereafter when the T_{avg} - T_{ref} deviation alarm is not reset and any RCS loop T_{avg} < 557°F. The 15 minute time period is long enough to allow the operator to adjust temperatures or delay criticality so the LCO will not be violated, thereby providing assurance that the safety analyses are not violated. The Note modifies the SR. When any RCS loop average temperature is < 557°F and the T_{avg} - T_{ref} deviation alarm is alarming, RCS loop average temperatures could fall below the LCO requirement without additional warning. The SR to verify RCS loop average temperatures every 30 minutes is frequent enough to prevent the inadvertent violation of the LCO.
REFERENCES	1. Watts Bar FSAR, Section 15.0, "Accident Analysis."

4

BASES

ACTIONS

C.1 and C.2 (continued)

pre-analyzed transients in the stress analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the PTLR limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This SR is modified by a Note that only requires this SR to be performed during system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

REFERENCES

1. TVA Calculation WBN-MTB-027 R2, "Pressure-Temperature Limits Based on Reg Guide 1.99 R2 for submittal to NRC." 2

(continued)

BASES

REFERENCES (continued)

2. Title 10, Code of Federal Regulations, Part 50, Appendix G, "Fracture Toughness Requirements."

3. ASME Boiler and Pressure Vessel Code, Section III, Appendix G, "Protection Against Non-Ductile Failure."

4. ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," July 1982.

5. Title 10, Code of Federal Regulations, Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

6. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

7. ASME Boiler and Pressure Vessel Code, Section XI, Appendix E, "Evaluation of Unanticipated Operating Events."

BASES

APPLICABLE SAFETY ANALYSES (continued)

met. For those conditions when the Rod Control System is not capable of rod withdrawal, two RCS loops are required to be OPERABLE, but only one RCS loop is required to be in operation to be consistent with MODE 3 accident analyses.

Failure to provide decay heat removal may result in challenges to a fission product barrier. The RCS loops are part of the primary success path that functions or actuates to prevent or mitigate a Design Basis Accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. RCS Loops — MODE 3 satisfy Criterion 3 of the NRC Policy Statement. A ¶

LCO

The purpose of this LCO is to require that at least two RCS loops be OPERABLE. In MODE 3 with the RTBs in the closed position and Rod Control System capable of rod withdrawal, two RCS loops must be in operation. Two RCS loops are required to be in operation in MODE 3 with RTBs closed and Rod Control System capable of rod withdrawal due to the postulation of a power excursion because of an inadvertent control rod withdrawal. The required number of RCS loops in operation ensures that the Safety Limit criteria will be met for all of the postulated accidents.

With the RTBs in the open position, or the CRDMs de-energized, the Rod Control System is not capable of rod withdrawal; therefore, only one RCS loop in operation is necessary to ensure removal of decay heat from the core and homogenous boron concentration throughout the RCS. An additional RCS loop is required to be OPERABLE to ensure that safety analyses limits are met.

The Note permits all RCPs to be de-energized for ≤ 1 hour per 8 hour period. The purpose of the Note is to perform tests that are designed to validate various accident analyses values. One of these tests is validation of the pump coastdown curve used as input to a number of accident analyses including a loss of flow accident. This test is generally performed in MODE 3 during the initial startup testing program, and as such should only be performed once. If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values of the coastdown curve must be revalidated by conducting the test again. The 1 hour time period

(continued)

BASES

5

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.5.3

Verification that the required RCPs ~~RCS pump(s)~~ are OPERABLE ensures that safety analyses limits are met. The requirement also ensures that an additional RCP ~~RCS pumps~~ can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to the required RCPs.

REFERENCES

None.

BASES (continued)

ACTIONS

A.1

If one required RCS loop is inoperable and two RHR loops are inoperable, redundancy for heat removal is lost. Action must be initiated to restore a second RCS or RHR loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1

If one required RHR loop is OPERABLE and in operation and there are no RCS loops OPERABLE, an inoperable RCS or RHR loop must be restored to OPERABLE status to provide a redundant means for decay heat removal.

If the parameters that are outside the limits cannot be restored, the plant must be brought to MODE 5 within 24 hours. Bringing the plant to MODE 5 is a conservative action with regard to decay heat removal. With only one RHR loop OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining RHR loop, it would be safer to initiate that loss from MODE 5 (≤ 200°F) rather than MODE 4 (200 to 350°F). The Completion Time of 24 hours is a reasonable time, based on operating experience, to reach MODE 5 from MODE 4 in an orderly manner and without challenging plant systems.

C.1 and C.2

If no loop is OPERABLE or in operation, except during ~~conditions permitted by Note 1 in the LCO~~ section, all ~~operations involving a reduction~~ of RCS boron concentration must be suspended and action to restore one RCS or RHR loop to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and the margin to criticality must not be reduced in this type of operation. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation. 4

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops — MODE 5, Loops Filled

BASES

6

BACKGROUND

In MODE 5 with the RCS loops the removal of filled, the primary function of the reactor coolant is ~~to remove~~ decay heat and transfer of this heat to either the steam generator~~s~~ (SG~~s~~) ~~or~~ Secondary side Coolant or the Component Cooling water via the residual heat removal (RHR) heat exchangers. While the principal means for decay heat removal is via the RHR System, the SGs are specified as a backup means for redundancy. Even though the SGs cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary water. As long as the SG secondary side water is at a lower temperature than the reactor coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

7

In MODE 5 with RCS loops filled, the reactor coolant is circulated by means of two RHR loops connected to the RCS ~~reactor vessel~~, each loop containing an RHR heat exchanger, an RHR pump, and appropriate flow and temperature instrumentation for control, protection, and indication. One RHR pump circulates the water through the RCS at a sufficient rate to prevent boric acid stratification.

The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR loop for decay heat removal and transport. The flow provided by one RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide redundancy for heat removal.

The LCO provides for redundant paths of decay heat removal capability. The first path can be an RHR loop that must be OPERABLE and in operation. The second path can be another OPERABLE RHR loop or maintaining two SGs with secondary side water levels above 10% narrow range to provide an alternate method for decay heat removal.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation.

RCS Loops — MODE 5 (Loops Filled) have been identified in the NRC Policy Statement as important contributors to risk reduction.

LCO

The purpose of this LCO is to require that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE or two SGs with secondary side water level ≥ 10% narrow range. One RHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE to meet single failure considerations. However, if the standby RHR loop is not OPERABLE, an acceptable alternate method is two SGs with their secondary side water levels ≥ 10% narrow range. Should the operating RHR loop fail, the SGs could be used to remove the decay heat.

Note 1 allows one RHR loop to be inoperable for a period of up to 2 hours, provided that the other RHR loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible.

Note 2 requires that the secondary side water temperature of each SG be ≤ 50°F above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) with an RCS cold leg temperature ≤ 310°F. This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

Note 3 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the ~~heat removal~~ RCS circulation function provided by the RHR loops. 8

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

9

If no required RHR loops are OPERABLE or in operation, except during conditions permitted by Note 1, all operations involving a reduction of RCS boron concentration must be suspended and ~~requires initiation of~~ action ~~to~~ immediately ~~start restoration of~~ an RHR loop to OPERABLE status and operation. Boron dilution requires forced circulation for uniform dilution, and the margin to criticality must not be reduced in this type of operation. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must continue until one loop is restored to OPERABLE status and operation.

to restore

must be initiated

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This SR requires verification every 12 hours that one loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

SR 3.4.8.2

Verification that the required number of pumps are OPERABLE ensures that additional pumps can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pumps. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

None.

BASES

BACKGROUND (continued)

a loss of single phase natural circulation and decreased capability to remove core decay heat.

APPLICABLE SAFETY ANALYSES

In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. Safety analyses performed for lower MODES are not limiting. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present.

Safety analyses presented in the FSAR (Ref. 1) do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure.

The maximum pressurizer water level limit satisfies Criterion 2 of the NRC Policy Statement. Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 2), is the reason for providing an LCO.

LCO

The LCO requirement for the pressurizer to be OPERABLE with a water volume ≤ 1656 cubic feet, which is equivalent to 92%, ensures that a steam bubble exists. Limiting the LCO maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions.

The LCO requires two groups of OPERABLE pressurizer heaters, each with a capacity ≥ 150 kW. The minimum heater capacity required is sufficient to maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer insulation. By maintaining the pressure

(continued)

BASES

ACTIONS

A.1 (continued)

coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with ~~all~~ any RCS Cold Leg Temperatures ≤ 310°F within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. With ~~all~~ any RCS cold leg temperatures at or below 310°F, overpressure protection is provided by the COMS System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves. A

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

~~The pressurizer safety valve setpoint is ± [3]% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.~~ 10

REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, NB 7614.3, 1971 Edition through Summer 1973.

(continued)

ACTIONS

A.1 (continued)

small break LOCA. For these reasons, the block valve may be closed but the Action requires power be maintained to the valve. This Condition is only intended to permit operation of the plant for a limited period of time not to exceed the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problematic condition. Normally, the PORVs should be available for automatic mitigation of overpressure events and should be returned to OPERABLE status prior to entering startup (MODE 2).

A

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that has shown that minor problems can be corrected or closure accomplished in this time period.

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

If one PORV is inoperable and not capable of being manually cycled, it must be either restored or isolated by closing the associated block valve and removing the power to the associated block valve. The Completion Times of 1 hour are reasonable, based on challenges to the PORVs during this time period, and provide the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is at least one PORV that remains OPERABLE, an additional 72 hours is provided to restore the inoperable PORV to OPERABLE status. If the PORV cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D.

C.1 and C.2

If one block valve is inoperable, then it is necessary to either restore the block valve to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in manual control. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Cold Overpressure Mitigation System (COMS)

BASES

BACKGROUND

The COMS controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. The PTLR provides the maximum allowable actuation logic setpoints for the power operated relief valves (PORVs) and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the COMS MODES.

The reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only while shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits.

This LCO provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires ~~deactivating~~ all but one charging pump and isolating the accumulators. The pressure relief capacity requires either two redundant RCS relief valves or a depressurized RCS and an RCS vent of sufficient size. One RCS relief valve or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

all Safety Injection pumps and

14

A

incapable of injection into the RCS

(continued)

BASES

BACKGROUND (continued)

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the safety injection (SI) actuation circuits blocked. Due to the lower pressures in the COMS MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve. ~~and, if needed, until the centrifugal charging pump is actuated by SI.~~

A

If conditions require the use of more than one charging pump or Safety Injection pump for makeup in the event of loss of inventory, then pumps can be made available through manual actions.

The COMS for pressure relief consists of two PORVs with reduced lift settings, or one PORV and the Residual Heat Removal (RHR) suction relief valve, or a depressurized RCS and an RCS vent of sufficient size. Two RCS relief valves are required for redundancy. One RCS relief valve has adequate relieving capability to keep from overpressurization for the required coolant input capability.

PORV Requirements

As designed for the COMS, each PORV is signaled to open if the RCS pressure approaches a limit determined by the COMS actuation logic. The COMS actuation logic monitors both RCS temperature and RCS pressure and determines when a condition not acceptable in the PTLR limits is approached. The wide range RCS temperature indications are auctioneered to select the lowest temperature signal.

The lowest temperature signal is processed through a function generator that calculates a pressure limit for that temperature. The calculated pressure limit is then compared with the indicated RCS pressure from a wide range pressure channel. If the indicated pressure meets or exceeds the calculated value, a PORV is signaled to open.

The PTLR presents the PORV setpoints for COMS. The setpoints are normally staggered so only one valve opens during a low temperature overpressure transient. Having the setpoints of both valves within the limits in the PTLR ensures that the Reference 1 limits will not be exceeded in any analyzed event.

When a PORV is opened in an increasing pressure transient, the release of coolant will cause the pressure increase to slow and reverse. As the PORV releases coolant, the RCS

(continued)

BASES

BACKGROUND

PORV Requirements (continued)

pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

RHR Suction Relief Valve Requirements

During COMS MODES, the RHR System is operated for decay heat removal and low pressure letdown control. Therefore, the RHR suction isolation valves are open in the piping from the RCS hot leg to the inlet header of the RHR pumps. While these valves are open, the RHR suction relief valve is exposed to the RCS and is able to relieve pressure transients in the RCS.

The RHR suction isolation valves must be open to make the RHR suction relief valve OPERABLE for RCS overpressure mitigation. Autoclosure interlocks are not permitted to cause the RHR suction isolation valves to close. The RHR suction relief valve is a spring loaded, bellows type water relief valve with pressure tolerances and accumulation limits established by Section III of the American Society of Mechanical Engineers (ASME) Code (Ref. 3) for Class 2 relief valves.

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at containment ambient pressure in an RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting COMS mass or heat input transient, and maintaining pressure below the P/T limits. The required vent capacity may be provided by one or more vent paths.

B

For an RCS vent to meet the flow capacity requirement, it requires removing a pressurizer safety valve, removing a PORV~~'s internals~~, and disabling its block valve in the open position, or opening the pressurizer manway. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

Heat Input Type Transients (continued)

The following are required during the COMS MODES to ensure that mass and heat input transients do not occur, which either of the COMS overpressure protection means cannot handle:

a. ~~Deactivating~~ Rendering all Safety Injection pumps and all but one charging pump ~~OPERABLE~~ incapable of injection; 14 A

b. Deactivating the accumulator discharge isolation valves in their closed positions; and

c. Disallowing start of an RCP if secondary temperature is more than 50°F above primary temperature in any one loop. LCO 3.4.6, "RCS Loops – MODE 4," and LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled," provide this protection. B

The Reference 4 analyses demonstrate that either one RCS relief valve or the depressurized RCS and RCS vent can maintain RCS pressure below limits when

no Safety Injection pumps and

only one centrifugal charging pump is actuated ~~by SI~~. Thus, the LCO allows only one charging pump OPERABLE during the COMS MODES. Since neither one RCS relief valve nor the RCS vent can handle ~~a~~ 14 A

the pressure transient induced from accumulator injection when RCS temperature is low,

~~full SI actuation~~, the LCO also requires the accumulators ~~isolated~~. A

isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

The isolated accumulators must have their discharge valves closed and the valve power supply breakers fixed in their open positions. The analyses show the effect of accumulator discharge is over a narrower RCS temperature range (175°F and below) than that of the LCO (310°F and below). Fracture mechanics analyses established the temperature of COMS Applicability at 310°F.

The consequences of a small break loss of coolant accident (LOCA) in COMS MODE 4 conform to 10 CFR 50.46 and 10 CFR 50, Appendix K (Refs. 5 and 6) requirements by having a maximum of one charging pump OPERABLE and SI actuation enabled.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the limit shown in the PTLR. The setpoints are derived by analyses that model the performance of the COMS, assuming the limiting COMS transient of ~~SI actuation of~~ one centrifugal charging pump. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensures the Reference 1 P/T limits will be met.

(A) injecting into the RCS.

(14) no safety injection pumps and only

The PORV setpoints in the PTLR will be updated when the revised P/T limits conflict with the COMS analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Revised limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," discuss these examinations.

The PORVs are considered active components. Thus, the failure of one PORV is assumed to represent the worst case, single active failure.

RHR Suction Relief Valve Performance

The RHR suction relief valve does not have variable pressure and temperature lift setpoints like the PORVs. Analyses must show that the RHR suction relief valve with a setpoint at or between 436.5 psig and 463.5 psig will pass flow greater than that required for the limiting COMS transient while maintaining RCS pressure less than the P/T limit curve. Assuming all relief flow requirements during the limiting COMS event, the RHR suction relief valve will maintain RCS pressure to within the valve rated lift setpoint, plus an accumulation ≤ [30]% of the rated lift setpoint. (B)

The RHR suction relief valve inclusion and location within

(continued)

BASES

APPLICABLE SAFETY ANALYSES

RHR Suction Relief Valve Performance (continued)

the RHR System does not allow it to meet single failure criteria when spurious RHR suction isolation valve closure is postulated. Also, as the RCS P/T limits are decreased to reflect the loss of toughness in the reactor vessel materials due to neutron embrittlement, the RHR suction relief valves must be analyzed to still accommodate the design basis transients for COMS.

The RHR suction relief valve is considered an active component. Thus, the failure of this valve is assumed to represent the worst case single active failure.

RCS Vent Performance

With the RCS depressurized, analyses show a vent capable of relieving > 475 gpm water flow is capable of mitigating the allowed COMS overpressure transient. The capacity of 475 gpm is greater than the flow of the limiting transient for the COMS configuration, ~~SI actuation~~ with one centrifugal charging pump OPERABLE, maintaining RCS pressure less than the maximum pressure on the P/T limit curve. A

B Three ~~Two~~ vent flow paths have been identified in the RCS which could serve as pressure release (vent) paths. With one Safety or PORV removed, the open line could serve as one vent path. The pressurizer manway could serve as an alternative vent path with the manway cover removed. ~~Both~~ These flow paths are capable of discharging 475 gpm at low pressure in the RCS. Thus, any ~~either~~ one of the ~~two~~ openings can be used for relieving the pressure to prevent violating the P/T limits.

The RCS vent size will be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure.

The COMS satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

This LCO requires that the COMS is OPERABLE. The COMS is OPERABLE when the minimum coolant input and pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

14

no Safety Injection pumps and only

To limit the coolant input capability, the LCO requires one charging pump capable of injecting into the RCS and all accumulator discharge isolation valves closed ~~and immobilized.~~ LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," defines SI actuation OPERABILITY for the COMS MODE 4 small break LOCA.

when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

A

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

a. Two RCS relief valves, as follows:

1. Two OPERABLE PORVs; or

A PORV is OPERABLE for COMS when its block valve is open, its lift setpoint is set to the limit required by the PTLR and testing proves its ability to open at this setpoint, and motive power is available to the valve and its control circuit.

2. One OPERABLE PORV and the OPERABLE RHR suction relief valve; or

An RHR suction relief valve is OPERABLE for COMS when both RHR suction isolation valves are open, its setpoint is at or between 436.5 psig and 463.5 psig, and testing has proven its ability to open at this setpoint.

b. A depressurized RCS and an RCS vent.

An RCS vent is OPERABLE when capable of relieving > 475 gpm water flow.

Each of these methods of overpressure prevention is capable of mitigating the limiting COMS transient.

(continued)

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BASES (continued)

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is ≤ 310°F, in MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above 310°F. When the reactor vessel head is off, overpressurization cannot occur.

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 above 310°F.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows operator action to mitigate the event.

The Applicability is modified by a Note stating that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.

ACTIONS

A.1 and B.1

or any Safety Injection pumps (14)

With two or more charging pumps or any Safety Injection pumps capable of injecting into the RCS, RCS overpressurization is possible.

To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.

Required Action ~~A~~ B.1 is modified by a Note that permits two charging pumps capable of RCS injection for ≤ 15 minutes to allow for pump swaps. (14)

(continued)

BASES

ACTIONS (continued)

C D D
~~B~~.1, ~~C~~.1, and ~~C~~.2 (14)

An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

If isolation is needed and cannot be accomplished in 1 hour, Required Action D ~~C~~.1 and Required Action D ~~C~~.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to > 310°F, an accumulator pressure of 661 psig cannot exceed the COMS limits if the accumulators are fully injected. Depressurizing the accumulators below the COMS limit from the PTLR also gives this protection. (14)

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring COMS is not likely in the allowed times.

E ~~D~~.1 (14)

In MODE 4 when any RCS cold leg temperature is ≤ 310°F, with one required RCS relief valve inoperable, the RCS relief valve must be restored to OPERABLE status within a Completion Time of 7 days. Two RCS relief valves are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

The Completion Time considers the facts that only one of the RCS relief valves is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low.

F ~~E~~.1 (14)

The consequences of operational events that will overpressurize the RCS are more severe at lower temperature (Ref. 7). Thus, with one of the two RCS relief valves inoperable in MODE 5 or in MODE 6 with the head on, the Completion Time to restore two valves to OPERABLE status is 24 hours.

(continued)

BASES

ACTIONS

F ~~E~~.1 (continued) (14)

The Completion Time represents a reasonable time to investigate and repair several types of relief valve failures without exposure to a lengthy period with only one OPERABLE RCS relief valve to protect against overpressure events.

G ~~F~~.1 (14)

The RCS must be depressurized and a vent must be established within 8 hours when:

a. Both required RCS relief valves are inoperable; or

b. A Required Action and associated Completion Time of Condition A, ~~C~~ B, D, E or ~~E~~ F is not met; or (14)

c. The COMS is inoperable for any reason other than Condition A, B, C, D, E or ~~E~~ F. (14)

This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

The Completion Time considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.~~2~~3 (14)

(A) incapable of injecting into the RCS

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, no Safety Injection pumps and all but one charging pump are verified ~~deactivated with power removed~~ and the accumulator discharge isolation valves are verified closed and locked out.

(A) INSERT

The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.

(continued)

INSERT

The Safety Injection pumps and charging pump are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control. Alternatively to this, at least two independent means such that a single failure or single action will not result in an injection into the RCS can be used. This is accomplished through the pump control switch being placed in "pull to lock" and at least one valve in the discharge flow path being closed.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.~~3~~4 14

The RCS vent capable of relieving > 475 gpm water flow is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a vent path that cannot be locked.

b. Once every 31 days for a vent path that is locked, sealed, or secured in position. A removed Safety or PORV fits this category. B

The passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12b.

SR 3.4.12.~~4~~5 14

The PORV block valve must be verified open every 72 hours to provide the flow path for each required PORV to perform its function when actuated. The valve must be remotely verified open in the main control room. This Surveillance is performed if the PORV satisfies the LCO.

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required removed, and the manual operator is not required locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure situation.

The 72 hour Frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify that the PORV block valve remains open.

SR 3.4.12.~~5~~6 14

The required RHR suction relief valve shall be demonstrated OPERABLE by verifying both RHR suction isolation valves are open and by testing it in accordance with the Inservice

(continued)

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.~~5~~6 (continued) 14

Testing Program. This Surveillance is only performed if the RHR suction relief valve is being used to satisfy this LCO.

Every 31 days both RHR suction isolation valves are verified locked open, with power to the valve operator removed, to ensure that accidental closure will not occur. The "locked open" valves must be locally verified in the open position with the manual actuator locked. The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve position.

SR 3.4.12.~~6~~7 14

Performance of a COT is required within 12 hours after decreasing RCS temperature to ≤ 310°F and every 31 days on each required PORV to verify and, as necessary, adjust its lift setpoint. The COT will verify the setpoint is within the PTLR allowed maximum limits in the PTLR. PORV actuation could depressurize the RCS and is not required.

The 12 hour Frequency considers the unlikelihood of a low temperature overpressure event during this time.

A Note has been added indicating that this SR is required to be met 12 hours after decreasing RCS cold leg temperature to ≤ 310°F. The COT cannot be performed until in the COMS MODES when the PORV lift setpoint can be reduced to the COMS setting. The test must be performed within 12 hours after entering the COMS MODES.

SR 3.4.12.~~7~~8 14

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required every 18 months to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix G, "Fracture Toughness Requirements."

(continued)

BASES

ACTIONS (continued)

degraded the ability of the interconnected system to perform its safety function.

A.1 and A.2

The flow path must be isolated. Required Actions A.1 and A.2 are modified by a Note that the valve used for isolation must meet the same leakage requirements as the PIVs and must be within the RCPB.

Required Action A.1 requires that the isolation with one valve must be performed within 4 hours. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the affected system if leakage cannot be reduced. The 4 hour Completion Time allows the actions and restricts the operation with leaking isolation valves.

The 72 hour Completion Time after exceeding the limit allows for the restoration of the leaking PIV to OPERABLE status. This timeframe considers the time required to complete this Action and the low probability of a second valve failing during this period.

B.1 and B.2

C

If leakage cannot be reduced with the system isolated, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The allowed Completion Times are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action A.2 is required to verify that leakage is below the specified limit and to identify each leaking valve. The

(continued)

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

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Testing is to be performed every 18 months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The 18 month Frequency is ~~required in~~ Consistent with 10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within the Frequency allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. 9), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complementary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

performed in the previous 9 months. In addition, this Surveillance is not required to be performed on the RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Section 50.2, "Definitions—Reactor Coolant Pressure Boundary."

2. Title 10, Code of Federal Regulations, Part 50, Section 50.55a, "Codes and Standards," Subsection (c), "Reactor Coolant Pressure Boundary."

3. Title 10, Code of Federal Regulations, Part 50, Appendix A, Section V, "Reactor Containment," General Design Criterion 55, "Reactor Coolant Pressure Boundary Penetrating Containment."

4. U.S. Nuclear Regulatory Commission (NRC), "Reactor Safety Study—An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," Appendix V, WASH-1400 (NUREG-75/014), October 1975.

5. U.S. NRC, "The Probability of Intersystem LOCA: Impact Due to Leak Testing and Operational Changes," NUREG-0677, May 1980.

6. Watts Bar FSAR, Section 3.9, "Mechanical Systems and Components" (Table 3.9-17).

7. ASME Boiler and Pressure Vessel Code, Section XI, ~~Subsection IWV, "Inservice Testing of Valves in Nuclear Power Plants," paragraph IWV-3423(e).~~ 13

8. Title 10, Code of Federal Regulations, Part 50, Section 50.55a, "Codes and Standards," Subsection (g), "Inservice Inspection Requirements."

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.15.3 and SR 3.4.15.4

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 18 months is a typical refueling cycle and considers channel reliability. Again, operating experience has proven that this Frequency is acceptable.

REFERENCES

1. 10 CFR 50, Appendix A, General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary."
2. Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," ~~U.S. Nuclear Regulatory Commission.~~ Rev. 0, May 1973. 12
3. Watts Bar FSAR, Section 5.2.7, "RCPB Leakage Detection Systems."

JUSTIFICATION FOR CHANGES TO SECTION 3.5

GENERIC JUSTIFICATIONS

C. Correction of typographical error in the Proof and Review TS.

SPECIFIC JUSTIFICATIONS

1. Delete "either." There is only one limit to restore the parameter within. The use of "either" implies an option between parameters.

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.5 Seal Injection Flow

LCO 3.5.5 Reactor coolant pump seal injection flow shall be ≤ 40 gpm with centrifugal charging pump discharge header pressure ≥ 2430 psig and the pressurizer level control valve full open.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Seal injection flow not within limit.	A.1 Adjust manual seal injection throttle valves to give a flow within limit with centrifugal charging pump discharge header pressure ≥ 2480 psig and the pressurizer level control valve full open.	4 hours 2430 C
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

BASES

APPLICABILITY (continued)

MODES 1, 2, 3, and 4. The seal injection flow limit is not applicable for MODE 4 and lower, however, because high seal injection flow is less critical as a result of the lower initial RCS pressure and decay heat removal requirements in these MODES. Therefore, RCP seal injection flow must be limited in MODES 1, 2, and 3 to ensure adequate ECCS performance.

ACTIONS

A.1

With the seal injection flow exceeding its limit, the amount of charging flow available to the RCS may be reduced. Under this Condition, action must be taken to restore the flow to below its limit. The operator has 4 hours from the time the flow is known to be above the limit to correctly position the manual valves and thus be in compliance with the accident analysis. The Completion Time minimizes the potential exposure of the plant to a LOCA with insufficient injection flow and ensures that seal injection flow is either restored to or below its limit. This time is conservative with respect to the Completion Times of other ECCS LCOs; it is based on operating experience and is sufficient for taking corrective actions by operations personnel. 1

B.1 and B.2

When the Required Actions cannot be completed within the required Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions in an orderly manner and without challenging plant systems.

(continued)

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JUSTIFICATION FOR CHANGES TO SECTION 3.6

GENERIC JUSTIFICATIONS

A. This change reflects a change to NUREG-1431 submitted to the NRC by the Industry Owners groups via the generic STS change process.

B. Removal of brackets and/or addition of plant specific parameter or information.

C. Correction of typographical error in the Proof and Review TS.

SPECIFIC JUSTIFICATIONS

1. The note refers to both a and b of the LCO, therefore, the word "temperature" should be plural.

2. The discussion refers to inservice "tests" not inspections.

3. According to Air Products "Limits of Flammability of Gases and Vapors," the flammability limit of hydrogen is 4.0 percent.

4. Values changed based on the TVA calculation referenced and operational limitations imposed.

5. Watts Bar does not have this system.

6. The sentence as worded doesn't read well. Proposed change to be consistent with the same paragraph in Bases 3.6.7.

7. The Bases for SR 3.6.8.2 does not discuss the frequency, therefore, a discussion has been proposed. Additionally, the discussion on required actions is not clear and is inappropriate for the bases for the SR.

8. The Watts Bar Containment Spray System does not perform a containment atmosphere cleanup function (the sodium tetraborate in the ice condenser does this), therefore, these GDCs do not apply to this system.

9. The Watts Bar EGTS design relies upon an initial negative pressure established by the annulus vacuum fans in the shield building. As long as the negative pressure is maintained, then the accident analysis shows that the annulus pressure will not go positive after EGTS initiation as long as the fans reach rated flow within 18 seconds. Because of this design, the time required to reestablish the negative pressure setpoint is not relevant since the pressure will not go positive. The more important parameter is the time required for the fan to achieve rated flow. Therefore, the acceptance criteria was deleted from the Shield Building negative pressure test and added to the EGTS fan flow test. This discussion was added to the Bases.

10. The Containment Spray System at Watt Bar does not perform an iodine removal function.

11. The Air Return Fans do not provide a heat removal capability, rather they provide flow through the ice condenser which removes heat.

12. Added discussion for basis of frequency consistent with other drawdown tests and the flow test in SR 3.6.9.4.

13. These tests are a function of the Type C test which are done in SR 3.6.1.1 and the frequency allowed by Appendix J (up to 24 months) is also applicable. This is consistent with STS Rev. 4a which allowed up to 24 months for the combined leakage rate testing.

14. There are some valves in the ERCW and CSS which close in approximately 66 seconds plus signal delays, however, because they are in liquid filled systems, they have been evaluated to have no impact on the DBA analysis.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. ---------NOTE--------- Only applicable to penetration flow paths with only one containment isolation valve and a closed system. ----------------------	C.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	4 hours
One or more penetration flow paths with one containment isolation valve inoperable.	AND		
	C.2	--------NOTE--------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. --------------------	
		Verify the affected penetration flow path is isolated.	Once per 31 days
D. Shield building bypass leakage not within limit.	D.1	Restore leakage within limit.	4 hours
E. One or more penetration flow paths with one or more containment purge valves not within purge valve leakage limits.	E.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	24 hours
	AND		
			(continued)

B

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. (continued)	E.2	--------NOTE--------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. ---------------------	
		Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside containment AND Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment
	AND		
	E.3	Perform SR 3.6.3.5 for the resilient seal purge valves closed to comply with Required Action E.1.	Once per 92 days
F. Required Action and associated Completion Time not met.	F.1	Be in MODE 3.	6 hours
	AND		
	F.2	Be in MODE 5.	36 hours

(B)

SURVEILLANCE		FREQUENCY
C SR 3.6.3.~~11~~ 8	--------NOTE-------- Results shall be evaluated against acceptance criteria of SR 3.6.1.1 in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions. -------------------- Verify the combined leakage rate for all shield building bypass leakage paths is $\leq 0.25\ L_a$ when pressurized to ≥ 15.0 psig.	--- NOTE --- SR 3.0.2 is not applicable ---------- 13 ~~18 months~~ In accordance with 10CFR50, Appendix J, as modified by approved exemptions

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be:

a. ≥ 85°F and ≤ 110°F for the containment upper compartment, and

b. ≥ 100°F and ≤ 120°F for the containment lower compartment.

-----------------------------NOTE-----------------------------
The minimum containment average air temperatures in MODES 2, 3, and 4 may be reduced to 60°F. 1

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limits.	A.1 Restore containment average air temperature to within limits.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	B.2 Be in MODE 5.	36 hours

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray System

LCO 3.6.6 Two containment spray trains and two residual heat removal (RHR) spray trains shall be OPERABLE.

---NOTE---
The RHR spray train is not required in MODE 4.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One containment spray train inoperable.	A.1	Restore containment spray train to OPERABLE status.	72 hours
B. One RHR spray train inoperable.	B.1	Restore RHR spray train to OPERABLE status.	72 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 5.	84 hours

C

3.6 CONTAINMENT SYSTEMS

3.6.7 Hydrogen Recombiners

LCO 3.6.7 Two hydrogen recombiners shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One hydrogen recombiner inoperable.	A.1	--------NOTE-------- LCO 3.0.4 is not applicable. -------------------- Restore hydrogen recombiner to OPERABLE status.	30 days
B. Two hydrogen recombiners inoperable.	B.1	Verify by administrative means that the hydrogen control function is maintained.	1 hour AND ~~Every~~ Once per 12 hours thereafter
	AND		
	B.2	Restore one hydrogen recombiner to OPERABLE status.	7 days
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	6 hours

A

3.6 CONTAINMENT SYSTEMS

3.6.8 Hydrogen Mitigation System (HMS)

LCO 3.6.~~10~~ 8 Two HMS trains shall be OPERABLE. C

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One HMS train inoperable.	A.1	Restore HMS train to OPERABLE status.	7 days
	<u>OR</u>		
	A.2	Perform SR 3.6.8.1 on the OPERABLE train.	Once per 7 days
B. One containment region with no OPERABLE hydrogen ignitor.	B.1	Restore one hydrogen ignitor in the affected containment region to OPERABLE status.	7 days
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.9.3	Verify each EGTS train actuates on an actual or simulated actuation signal.	18 months
SR 3.6.9.4	Verify each EGTS train produces a flow rate ~~is~~ ≥ 3600 and ≤ 4400 cfm within 18 seconds of a start signal.	18 months on a STAGGERED TEST BASIS

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.10.3	Verify, with the ARS fan not operating, each ARS fan damper opens when ≤ ~~150~~ [TBD] in-lb is applied to the counterweight.	92 days

B

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.15.1	Verify annulus negative pressure is ~~0.5~~ inches water gauge with respect to the atmosphere. equal to or more negative than -5	12 hours
SR 3.6.15.2	Verify the door in each access opening is closed, except when the access opening is being used for normal transient entry and exit.	31 days
SR 3.6.15.3	Verify shield building structural integrity by performing a visual inspection of the exposed interior and exterior surfaces of the shield building.	During shutdown for SR 3.6.1.1 Type A tests
SR 3.6.15.4	Verify each Emergency Gas Treatment System train with final flow ≥ 3600 and ≤ 4400 cfm produces an annulus pressure equal to or more negative than ~~0.5~~ -1.036 inch water gauge with respect to ~~EL. 772 mechanical equipment room~~ the atmosphere and with an inleakage of ≤ 250 cfm ~~within [TBD] seconds after a start signal~~.	18 months on a STAGGERED TEST BASIS

B

B

9

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BASES (continued)

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident and a rod ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.25% of containment air weight per day (Ref. 2). This leakage rate is defined in 10 CFR 50, Appendix J (Ref. 1), as L_a = 0.25% of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a following a DBA. A P_a value of 15.0 psig is utilized which bounds the calculated peak containment internal pressure following a DBA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

B

The containment air locks satisfy Criterion 3 of the NRC Policy Statement.

LCO

Each containment air lock forms part of the containment pressure boundary. As part of containment, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into and exit from containment.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.~~3~~4, "Containment Penetrations." C

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed to repair. If the inner door is the one that is inoperable, however, then a short time exists when the containment boundary is not intact (during access through the outer door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock.

In the event the air lock leakage results in exceeding the overall containment leakage rate, Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment."

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

With one air lock door in one or more containment air locks inoperable, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which

(continued)

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and for valves in the Essential Raw Cooling Water (ERCW) System and Component Cooling System (CSS). These valves are in liquid containing systems and have been evaluated to have no impact on the DBA analysis.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The DBA analysis assumes that, within 60 seconds after the accident, isolation of the containment is complete and leakage terminated except for the design leakage rate, L_a. The containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (for loss of offsite power), and containment isolation valve stroke times.

The single failure criterion required to be imposed in the conduct of plant safety analyses was considered in the original design of the containment purge valves. Two valves in series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred. The inboard and outboard isolation valves on each line are provided with diverse power sources, motor operated and pneumatically operated spring closed, respectively. This arrangement was designed to preclude common mode failures from disabling both valves on a purge line.

The containment isolation valves satisfy Criterion 3 of the NRC Policy Statement.

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valves' safety function is related to control of containment leakage rates during a DBA.

The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The 24 inch containment lower compartment purge valves must have blocks installed to prevent full opening. Blocked purge valves also actuate on an automatic signal. The valves covered by this LCO are listed along with their associated stroke times in the FSAR (Ref. 2).

The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact. These passive isolation valves/devices are those listed in Reference 1.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.3

This SR requires verification that each containment isolation manual valve and blind flange located inside containment and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. For valves inside containment, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate since these valves and flanges are operated under administrative controls and the probability of their misalignment is low. The SR specifies that valves that are open under administrative controls are not required to meet the SR during the time they are open.

A Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3 and 4, for ALARA reasons. Therefore, the probability of misalignment of these valves, once they have been verified to be in their proper position, is small.

SR 3.6.3.4

Verifying that the isolation time of each power operated and automatic containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program or 92 days. B

SR 3.6.3.5

For containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, is required to ensure OPERABILITY.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.7

Verifying that each 24 inch containment lower compartment purge valve is blocked to restrict opening to ≤ ~~50%~~ 50° is required to ensure that the valves can close under DBA conditions within the times assumed in the analyses of References 1 and 2. If a LOCA occurs, the purge valves must close to maintain containment leakage within the values assumed in the accident analysis. At other times when purge valves are required to be capable of closing (e.g., during ~~movement of irradiated fuel assemblies~~), pressurization concerns are not present, thus the purge valves can be fully open. The 18 month Frequency is appropriate because the blocking devices are typically removed only during a refueling outage.

C

SR 3.6.3.8

This SR ensures that the combined leakage rate of all shield building bypass leakage paths is less than or equal to the specified leakage rate. This provides assurance that the assumptions in the safety analysis are met. The leakage rate of each bypass leakage path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves) unless the penetration is isolated by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. In this case, the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves. This method of quantifying maximum pathway leakage is only to be used for this SR (i.e., Appendix J maximum pathway leakage limits are to be quantified in accordance with Appendix J). ~~The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ A Note has been added to this SR requiring the

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The SR Frequency is as required by Appendix J, as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply.

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(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Containment Spray System

BASES

BACKGROUND

The Containment Spray System provides containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduce the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA). The Containment Spray System is designed to meet the requirements of 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal Systems," GDC 40, "Testing of Containment Heat Removal Systems," GDC 41, "Containment Atmosphere Cleanup," GDC 42, "Inspection of Containment Atmosphere Cleanup Systems," GDC 43, "Testing of Containment Atmosphere Cleanup Systems" and GDC 50, "Containment Design Basis," (Ref. 1), or other documents that were appropriate at the time of licensing (identified on a plant specific basis). 8

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the system design basis spray coverage. Each train includes a containment spray pump, one containment spray heat exchanger, a spray header, nozzles, valves, and piping. Each train is powered from a separate Engineered Safety Feature (ESF) bus. The refueling water storage tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWST to the containment recirculation sump(s).

The diversion of a portion of the recirculation flow from each train of the Residual Heat Removal (RHR) System to additional redundant spray headers completes the Containment Spray System heat removal capability. Each RHR train is capable of supplying spray coverage, if required, to supplement the Containment Spray System.

The Containment Spray System and RHR System provide a spray of cold or subcooled borated water into the upper and lower regions of containment and in dead ended volumes to limit

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time and is reasonable when considering that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment of manual, power operated, and automatic valves, excluding check valves, in the Containment Spray System provides assurance that the proper flow path exists for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since they were verified in the correct position prior to being secured. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned, are in the correct position.

SR 3.6.6.2

Verifying that each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 5). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on bypass flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice

2 tests

~~inspections~~ confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

(continued)

BASES (continued)

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criterion (GDC) 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal System," GDC 40, "Testing of Containment Heat Removal Systems," GDC 41, "Containment Atmosphere Cleanup," GDB 43, "Testing of Containment Atmosphere Cleanup Systems, " and GDC 50, "Containment Design Basis." 8

2. Watts Bar FSAR, Section 6.2, "Containment Systems."

3. Title 10, Code of Federal Regulations, Part 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."

4. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."

5. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," American Society of Mechanical Engineers, New York.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Hydrogen Recombiners

BASES

BACKGROUND

The function of the hydrogen recombiners is to eliminate the potential breach of containment due to a hydrogen oxygen reaction.

Per 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors" (Ref. 1), and GDC 41, "Containment Atmosphere Cleanup" (Ref. 2), hydrogen recombiners are required to reduce the hydrogen concentration in the containment following a loss of coolant accident (LOCA) or steam line break (SLB). The recombiners accomplish this by recombining hydrogen and oxygen to form water vapor. The vapor remains in containment, thus eliminating any discharge to the environment. The hydrogen recombiners are manually initiated since flammable limits would not be reached until several days after a Design Basis Accident (DBA).

Two 100% capacity independent hydrogen recombiner systems are provided. Each consists of controls located in the control room, a power supply and a recombiner. Recombination is accomplished by heating a hydrogen air mixture above 1150°F. The resulting water vapor and discharge gases are cooled prior to discharge from the recombiner. A single recombiner is capable of maintaining the hydrogen concentration in containment below the 4.0 ~~4.1~~ volume percent (v/o) flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate Engineered Safety Features bus, and is provided with a separate power panel and control panel. 3

APPLICABLE SAFETY ANALYSES

The hydrogen recombiners provide for the capability of controlling the bulk hydrogen concentration in containment to less than the lower flammable concentration of ~~4.1~~ 4.0 v/o following a DBA. This control would prevent a containment wide hydrogen burn, thus ensuring the pressure and temperature assumed in the analyses are not exceeded. The limiting DBA relative to hydrogen generation is a LOCA. 3

(continued)

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BASES

APPLICABLE SAFETY ANALYSES (continued)

Hydrogen may accumulate in containment following a LOCA as a result of:

a. A metal steam reaction between the zirconium fuel rod cladding and the reactor coolant;

b. Radiolytic decomposition of water in the Reactor Coolant System (RCS) and the containment sump;

c. Hydrogen in the RCS at the time of the LOCA (i.e., hydrogen dissolved in the reactor coolant and hydrogen gas in the pressurizer vapor space); or

d. Corrosion of metals exposed to containment spray and Emergency Core Cooling System solutions.

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident is calculated. Conservative assumptions recommended by Reference 3 are used to maximize the amount of hydrogen calculated.

4

Based on the conservative assumptions used to calculate the hydrogen concentration versus time after a LOCA, the hydrogen concentration in the primary containment would reach 3.4 ~~3.5~~ v/o about 5 ~~6~~ days after the LOCA and 4.1 ~~4.0~~ v/o about 2 days later if no recombiner was functioning (Ref. 5 ~~3~~). Initiating the hydrogen recombiners within 24 hours after a DBA ~~when the primary containment hydrogen concentration reaches 3.5 v/o~~ will maintain the hydrogen concentration in the primary containment below flammability limits.

5

The hydrogen recombiners are designed such that, with the conservatively calculated hydrogen generation rates discussed above, a single recombiner is capable of limiting the peak hydrogen concentration in containment to less than 4.0 v/o (Ref. 4). ~~The Hydrogen Purge System is similarly designed such that~~ one of two redundant trains is an adequate backup to the redundant hydrogen recombiners.

The hydrogen recombiners satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

Two hydrogen recombiners must be OPERABLE. This ensures operation of at least one hydrogen recombiner in the event of a worst case single active failure.

Operation with at least one hydrogen recombiner ensures that the post LOCA hydrogen concentration can be prevented from exceeding the flammability limit.

APPLICABILITY

In MODES 1 and 2, two hydrogen recombiners are required to control the hydrogen concentration within containment below its flammability limit of ~~4.1~~ 4.0 v/o following a LOCA, assuming a worst case single failure. 3

In MODES 3 and 4, both the hydrogen production rate and the total hydrogen produced after a LOCA would be less than that calculated for the DBA LOCA. Also, because of the limited time in these MODES, the probability of an accident requiring the hydrogen recombiners is low. Therefore, the hydrogen recombiners are not required in MODE 3 or 4.

In MODES 5 and 6, the probability and consequences of a LOCA are low, due to the pressure and temperature limitations in these MODES. Therefore, hydrogen recombiners are not required in these MODES.

ACTIONS

A.1

With one containment hydrogen recombiner inoperable, the inoperable recombiner must be restored to OPERABLE status within 30 days. In this condition, the remaining OPERABLE hydrogen recombiner is adequate to perform the hydrogen control function. However, the overall reliability is reduced because a single failure in the OPERABLE recombiner could result in reduced hydrogen control capability. The 30 day Completion Time is based on the availability of the other hydrogen recombiner, the small probability of a LOCA or SLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA or SLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.7.3

This SR requires performance of a resistance to ground test for each heater phase to ensure that there are no detectable grounds in any heater phase. This is accomplished by verifying that the resistance to ground for any heater phase is ≥ 10,000 ohms.

The 18 month Frequency for this Surveillance was developed considering the incidence of hydrogen recombiners failing the surveillance in the past is low.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Power Reactors."

2. Title 10, Code of Federal Regulations, Part 50, Appendix A, GDC 41, "Containment Atmosphere Cleanup."

3. Regulatory Guide 1.7, Revision 2, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," U.S. Nuclear Regulatory Commission.

4. Watts Bar FSAR, Section 6.2.5, "Combustible Gas Control."

5. TVA Calculation WBNSSG4-002, "WBN Hydrogen Volume Percent in Containment Following a LOCA."

BASES

BACKGROUND (continued)

would occur in each region even if one train failed to energize.

When the HMS is initiated, the ignitor elements are energized and heat up to a surface temperature ≥ 1700°F. At this temperature, they ignite the hydrogen gas that is present in the airspace in the vicinity of the ignitor. The HMS depends on the dispersed location of the ignitors so that local pockets of hydrogen at increased concentrations would burn before reaching a hydrogen concentration significantly higher than the lower flammability limit. Hydrogen ignition in the vicinity of the ignitors is assumed to occur when the local hydrogen concentration reaches

4 a minimum 5.0 ~~8.0~~ volume percent (v/o). ~~and results in 85% of the hydrogen present being consumed.~~

APPLICABLE SAFETY ANALYSES

The HMS causes hydrogen in containment to burn in a controlled manner as it accumulates following a degraded core accident (Ref. 3). Burning occurs at the lower flammability concentration, where the resulting temperatures and pressures are relatively benign. Without the system, hydrogen could build up to higher concentrations that could result in a violent reaction if ignited by a random ignition source after such a buildup.

The hydrogen ignitors are not included for mitigation of a Design Basis Accident (DBA) because an amount of hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding with water is far in excess of the hydrogen calculated for the limiting DBA loss of coolant accident (LOCA). The hydrogen concentration resulting from a DBA can be maintained less than the flammability limit using the hydrogen recombiners. The hydrogen ignitors, however, have been shown by probabilistic risk analysis to be a significant contributor to limiting the severity of accident sequences that are commonly found to dominate risk for plants with ice condenser containments. As such, the hydrogen ignitors are considered to be risk significant in accordance with the NRC Policy Statement.

LCO

Two HMS trains must be OPERABLE with power from two independent, safety related power supplies.

(continued)

BASES

LCO (continued)

For this plant, an OPERABLE HMS train consists of 33 of 34 ignitors energized on the train.

Operation with at least one HMS train ensures that the hydrogen in containment can be burned in a controlled manner. Unavailability of both HMS trains could lead to hydrogen buildup to higher concentrations, which could result in a violent reaction if ignited. The reaction could take place fast enough to lead to high temperatures and overpressurization of containment and, as a result, breach containment or cause containment leakage rates above those assumed in the safety analyses. Damage to safety related equipment located in containment could also occur.

APPLICABILITY

Requiring OPERABILITY in MODES 1 and 2 for the HMS ensures its immediate availability after safety injection and scram actuated on a LOCA initiation. In the post accident environment, the two HMS subsystems are required to control the hydrogen concentration within containment to near its

3 4.0

flammability limit of 4.1 v/o assuming a worst case single failure. This prevents overpressurization of containment and damage to safety related equipment and instruments located within containment.

both the hydrogen production rate and the total hydrogen production after a LOCA would be less than that calculated for the DBA LOCA.

6

In MODES 3 and 4, also, because of the limited time in these MODES, the probability of an accident requiring the HMS is low. Therefore, the HMS is not required in MODES 3 and 4.

In MODES 5 and 6, the probability and consequences of a LOCA are reduced due to the pressure and temperature limitations of these MODES. Therefore, the HMS is not required to be OPERABLE in MODES 5 and 6.

ACTIONS

A.1 and A.2

With one HMS train inoperable, the inoperable train must be restored to OPERABLE status within 7 days or the OPERABLE train must be verified OPERABLE frequently by performance of SR 3.6.8.1. The 7 day Completion Time is based on the low probability of the occurrence of a degraded core event that would generate hydrogen in amounts equivalent to a metal water reaction of 75% of the core cladding, the length of

(continued)

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.8.1 (continued)

hydrogen ignitor is acceptable because, although one inoperable hydrogen ignitor in a region would compromise redundancy in that region, the containment regions are interconnected so that ignition in one region would cause burning to progress to the others (i.e., there is overlap in each hydrogen ignitor's effectiveness between regions). The Frequency of 92 days is based on the Inservice Testing Program requirements for determining equipment OPERABILITY and has been shown to be acceptable through operating experience.

SR 3.6.8.2

This SR confirms that the two inoperable hydrogen ignitors allowed by SR 3.6.8.1 (i.e., one in each train) are not in the same containment region. The containment regions and hydrogen ignitor locations are provided in Reference ~~4~~ 3. ~~As such, failure of this SR results in entry into Condition B.~~ See Required Action B.1 for a discussion regarding how Conditions A and B and the associated Required Actions ensure that no more than one containment region can be without an OPERABLE hydrogen ignitor for any length of time without commencing a shutdown.

The Frequency of 92 days is consistent with SR 3.6.8.1.

C

7

SR 3.6.8.3

A more detailed functional test is performed every 18 months to verify system OPERABILITY. Each glow plug is visually examined to ensure that it is clean and that the electrical circuitry is energized. All ignitors (glow plugs), including normally inaccessible ignitors, are visually checked for a glow to verify that they are energized. Additionally, the surface temperature of each glow plug is measured to be ≥ 1700°F to demonstrate that a temperature sufficient for ignition is achieved. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency, which is based

(continued)

HOWEVER, AN ANALYSIS SHOWS THAT THE RELATIVE HUMIDITY HEATERS ARE NOT NEEDED TO MEET THE REQUIREMENTS OF REGULATORY GUIDE 1.52 SINCE THE ENTERING AIR RELATIVE HUMIDITY UNDER POSTULATED LOCA CONDITIONS IS LESS THAN 70%.

B

BASES

BACKGROUND (continued)

The prefilters remove large particles in the air, and the moisture separators remove entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal absorbers. Heaters may be included to reduce the relative humidity of the airstream on systems that operate in high humidity. Continuous operation of each train, for at least 10 hours per month, with heaters on, reduces moisture buildup on their HEPA filters and adsorbers. Cross-over flow ducts are provided between the two trains to allow the active train to draw air through the inactive train and cool the air to keep the charcoal beds on the inactive train from becoming too hot due to absorption of fission products.

The containment annulus vacuum fans maintain the annulus at -5 inches water gauge vacuum during normal operations. During accident Conditions, the containment annulus vacuum fans are isolated from the air cleanup portion of the system.

The EGTS reduces the radioactive content in the shield building atmosphere following a DBA. Loss of the EGTS could cause site boundary doses, in the event of a DBA, to exceed the values given in the licensing basis.

APPLICABLE SAFETY ANALYSES

The EGTS design basis is established by the consequences of the limiting DBA, which is a LOCA. The accident analysis (Ref. 3) assumes that only one train of the EGTS is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the remaining one train of this filtration system. The amount of fission products available for release from containment is determined for a LOCA.

~~The modeled EGTS actuation in the safety analysis assumes an initial annulus vacuum pressure of -5.0 inches water gauge upon receipt of the Phase A isolation signal. The fans automatically start within 20 seconds (excluding 10 seconds for diesel generator start) after receipt of the initiating signal. The analysis shows that after an initial step increase, the pressure will rise to a peak value of -0.13 inches water gauge in approximately 90 seconds after the~~

B

THE SAFETY ANALYSIS ASSUMES AN INITIAL ANNULUS VACUUM PRESSURE OF -5.0 INCHES WATER GAUGE PRIOR TO THE LOCA. THE ANALYSIS FURTHER ASSUMES THAT UPON RECEIPT OF A SIMULATED PHASE A ISOLATION SIGNAL FROM THE RPS, THE EGTS FANS AUTOMATICALLY START AND ACHIEVE A MINIMUM [illegible] OF 3600 CFM WITHIN 18 SECONDS (20 SECONDS FROM THE INITIATING EVENT). THIS [illegible]S NOT INCLUDE 10 SECONDS FOR DIESEL GENERATOR STARTUP. THE ANALYSIS SHOWS THAT THE ANNULUS PRESSURE WILL RISE TO A VALUE ABOVE THE EGTS SETPOINT OF -1.036 INCHES OF WATER (BECOME LESS NEGATIVE) BUT WILL NOT GO POSITIVE.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

LOCA and does not go positive. The annulus pressure then returns to the post accident setpoint of -0.5 inches water gauge.

(B)

The EGTS satisfies Criterion 3 of the NRC Policy Statement.

LCO

In the event of a DBA, one EGTS train is required to provide the minimum particulate iodine removal assumed in the safety analysis. Two trains of the EGTS must be OPERABLE to ensure that at least one train will operate, assuming that the other train is disabled by a single active failure.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could lead to fission product release to containment that leaks to the shield building. The large break LOCA, on which this system's design is based, is a full power event. Less severe LOCAs and leakage still require the system to be OPERABLE throughout these MODES. The probability and severity of a LOCA decrease as core power and Reactor Coolant System pressure decrease. With the reactor shut down, the probability of release of radioactivity resulting from such an accident is low.

In MODES 5 and 6, the probability and consequences of a DBA are low due to the pressure and temperature limitations in these MODES. Under these conditions, the Filtration System is not required to be OPERABLE (although one or more trains may be operating for other reasons, such as habitability during maintenance in the shield building annulus).

ACTIONS

A.1

With one EGTS train inoperable, the inoperable train must be restored to OPERABLE status within 7 days. The components in this degraded condition are capable of providing 100% of the iodine removal needs after a DBA. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant EGTS train and the low probability of a DBA occurring during this period. The Completion Time is adequate to make most repairs.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the EGTS train cannot be restored to OPERABLE status within the required Completion Time, 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 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.9.1

Operating each EGTS train for ≥ 10 hours ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation with the heaters on (automatic heater cycling to maintain temperature) for ≥ 10 continuous hours eliminates moisture on the adsorbers and HEPA filters. Experience from filter testing at operating units indicates that the 10 hour period is adequate for moisture elimination on the adsorbers and HEPA filters. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls, the two train redundancy available, and the iodine removal capability of the Containment Spray System. 10

SR 3.6.9.2

This SR verifies that the required EGTS filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The EGTS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 4). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

(continued)

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.9.3

The automatic startup ensures that each EGTS train responds properly. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore the Frequency was concluded to be acceptable from a reliability standpoint. Furthermore, the SR interval was developed considering that the EGTS equipment OPERABILITY is demonstrated at a 31 day Frequency by SR 3.6.13.1. C

9

SR 3.6.9.4

within the specified time frame.

The proper functioning of the fans, dampers, filters, adsorbers, etc., as a system is verified by the ability of each train to produce the required system flow rate. The 18 month Frequency on a STAGGERED TEST BASIS is consistent with Regulatory Guide 1.52 (Ref. 4) guidance for functional testing. 9

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 41, "Containment Atmosphere Cleanup."
2. Watts Bar FSAR, Section 6.5, "Fission Product Removal and Control Systems."
3. Watts Bar FSAR, Section 15.0, "Accident Analysis."
4. Regulatory Guide 1.52, Rev. 2, "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Absorption Units of Light-Water Cooled Nuclear Power Plants."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.10 Air Return System (ARS)

BASES

BACKGROUND

The ARS is designed to assure the rapid return of air from the upper to the lower containment compartment after the initial blowdown following a Design Basis Accident (DBA). The return of this air to the lower compartment and subsequent recirculation back up through the ice condenser assists in cooling the containment atmosphere and limiting post accident pressure and temperature in containment to less than design values. Limiting pressure and temperature reduces the release of fission product radioactivity from containment to the environment in the event of a DBA.

The ARS provides post accident hydrogen mixing in selected areas of containment. The ARS draws air from the dome of the containment vessel, from the reactor cavity, and from the ten dead ended (pocketed) spaces in the containment where there is potential for the accumulation of hydrogen. The minimum design flow from each potential hydrogen pocket is sufficient to limit the local concentration of hydrogen.

The ARS consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a 100% capacity air return fan, associated damper, and hydrogen collection headers. Each train is powered from a separate Engineered Safety Features (ESF) bus.

Isolation phase B

B

The ARS fans are automatically started by the containment ~~pressure High-High~~ signal 8 to 10 minutes after the containment pressure reaches the pressure setpoint. The time delay ensures that no energy released during the initial phase of a DBA will bypass the ice bed through the ARS fans into the upper containment compartment.

After starting, the fans displace air from the upper compartment to the lower compartment, thereby returning the air that was displaced by the high energy line break blowdown from the lower compartment and equalizing pressures throughout containment. After discharge into the lower compartment, air flows with steam produced by residual heat

(continued)

BASES

BACKGROUND (continued)

through the ice condenser doors into the ice condenser compartment where the steam portion of the flow is condensed. The air flow returns to the upper compartment through the top deck doors in the upper portion of the ice condenser compartment. The ARS fans operate continuously after actuation, circulating air through the containment volume and purging all potential hydrogen pockets in containment. When the containment pressure falls below a predetermined value, the ARS fans are ~~automatically~~ manually de-energized. Thereafter, the fans are ~~automatically~~ manually cycled on and off if necessary to control any additional containment pressure transients.

B

The ARS also functions, after all the ice has melted, to circulate any steam still entering the lower compartment to the upper compartment where the Containment Spray System can cool it.

The ARS is an ESF system. It is designed to ensure that the heat removal capability required during the post accident period can be attained. The operation of the ARS, in conjunction with the ice bed, the Containment Spray System, and the Residual Heat Removal (RHR) System spray, provides the required heat removal capability to limit post accident conditions to less than the containment design values.

APPLICABLE SAFETY ANALYSES

The limiting DBAs considered relative to containment temperature and pressure are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. DBAs are assumed not to occur simultaneously or consecutively. The postulated DBAs are analyzed, in regard to ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure and results in one train each of the Containment Spray System, RHR System, and ARS being inoperable (Ref. 1). The DBA analyses show that the maximum peak containment pressure results from the LOCA analysis and is calculated to be less than the containment design pressure.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of

(continued)

APPLICABLE SAFETY ANALYSES (continued)

the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures, in accordance with 10 CFR 50, Appendix K (Ref. 2).

The maximum peak containment atmosphere temperature of 326°F results from the SLB analysis and was calculated to exceed the containment design temperature of 250°F for a short time. This analysis is discussed in the Bases for LCO 3.6.5. "Containment Air Temperature." Thermal analyses show that the time interval during which the containment atmosphere temperature exceeds the containment design temperature is short enough that equipment surface temperatures remain below the design temperature. Therefore, it is concluded that the calculated transient containment atmosphere temperatures are acceptable for the SLB.

The modeled ARS actuation from the containment analysis is based upon a response time associated with exceeding the containment pressure High-High signal setpoint to achieving full ARS air flow. A delayed response time initiation provides conservative analyses of peak calculated containment temperature and pressure responses. The ARS total response time of 540 ± 600 seconds consists of the built in signal delay. c

The ARS satisfies Criterion 3 of the NRC Policy Statement.

LCO

In the event of a DBA, one train of the ARS is required to provide the minimum air recirculation for heat removal and hydrogen mixing assumed in the safety analyses. To ensure this requirement is met, two trains of the ARS must be OPERABLE. This will ensure that at least one train will operate, assuming the worst case single failure occurs, which is in the ESF power supply.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the operation of the ARS. Therefore, the LCO is applicable in MODES 1, 2, 3, and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the ARS is not required to be OPERABLE in these MODES.

ACTIONS

A.1

If one of the required trains of the ARS is inoperable, it must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the ~~heat removal~~ Flow capability after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal and hydrogen skimming capability of the OPERABLE ARS train and the low probability of a DBA occurring in this period. 11

B.1 and B.2

If the ARS train cannot be restored to OPERABLE status within the required Completion Time, 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 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.10.1

Verifying that each ARS fan starts on an actual or simulated actuation signal, after a delay of ≥ 8.0 minutes and ≤ 10.0 minutes, and operates for ≥ 15 minutes is sufficient to ensure that all fans are OPERABLE and that all associated controls and time delays are functioning properly. It also ensures that blockage, fan and/or motor failure, or

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.10.1 (continued)

excessive vibration can be detected for corrective action. The 92 day Frequency was developed considering the known reliability of fan motors and controls and the two train redundancy available.

Verifying ARS fan motor current with the return air backdraft dampers closed confirms one operating condition of the fan. This test is indicative of overall fan motor performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of 92 days conforms with the testing requirements for similar ESF equipment and considers the known reliability of fan motors and controls and the two train redundancy available.

SR 3.6.10.3

Verifying the OPERABILITY of the return air damper provides assurance that the proper flow path will exist when the fan is started. By applying the correct counterweight, the damper operation can be confirmed. The Frequency of 92 days was developed considering the importance of the dampers, their location, physical environment, and probability of failure. Operating experience has also shown this Frequency to be acceptable.

REFERENCES

1. Watts Bar FSAR, Section 6.~~2~~ 8, "~~Containment System.~~ "Air Return Fans."" (B)
2. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.13.5 (continued)

mechanisms used to secure the seal, and the plant conditions needed to perform the SR. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. Watts Bar FSAR, Section 6.2, "Containment ~~Spray~~ Systems." C

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.15.1

Verifying that shield building annulus negative pressure is within limit ensures that operation remains within the limit assumed in the containment analysis. The 12 hour Frequency of this SR was developed considering operating experience related to shield building annulus pressure variations and pressure instrument drift during the applicable MODES.

SR 3.6.15.2

Maintaining shield building OPERABILITY requires maintaining each door in the access opening closed, except when the access opening is being used for normal transient entry and exit. The 31 day Frequency of this SR is based on engineering judgment and is considered adequate in view of the other indications of door status that are available to the operator.

SR 3.6.15.3

This SR would give advance indication of gross deterioration of the concrete structural integrity of the shield building. The Frequency of this SR is the same as that of SR 3.6.1.1. The verification is done during shutdown.

SR 3.6.15.4

B 9 The ability of a EGTS train with final flow ≥ 3600 and ≤ 4400 cfm to produce the required negative pressure ~~0.5~~ -1.036 inch water gauge with respect to ~~El. 772 mechanical equipment room~~ the atmosphere during the test operation ~~within [TBD] seconds~~ provides assurance that the building is adequately sealed. The negative pressure prevents leakage from the building, since outside air will be drawn in by the low pressure at a maximum rate ≤ 250 cfm. 9 The negative pressure must be established within the time limit to ensure that no significant quantity of radioactive material leaks from the shield building prior to developing the negative pressure. 12 The 18 month Frequency on a STAGGERED TEST BASIS is consistent with Regulatory Guide 1.52 (Ref. 1) guidance for functional testing.

REFERENCES

~~None.~~

1. Regulatory Guide 1.52, Rev. 2, "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Power Plants."

JUSTIFICATION FOR CHANGES TO SECTION 3.7

GENERIC JUSTIFICATIONS

A. This change reflects a change to NUREG-1431 submitted to the NRC by the Industry Owners groups via the generic STS change process.

B. Removal of brackets and/or addition of plant specific parameter or information.

C. Correction of typographical error in the Proof and Review TS.

SPECIFIC JUSTIFICATIONS

1. The ASME code now references the OM standard rather than a specific subarticle, therefore, the reference to the subarticle has been deleted.

2. The MSIV design at Watts Bar permits part stroke testing at power, therefore, an exception to the code testing frequency is not applicable.

3. The valves are <u>not</u> exempt from the ASME code requirements, rather the code provides an exception to the testing frequency during power operations.

4. Watts Bar utilizes a motor driven main feedwater pump for startup and shutdown operations rather than the auxiliary feedwater system.

5. The setpoint for Watts Bar is $\pm$ 1%, therefore, this paragraph does not apply and it should be deleted.

6. The statement is technically incorrect. Depressurization of the ruptured steam generator is controlled by the atmospheric dump valves and is not necessarily affected by the MSIVs since the turbine stop valves are closed and the steam dump valves are regulated.

7. The discussion is not relevant to the Watts Bar design and should be revised consistent with Condition E. The two valves referred to in this condition is for two valves in series, not in parallel.

8. The ADV automatic operation is not credited by any safety analysis.

9. The suction lines from the CST are not separate. The discharge lines of the Turbine AFW pump tie into the motor driven pump lines.

10. The CST for Watts Bar is non-safety related and there are no non-limiting events.

11. The CST is not credited in any safety analysis and does not meet any Policy Statement Criterion; therefore, the CST should be relocated from the TS.

JUSTIFICATION FOR CHANGES TO SECTION 3.7
(continued)

12. The issue of importance to the TS is the pressurization value at a maximum allowed inlet airflow. If less than the maximum airflow is needed to maintain the pressure, the air exchange rate will be less than that specified. At any rate, the number of air exchanges is not relevant to the pressurization and should be removed.

13. The word "assumed" is unclear. The appropriate value which can be verified is the "design" heat load.

14. The analysis has changed since the August 27, 1992 TS submittal such that the vacuum relief flow is no longer a concern in the negative pressure test.

15. This statement cannot be supported since the downstream HEPA is not credited in any analysis.

16. MODES 5 and 6 have been added back to these LCOs consistent with the NUREG for a waste gas decay tank rupture, and CORE ALTERATIONS has been deleted since it is bounded by MODE 6.

17. Section 9.1.2 of the FSAR states that the spent fuel pool racks are designed such that Keff $\leq$ 0.95 when flooded with nonborated water, the fuel is enriched to 3.5 weight percent, and the geometry is the worst possible considering mechanical tolerances and abnormal conditions. This design was reviewed and accepted in the SER section 9.1.2. The Watts Bar design does not rely on boron to meet the double contingency requirements for criticality anaylses that plants with high enrichents and maximum density rack designs require. Therefore, this specification is inappropriate for the Watts Bar design as reviewed and approved without borated water.

3.7 PLANT SYSTEMS

3.7.3 Main Feedwater Isolation Valves (MFIVs) and Main Feedwater Regulation Valves (MFRVs) and Associated Bypass Valves

LCO 3.7.3 Four MFIVs, four MFRVs, and associated bypass valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, [and 3] except when MFIV, MFRV, or associated bypass valve[] is closed and de-activated or isolated by a closed manual [valve].

B

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more MFIVs inoperable.	A.1 Close or isolate MFIV.	72 hours
	<u>AND</u>	
	A.2 Verify MFIV is closed or isolated.	Once per 7 days
B. One or more MFRVs inoperable.	B.1 Close or isolate MFRV.	72 hours
	<u>AND</u>	
	B.2 Verify MFRV is closed or isolated.	Once per 7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more MFIV or MFRV bypass valves inoperable.	C.1	Restore bypass valve to OPERABLE status.	72 hours
D. One MFIV and MFRV in the same flow path inoperable.	D.1	Isolate affected flow path.	8 hours
E. One MFIV bypass valve and MFRV bypass valve in the same flow path inoperable.	E.1	Restore one MFIV bypass valve or MFRV bypass valve to OPERABLE status.	8 hours
F. Required Action and associated Completion Time not met.	F.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	F.2	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify the closure time of each MFIV, MFRV, and associated bypass valve is ≤ 6.5 seconds on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program or 18 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each AFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.5.2	-------------------NOTE------------------ Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 1092 psig in the steam generator. --- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	31 days on a STAGGERED TEST BASIS
SR 3.7.5.3	Verify each AFW automatic valve actuates to the correct position on an actual or simulated actuation signal. ~~when in MODE 1, 2, or 3.~~	18 months 4
SR 3.7.5.4	-------------------NOTE------------------ Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 1092 psig in the steam generator. --- Verify each AFW pump starts automatically on an actual or simulated actuation signal. when in MODE 1, 2, or 3.	18 months 4

(continued)

3.7 PLANT SYSTEMS

3.7.7 Component Cooling System (CCS)

LCO 3.7.7 Two CCS trains shall be OPERABLE.

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

ACTIONS

C

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CC~~W~~S train inoperable.	A.1 --------NOTE--------- Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops — MODE 4," for residual heat removal loops made inoperable by CCS. --------------------- Restore CCS train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

3.7 PLANT SYSTEMS

3.7.10 Control Room Emergency Ventilation System (CREVS)

LCO 3.7.10 Two CREVS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, ~~and~~ 4, 5, and 6, (16)
During movement of irradiated fuel assemblies.
~~During CORE ALTERATIONS.~~

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREVS train inoperable.	A.1 Restore CREVS train to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies ~~or during CORE ALTERATIONS.~~ (16)	C.1 Place OPERABLE CREVS train in emergency mode. OR C.2.1 Suspend CORE ALTERATIONS. AND C.2.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two CREVS trains inoperable in MODE 1, 2, 3, or 4.	D.1 Enter LCO 3.0.3.	Immediately
E. Two CREVS trains ~~inoperable~~ during movement of irradiated fuel assemblies. ~~or during CORE ALTERATIONS.~~ in MODE 5 or 6, or	E.1 Suspend CORE ALTERATIONS. AND E.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately

16

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each CREVS train for ≥ 15 minutes.	31 days
SR 3.7.10.2	Perform required CREVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP
SR 3.7.10.3	Verify each CREVS train actuates on an actual or simulated actuation signal.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.10.4 and adjacent areas B	Verify one CREVS train can maintain a positive pressure of ≥ 0.125 inches water gauge, relative to the outside atmosphere during the pressurization mode of operation at a makeup flow rate of ≤ 325 cfm and a recirculation flow rate ≥ 3308 and ≤ 4042 cfm.	18 months on a STAGGERED TEST BASIS

3.7 PLANT SYSTEMS

3.7.11 Control Room Emergency Air Temperature Control System (CREATCS)

LCO 3.7.11 Two CREATCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, ~~and~~ 4, 5, and 6,
During movement of irradiated fuel assemblies.
~~During CORE ALTERATIONS.~~ 16

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREATCS train inoperable.	A.1 Restore CREATCS train to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies. ~~or during CORE ALTERATIONS.~~ 16	C.1 Place OPERABLE CREATCS train in operation. OR C.2.1 Suspend CORE ALTERATIONS. AND C.2.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately Immediately

(continued)

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two CREATCS trains inoperable in MODE 1, 2, 3, or 4.	D.1 Enter LCO 3.0.3.	Immediately
E. Two CREATCS trains inoperable, during in MODE 5 or 6, or movement of irradiated fuel assemblies. ~~or during CORE ALTERATIONS~~. (16)	E.1 Suspend CORE ALTERATIONS. AND E.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.11.1 Verify each CREATCS train has the capability to remove the ~~assumed~~ design heat load.	18 months (13)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two ABGTS trains inoperable during movement of irradiated fuel assemblies in the fuel handling area.	D.1 Suspend movement of irradiated fuel assemblies in the fuel handling area.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Operate each ABGTS train for ≥ 10 continuous hours with the heaters operating.	31 days
SR 3.7.12.2	Perform required ABGTS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.12.3	Verify each ABGTS train actuates on an actual or simulated actuation signal.	18 months
SR 3.7.12.4	Verify one ABGTS train can maintain a pressure between -0.25 and -0.5 inches water gauge with respect to atmospheric pressure during the post accident mode of operation at a flow rate ≥ 8100 and ≤ 9900 cfm ~~while maintaining a vacuum relief flow ≥ 2200 cfm~~.	18 months on a STAGGERED TEST BASIS

14

17

3.7 PLANT SYSTEMS

3.7.14 Fuel Storage Pool Boron Concentration

LCO 3.7.14 The fuel storage pool boron concentration shall be ≥ [2300] ppm.

APPLICABILITY: When fuel assemblies are stored in the fuel storage pool and a fuel storage pool verification has not been performed since the last movement of fuel assemblies in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool boron concentration not within limit.	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------	
	A.1 Suspend movement of fuel assemblies in the fuel storage pool.	Immediately
	AND	
	A.2.1 Initiate action to restore fuel storage pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.14.1 Verify the fuel storage pool boron concentration is within limit.	7 days

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BASES

ACTIONS (continued)

B.1 and B.2

If the MSSVs cannot be restored to OPERABLE status within the associated Completion Time, or if one or more steam generators have less than two MSSVs OPERABLE, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code, Section XI (Ref. 4), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 5). According to Reference 5, the following tests are required:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

d. Compliance with owner's seat tightness criteria; and

e. Verification of the balancing device integrity on balanced valves.

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. ~~Table 3.7.1-2 allows a ± 1% setpoint tolerance for~~ OPERABILITY; however, the valves are reset to ± 1% during the Surveillance as a check for drift. e

5

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. Watts Bar FSAR, Section 10.3, "Main Steam Supply System."

2. American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, Section III, Article NC-7000, "Overpressure Protection," Class 2 Components.

3. Watts Bar FSAR, Section 15.2, "Condition II - Faults of Moderate Frequency," and Section 15.4, "Condition IV - Limiting Faults."

4. American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, Section XI, Article IWV-3500, "Inservice Test: Category C Valves."

5. ANSI/ASME OM-1-1987, "Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices."

1

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a high energy line break (HELB). MSIV closure terminates flow from the unaffected (intact) steam generators.

One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are downstream from the main steam safety valves (MSSVs) and auxiliary feedwater (AFW) pump turbine steam supply, to prevent MSSV and AFW isolation from the steam generators by MSIV closure. Closing the MSIVs isolates each steam line from the others, and isolates the turbine, Steam Dump System, and other auxiliary steam supplies from the steam generators.

The MSIVs close on a main steam isolation signal generated by either low steam line pressure, high negative steam pressure rate (below P-11), or high-high containment pressure. The MSIVs fail closed on loss of control or actuation power.

Each MSIV has an MSIV bypass valve. Although these bypass valves are normally closed, they receive the same emergency closure signal as do their associated MSIVs. The MSIVs may also be actuated manually.

A description of the MSIVs is found in the FSAR, Section 10.3 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, discussed in the FSAR, Section 6.2 (Ref. 2). It is also affected by the accident analysis of the SLB events presented in the FSAR, Section 15.4.2.1 (Ref. 3). The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand).

The limiting case for the Containment analysis is the SLB inside Containment, with a loss of offsite power following turbine trip, and failure of the MSIV on the affected steam

C

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. A break outside of containment and upstream from the MSIVs is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs isolates the break and limits the blowdown to a single steam generator.

c. A break downstream of the MSIVs will be isolated by the closure of the MSIVs.

d. Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator from the intact steam generators~~. In addition~~ to minimizing radiological releases~~, this enables the operator to maintain~~ the pressure of the steam generator with the ruptured tube below the MSSV setpoints, a necessary step toward isolating the flow through the rupture.

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e. The MSIVs are also utilized during other events such as a feedwater line break. This event is less limiting so far as MSIV OPERABILITY is concerned.

The MSIVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

This LCO requires that four MSIVs in the steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation actuation signal.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 (Ref. 4) limits or the NRC staff approved licensing basis.

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when closed and de-activated, when there is significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing the safety function.

(continued)

BASES

C

significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing the safety function.

APPLICABILITY (continued)

In MODE 4, normally most of the MSIVs are closed, and the steam generator energy is low.

In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A.1

With one MSIV inoperable in MODE 1, action must be taken to restore OPERABLE status within 8 hours. Some repairs to the MSIV can be made with the unit hot. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the MSIVs.

The 8 hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from other containment isolation valves in that the closed system provides an additional means for containment isolation.

B.1

If the MSIV cannot be restored to OPERABLE status within 8 hours, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in MODE 2 within 6 hours and Condition C would be entered. The Completion Times are reasonable, based on operating experience, to reach MODE 2 and to close the MSIVs in an orderly manner and without challenging plant systems.

C

C.1 and C.2

Condition C is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODES 2 and 3, the inoperable MSIVs may either be restored to

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

The 8 hour Completion Time is consistent with that allowed in Condition A.

For inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, the inoperable MSIVs must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of MSIV status indications available in the control room, and other administrative controls, to ensure that these valves are in the closed position.

D.1 and D.2

If the MSIVs cannot be restored to OPERABLE status or are not closed within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from MODE 2 conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that MSIV closure time is ≤ 5.0 seconds on an actual or simulated actuation signal. The MSIV closure time is assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The MSIVs should not be tested at power, since even a part stroke exercise increases the risk of a valve closure when the unit is generating power. As the MSIVs are not tested at power, they are exempt from the ASME Code,

2

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1 (continued)

2

Section XI (Ref. 5), requirements during operation in MODE 1 or 2.

The Frequency is in accordance with the Inservice Testing Program or 18 months. The 18 month Frequency for valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

This test is conducted in MODE 3 with the unit at operating temperature and pressure, as discussed in Reference 5 exercising requirements. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

REFERENCES

1. Watts Bar FSAR, Section 10.3, "Main Steam Supply System."
2. Watts Bar FSAR, Section 6.2, "Containment Systems."
3. Watts Bar FSAR, Section 15.4.2.1, "Major Rupture of a Main Steam Line."
4. 10 CFR 100.11.
5. American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, Section XI, Inservice Inspection, Article IWV-3400, "Inservice Tests - Category A and B Valves."

1

BASES

ACTIONS (continued)

C.1

With one MFIV or MFRV associated bypass valve in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status within 72 hours. The inoperable valve should not be closed and isolated for long periods of time since the 6-inch bypass line provides a small tempering flow to the upper SG nozzle. This limits the temperature difference between the SG and condensate storage tank fluid which would be supplied by the AFW system. The 6-inch line may be isolated for short periods of time to support calorimetric flow measurements.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

D.1

7

With an MFIV and MFRV in the same flow path inoperable, there may be no redundant system to operate automatically and perform the required safety function. ~~Although the containment can be isolated with the failure~~ of two valves in parallel in the same flow path, the double failure can be an indication of a common mode failure in the valves of this flow path, and as such, is treated the same as a loss of the isolation capability of this flow path. Under these conditions, ~~affected valves in each~~ at least one valve in the flow path must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. This action returns the system to the condition where at least one valve in each flow path is performing the required safety function. The 8 hour Completion Time is reasonable, based on operating experience, to complete the actions required to close the MFIV or MFRV, or otherwise isolate the affected flow path.

E.1

With two bypass valves in the same flow path inoperable, there may be no redundant system to operate automatically and perform the required safety function. Under these

(continued)

BASES

ACTIONS

E.1 (continued)

conditions, at least one valve in the flow path must be restored to OPERABLE status within 8 hours. The Completion Time of 8 hours is consistent with Condition D.

F.1 and F.2

If the MFIV(s) and MFRV(s) and the associated bypass valve(s) cannot be restored to OPERABLE status, or the MFIV(s) or MFRV(s) closed, or isolated within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

3

This SR verifies that the closure time of each MFIV, MFRV, and associated bypass valves is ≤ 6.5 seconds on an actual or simulated actuation signal. The MFIV and MFRV closure times are assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. These valves should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. ~~As these valves are not tested at power, they are exempt from~~ This is consistent with the ASME Code, Section XI (Ref. 2), requirements during operation in MODES 1 and 2.

The Frequency for this SR is in accordance with the Inservice Testing Program or 18 months. The 18 month Frequency for valve closure is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

(continued)

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B 3.7 PLANT SYSTEMS

B 3.7.4 Atmospheric Dump Valves (ADVs)

BASES

BACKGROUND

The ADVs provide a method for cooling the unit to residual heat removal (RHR) entry conditions should the preferred heat sink via the Steam Dump System to the condenser not be available, as discussed in the FSAR, Section 10.3 (Ref. 1). This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the condensate storage tank (CST). The ADVs may also be required to meet the design cooldown rate during a normal cooldown when steam pressure drops too low for maintenance of a vacuum in the condenser to permit use of the Steam Dump System.

One ADV line for each of the four steam generators is provided. Each ADV line consists of one ADV and an associated block valve.

The ADVs are provided with upstream block valves to permit their being tested at power, and to provide an alternate means of isolation. The ADVs are equipped with pneumatic controllers to permit control of the cooldown rate.

The ADVs are provided with a pressurized air supply from the auxiliary air compressors that, on a loss of pressure in the normal instrument air supply, automatically supplies backup air to operate the ADVs.

and B

A description of the ADVs is found in Reference 1. The ADVs are OPERABLE with only a DC power source or control air available. In addition, handwheels are provided for local manual operation.

APPLICABLE SAFETY ANALYSES

The design basis of the ADVs is established by the capability to cool the unit to RHR entry conditions. The maximum design rate of 50°F per hour is applicable for two steam generators, each with one ADV. This rate is adequate to cool the unit to RHR entry conditions utilizing the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

cooling water supply available in the CST.

In the accident analysis presented in Reference 1, the ADVs are assumed to be used by the operator to cool down the unit to RHR entry conditions for accidents accompanied by a loss of offsite power. Prior to operator actions to cool down the unit, the ~~ADVs and~~ main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the steam generator pressure below the design value. For the recovery from a steam generator tube rupture (SGTR) event, the operator is also required to perform a limited cooldown to establish adequate subcooling as a necessary step to terminate the primary to secondary break flow into the ruptured steam generator. The time required to terminate the primary to secondary break flow for an SGTR is more critical than the time required to cool down to RHR conditions for this event and also for other accidents. Thus, the SGTR is the limiting event for the ADVs. Three ADVs are required to be OPERABLE to satisfy the SGTR accident analysis requirements. This considers any single failure assumptions regarding the failure of one ADV to open on demand.

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The ADVs are equipped with block valves in the event an ADV spuriously fails to open or fails to close during use.

The ADVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

Three ADV lines are required to be OPERABLE. One ADV line is required from each of three steam generators to ensure that at least one ADV line is available to conduct a unit cooldown following an SGTR, in which one steam generator becomes unavailable, accompanied by a single, active failure of a second ADV line on an unaffected steam generator. The block valves must be OPERABLE to isolate a failed open ADV line.

Failure to meet the LCO can result in the inability to cool the unit to RHR entry conditions following an event in which

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

MODE 3 within 6 hours, and in MODE 4, without reliance upon steam generator for heat removal, within 18 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

To perform a controlled cooldown of the RCS, the ADVs must be able to be opened either remotely or locally. ~~and throttled through their full range.~~ This SR ensures that the ADVs are tested through a full control cycle at least once per fuel cycle. Performance of inservice testing or use of an ADV during a unit cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. The Frequency is acceptable from a reliability standpoint. 8

SR 3.7.4.2

The function of the block valve is to isolate a failed open ADV. Cycling the block valve both closed and open demonstrates its capability to perform this function. Performance of inservice testing or use of the block valve during unit cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. The Frequency is acceptable from a reliability standpoint.

REFERENCES

1. Watts Bar FSAR, Section 10.3, "Main Steam Supply System."

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

9

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply. The AFW pumps take suction ~~through separate and independent suction lines~~ from the condensate storage tank (CST) (LCO 3.7.6) and pump to the steam generator secondary side via separate ~~and independent~~ connections to the main feedwater (MFW) bypass line piping. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1) or atmospheric dump valves (LCO 3.7.4). If the main condenser is available, steam may be released via the steam dump valves and recirculated to the CST.

The AFW System consists of two motor driven AFW pumps and one steam turbine driven pump configured into three trains. Each motor driven pump provides 410 gpm of AFW flow capacity, and the turbine driven pump provides 720 gpm to the steam generators, as assumed in the accident analysis. The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system. Each motor driven AFW pump is powered from an independent Class IE power supply and feeds two steam generators. The steam turbine driven AFW pump receives steam from one of two main steam lines upstream of the main steam isolation valves. Each of the steam feed lines will supply 100% of the requirements of the turbine driven AFW pump.

The AFW System is capable of supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions, however, the main feedwater system will normally perform these functions.

The turbine driven AFW pump supplies a common header capable of feeding all steam generators. One pump at full flow is sufficient to remove decay heat and cool the unit to residual heat removal (RHR) entry conditions. Thus, the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

B

The AFW System design is such that it can perform its function following an FWLB between the MFW isolation valves and containment, combined with a loss of offsite power following turbine trip, and a single active failure of the steam turbine driven AFW pump. One motor driven AFW pump would deliver to the broken MFW header until the problem was detected, and flow terminated by the operator within 10 minutes. Sufficient flow would be delivered to the intact steam generator S by the redundant AFW pump.

The ESFAS automatically actuates the AFW turbine driven pump and associated power operated valves and controls when required to ensure an adequate feedwater supply to the steam generators during loss of power.

Each motor-driven auxiliary feedwater pump (one Train A and one Train B) supplies flow paths to two steam generators. Each flow path contains automatic air-operated level control valves (LCVs). The LCVs have the same train designation as the associated pump and are provided trained air. The turbine-driven auxiliary feedwater pump supplies flow paths to all four steam generators. Each of these flow paths contains an automatic air-operated LCV, two of which are designated as Train A, receive A-train air and provide flow to the same steam generators that are supplied by the B-train motor-driven auxiliary feedwater pump. The remaining two LCVs are designated as Train B, receive B-train air, and provide flow to the same steam generators that are supplied by the A-train motor-driven pump. This design provides the required redundancy to ensure that at least two steam generators receive the necessary flow assuming any single failure. It can be seen from the description provided above that the loss of a single train of air (A or B) will not prevent the auxiliary feedwater system from performing its intended safety function and is no more severe than the loss of a single auxiliary feedwater pump. Therefore, the loss of a single train of auxiliary air only affects the capability of a single motor-driven auxiliary feedwater pump because the turbine-driven pump is still capable of providing flow to the two steam generators that are separated from the other motor-driven pump.

The AFW System satisfies the requirements of Criterion 3 of the NRC Policy Statement.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.2

This SR verifies that the AFW pumps develop sufficient discharge pressure to deliver the required flow at the lowest set pressure of the MSSVs plus 1% setpoint tolerance and 3% accumulation. Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. Periodically comparing the reference differential pressure developed at this reduced flow detects trends that might be indicative of incipient failure. Performance of inservice testing discussed in the ASME Code, Section XI (Ref. 2) (only required at 3 month intervals) satisfies this requirement. The 31 day Frequency on a STAGGERED TEST BASIS results in testing each pump once every 3 months, as required by Reference 2.

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.

SR 3.7.5.3

4

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal in MODES 1, 2, and 3. In MODE 4, the required AFW train is already aligned and operating; therefore, this SR is not required. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is acceptable based on operating experience and the design reliability of the equipment.

SR 3.7.5.4

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an ESFAS by demonstrating that each AFW pump starts ~~automatically~~ on an actual or simulated actuation signal in MODES 1, 2, and 3. 4

(continued)

BASES

SURVEILLANCE REQUIREMENTS

4

SR 3.7.5.4 (continued)

In MODE 4, the required pump is already operating and the autostart function is not required. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

This SR is modified by a Note indicating that the SR be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.

SR 3.7.5.5

This SR verifies that the AFW is properly aligned by verifying the flow paths from the CST to each steam generator prior to entering MODE 2 after more than 30 days in MODE 5 or 6. OPERABILITY of AFW flow paths must be verified before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgment and other administrative controls that ensure that flow paths remain OPERABLE. To further ensure AFW System alignment, flow path OPERABILITY is verified following extended outages to determine no misalignment of valves has occurred. This SR ensures that the flow path from the CST to the steam generators is properly aligned.

REFERENCES

1

1. Watts Bar FSAR, Section 10.4.9, "Auxiliary Feedwater System."

2. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400, "Inservice Tests - Category A and B Valves."

BASES

APPLICABLE SAFETY ANALYSES (continued)

power. Single failures that also affect this event include the following:

a. Failure of the diesel generator powering the motor driven AFW pump to the unaffected steam generator (requiring additional steam to drive the remaining AFW pump turbine); and

b. Failure of the steam driven AFW pump (requiring a longer time for cooldown using only one motor driven AFW pump).

These are not usually the limiting failures in terms of consequences for these events.

9 A nonlimiting event considered in CST inventory determinations is a break in either the main feedwater bypass line or AFW line near where the two join. This break has the potential for dumping condensate until terminated by operator action. This loss of condensate inventory is partially compensated for by the retention of steam generator inventory. 10

The CST satisfies Criterion 3 of the NRC Policy Statement. e 11

LCO

As the preferred water source to satisfy accident analysis assumptions, the CST must contain sufficient cooling water to remove decay heat for 2 hours following a reactor trip from 102% RTP, and then to cool down the RCS to RHR entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. In doing this, it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during cooldown, as well as account for any losses from the steam driven AFW pump turbine, or before isolating AFW to a broken line.

The CST level required is equivalent to a usable volume of ≥ 200,000 gallons, which is based on holding the unit in MODE 3 for 2 hours, followed by a cooldown to RHR entry conditions at 50°F/hour. This basis is established in Reference 4 and exceeds the volume required by the accident analysis.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This SR verifies that the C-S pump is powered from the normal power source when it is aligned for OPERABLE status. Verification of the correct power alignment ensures that the two CCS trains remain independent. The 7-day Frequency is based on engineering judgment, is consistent with procedural controls governing breaker operation, and ensures correct breaker position.

SR 3.7.7.2

c

This SR is modified by a Note indicating that the isolation of the CCS flow to individual components may render those components inoperable but does not affect the OPERABILITY of the CCS System.

Verifying the correct alignment for manual, power operated, and automatic valves in the CCS flow path provides assurance that the proper flow paths exist for CCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.10 Control Room Emergency Ventilation System (CREVS)

BASES

BACKGROUND

The CREVS provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity.

The CREVS consists of two independent, redundant trains that recirculate and filter the control room air. Each train consists of a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system.

The CREVS is an emergency system, parts of which also operate during normal unit operations.

Actuation of the CREVS occurs automatically upon receipt of a safety injection signal in either unit or upon indication of high radiation in the outside air supply. Actuation of the system to the emergency mode of operation, closes the unfiltered outside air intake and unfiltered exhaust dampers, and aligns the system for recirculation of the control room air through the redundant trains of air handling units, with a portion of the stream of air directed through HEPA and the charcoal filters. The emergency mode also initiates pressurization and filtered ventilation of the air supply to the control room. Pressurization of the control room prevents infiltration of unfiltered air from the surrounding areas of the building.

B B

A single train will pressurize the control room to ~~about~~ a minimum 0.125 inches water gauge with respect to the outside atmosphere and adjacent areas.~~, and provide an air exchange rate in excess of [7.5]% per hour.~~ 12 The CREVS operation in maintaining the control room habitable is discussed in the FSAR, Section 6.4 (Ref. 1).

Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across the other filter train. Normally open

(continued)

BASES

LCO (continued)

The CREVS is considered OPERABLE when the individual components necessary to limit operator exposure are OPERABLE in both trains. A CREVS train is OPERABLE when the associated:

a. Fan is OPERABLE;

b. HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

APPLICABILITY

, 5, and 6,

In MODES 1, 2, 3, ~~and~~ 4 and during movement of irradiated fuel assemblies, ~~and during CORE ALTERATIONS,~~ CREVS must be OPERABLE to control operator exposure during and following a DBA. 16

16

During movement of irradiated fuel assemblies and CORE ALTERATIONS, the CREVS must be OPERABLE to cope with the release from a fuel handling accident.

¶ In MODE 5 or 6, the CREFS is required to cope with the release from the rupture of a waste gas decay tank.

ACTIONS

A.1

When one CREVS train is inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREVS train is adequate to perform the control room protection function. However, the overall reliability is reduced because a single failure in the OPERABLE CREVS train could result in loss of CREVS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

In MODE 1, 2, 3, or 4, if the inoperable CREVS train cannot be restored to OPERABLE status within the required Completion Time, the plant must be placed in a MODE that minimizes accident risk. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1, C.2.1, and C.2.2

In MODE 5 or 6, or

16

During movement of irradiated fuel assemblies ~~or during CORE ALTERATIONS~~, if the inoperable CREVS train cannot be restored to OPERABLE status within the required Completion Time, action must be taken to immediately place the OPERABLE CREVS train in the emergency mode. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

D.1

If both CREVS trains are inoperable in MODE 1, 2, 3, or 4, the CREVS may not be capable of performing the intended function and the plant is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

E.1 and E.2

16

In MODE 5 or 6, or

During movement of irradiated fuel assemblies ~~or during CORE ALTERATIONS~~ with two CREVS trains inoperable, action must be taken immediately to suspend activities that could result in

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.10.4 (continued)

(B)

pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning of the CREVS. During the emergency mode of operation, the CREVS is designed to pressurize the control room $\geq$ 0.125 inches water gauge positive pressure with respect to the outside atmosphere and adjacent areas in order to prevent unfiltered inleakage. The CREVS is designed to maintain this positive pressure with one train at a makeup flow rate ~~of~~ ≤ 325 cfm and a recirculation flow rate $\geq$ 3308 and $\leq$ 4042 cfm. The Frequency of 18 months on a STAGGERED TEST BASIS is consistent with the guidance provided in NUREG-0800 (Ref. 4). (B)

REFERENCES

1. Watts Bar FSAR, Section 6.4, "Habitability Systems."
2. Watts Bar FSAR, Section 15.5.3, "Environmental Consequences of a Postulated Loss of Coolant Accident."
3. Regulatory Guide 1.52, Rev. 2, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water Cooled Nuclear Power Plants."
4. NUREG-0800, Standard Review Plan, Section 6.4, "Control Room Habitability System," Rev. 2, July 1981.

BASES (continued)

LCO

Two independent and redundant trains of the CREATCS are required to be OPERABLE to ensure that at least one is available, assuming a single failure disabling the other train. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident.

The CREATCS is considered to be OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in both trains. These components include the chillers, AHUs, and associated temperature control instrumentation. In addition, the CREATCS must be operable to the extent that air circulation can be maintained.

APPLICABILITY (16)

In MODES 1, 2, 3, ~~and~~ 4, 5, and 6, and during movement of irradiated fuel assemblies, ~~and during CORE ALTERATIONS,~~ the CREATCS must be OPERABLE to ensure that the control room temperature will not exceed equipment operational requirements following isolation of the control room.

(16) In MODE 5 or 6, CREATCS ~~may not be~~ is required ~~for those~~ during a control room isolation following a waste gas decay tank rupture.

ACTIONS

A.1

With one CREATCS train inoperable, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CREATCS train is adequate to maintain the control room temperature within limits. However, the overall reliability is reduced because a single failure in the OPERABLE CREATCS train could result in loss of CREATCS function. The 30 day Completion Time is based on the low probability of an event requiring control room isolation, the consideration that the remaining train can provide the required protection, and that alternate safety or nonsafety related cooling means are available.

B.1 and B.2

In MODE 1, 2, 3, or 4, if the inoperable CREATCS train cannot be restored to OPERABLE status within the required Completion Time, the plant must be placed in a MODE that

(continued)

BASES (continued)

ACTIONS

B.1 and B.2 (continued)

minimizes the risk. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

16

C.1, C.2.1, and C.2.2

In MODE 5 or 6, or

During movement of irradiated fuel ~~or during CORE ALTERATIONS~~, if the inoperable CREATCS train cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CREATCS train must be placed in operation immediately. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that active failures will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

D.1

If both CREATCS trains are inoperable in MODE 1, 2, 3, or 4 the control room CREATCS may not be capable of performing its intended function. Therefore, LCO 3.0.3 must be entered immediately.

16

E.1 and E.2

In MODE 5 or 6, or

During movement of irradiated fuel assemblies or during CORE ALTERATIONS, with two CREATCS trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

design

This SR verifies that the heat removal capability of the system is sufficient to remove the ~~assumed~~ design heat load in the control room. This SR consists of a combination of testing and calculations. This is accomplished by verifying that the system has not degraded. The only measurable parameters that could degrade undetected during normal operation is the system air flow and chilled water flow rate. Verification of these two flow rates will provide assurance that the heat removal capacity of the system is still adequate. The 18 month Frequency is appropriate since significant degradation of the CREATCS is slow and is not expected over this time period. (13)

REFERENCES

1. Watts Bar FSAR, Section 9.4.1, "Control Room Area Ventilation System."

B 3.7 PLANT SYSTEMS

B 3.7.12 Auxiliary Building Gas Treatment System (ABGTS)

BASES

BACKGROUND

The ABGTS filters airborne radioactive particulates from the area of the fuel pool following a fuel handling accident and from the area of active Unit 1 ECCS components and Unit 1 penetration rooms following a loss of coolant accident (LOCA).

The ABGTS consists of two independent and redundant trains. Each train consists of a heater, a prefilter, moisture separator, a high efficiency particulate air (HEPA) filter, B two ~~an~~ activated charcoal adsorber sections for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system. A second bank of HEPA filters follows the adsorber section to collect carbon fines. ~~and provide backup in case the main HEPA filter bank fails.~~ 15 The downstream HEPA filter is not credited in the analysis. The system initiates filtered ventilation of the Auxiliary Building Secondary Containment Enclosure (ABSCE) exhaust air following receipt of a Phase A containment isolation signal or a high radiation signal from the spent fuel pool area. B

The ABGTS is a standby system, not used during normal plant operations. During emergency operations, the ABSCE dampers are realigned and ABGTS fans are started to begin filtration. Air is exhausted from the Unit 1 ECCS pump rooms, Unit 1 penetration rooms, and fuel handling area through the filter trains. The prefilters or moisture separators remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The ABGTS is discussed in the FSAR, Sections 6.5.1, 9.4.2, 15.0, and 6.2.3 (Refs. 1, 2, 3, and 4, respectively).

APPLICABLE SAFETY ANALYSES

The ABGTS design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident. The analysis of the fuel handling accident, given in Reference 3, assumes that all fuel rods in an assembly are damaged. The analysis of the LOCA

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.3

This SR verifies that each ABGTS train starts and operates on an actual or simulated actuation signal. The 18 month Frequency is consistent with Reference 8.

SR 3.7.12.4

This SR verifies the integrity of the ABSCE. The ability of the ABSCE to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the ABGTS. During the post accident mode of operation, the ABGTS is designed to maintain a slight negative pressure in the ABSCE, to prevent unfiltered LEAKAGE. The ABGTS is designed to maintain a negative pressure between -0.25 and -0.5 inches water gauge with respect to atmospheric pressure at a nominal rate ≥ 8100 and ≤ 9900 cfm. ~~while maintaining a vacuum relief rate~~ ~~≥ 2000 cfm~~. The Frequency of 18 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 9).

14

15

~~This test is conducted with the tests for filter penetration; thus,~~ an 18 month Frequency (on a STAGGERED TEST BASIS) is consistent with Reference 8.

REFERENCES

1. Watts Bar FSAR, Section 6.5.1, "Engineered Safety Feature (ESF) Filter Systems."
2. Watts Bar FSAR, Section 9.4.2, "Fuel Handling Area Ventilation System."
3. Watts Bar FSAR, Section 15.0, "Accident Analysis."
4. Watts Bar FSAR, Section 6.2.3, "Secondary Containment Functional Design."
5. Regulatory Guide 1.25, March 1972, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.14 Fuel Storage Pool Boron Concentration

BASES

BACKGROUND

16 17

In the High Density Rack (HDR) [(Refs. 1 and 2)] design, the spent fuel storage pool is divided into two separate and distinct regions which, for the purpose of criticality considerations, are considered as separate pools. [Region 1], with [336] storage positions, is designed to accommodate new fuel with a maximum enrichment of [4.65] wt% U-235, or spent fuel regardless of the discharge fuel burnup. [Region 2], with [2670] storage positions, is designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within the acceptable domain according to Figure [3.7.17-1], in the accompanying LCO. Fuel assemblies not meeting the criteria of Figure [3.7.17-1] shall be stored in accordance with paragraph 4.3.1.1 in Section 4.3, Fuel Storage.

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 3) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is associated with the movement of fuel from [Region 1 to Region 2], and accidental misloading of a fuel assembly in [Region 2]. This could potentially increase the criticality of [Region 2]. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the MDR with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with LCO 3.7.17, "Spent Fuel Assembly Storage." Prior to movement of an assembly, it is necessary to perform SR 3.7.16.1.

(continued)

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BASES (continued)

APPLICABLE SAFETY ANALYSES

(16)(17)

Most accident conditions do not result in an increase in the activity of either of the two regions. Examples of these accident conditions are the loss of cooling (reactivity increase with decreasing water density) and the dropping of a fuel assembly on the top of the rack. However, accidents can be postulated that could increase the reactivity. This increase in reactivity is unacceptable with unborated water in the storage pool. Thus, for these accident occurrences, the presence of soluble boron in the storage pool prevents criticality in both regions. The postulated accidents are basically of two types. A fuel assembly could be incorrectly transferred from [Region 1 to Region 2] (e.g., an unirradiated fuel assembly or an insufficiently depleted fuel assembly). The second type of postulated accidents is associated with a fuel assembly which is dropped adjacent to the fully loaded [Region 2] storage rack. This could have a small positive reactivity effect on [Region 2]. However, the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by either one of the two postulated accident scenarios. The accident analyses is provided in the FSAR, Section [15.7.4] (Ref. 4).

The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of the NRC Policy Statement.

LCO

The fuel storage pool boron concentration is required to be $\geq$ [2300] ppm. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in Reference 4. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the fuel storage pool.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel storage pool, until a complete spent fuel storage pool verification has been performed following the last movement of fuel assemblies in the spent fuel storage pool. This LCO does not apply following the verification, since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in

(continued)

BASES

APPLICABILITY (continued)	progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.
ACTIONS	<u>A.1 and A.2.1</u> The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply. When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This does not preclude movement of a fuel assembly to a safe position. If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.
SURVEILLANCE REQUIREMENTS	<u>SR 3.7.14.1</u> This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.
REFERENCES	1. Callaway FSAR, Appendix 9.1A, "The Maximum Density Rack (MDR) Design Concept."

16 17

(continued)

BASES

REFERENCES (continued)

16 17

2. Description and Evaluation for Proposed Changes to Facility Operating Licenses DPR-39 and DPR-48 (Zion Power Station).

3. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

4. FSAR, Section [15.7.4].

B 3.7 PLANT SYSTEMS

B 3.7.15 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives, (i.e., < 20 hours). I-131, with a half life of 8.04 days, concentrates faster than it decays, but does not reach equilibrium because of blowdown and other losses.

With the specified activity limit, the resultant 2 hour thyroid dose to a person at the exclusion area boundary (EAB) would be about 0.58 rem if the main steam safety valves (MSSVs) open for 2 hours following a trip from full power.

Operating a unit at the allowable limits could result in a 2 hour EAB exposure of a small fraction of the 10 CFR 100 (Ref. 1) limits, or the limits established as the NRC staff approved licensing basis.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the FSAR, Section 15.0 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 μCi/gm DOSE EQUIVALENT I-131. C This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed a small fraction of the unit EAB limits (Ref. 1) for whole body and thyroid dose rates.

With the loss of offsite power, the remaining steam generators are available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator atmospheric dump valves (ADVs). The Auxiliary Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through the MSSVs and ADVs during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

C

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to a small fraction of the required limit (Ref. 1).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner

(continued)

JUSTIFICATION FOR CHANGES TO SECTION 3.9

GENERIC JUSTIFICATIONS

A. This change reflects a change to NUREG-1431 submitted to the NRC by the Industry Owners groups via the generic STS change process.

C. Correction of typographical error in the Proof and Review TS.

SPECIFIC JUSTIFICATIONS

1. The SR was added as part of the generic change process, however, Bases were not provided. Bases from RCS Loops Not Filled have been added which uses the same surveillance requirement.

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Boron concentration not within limit.	A.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	A.2	Suspend positive reactivity additions.	Immediately
	AND		
	A.3	Initiate action~~s~~ to restore boron concentration to within limit~~s~~.	Immediately A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Verify boron concentration is within the limit specified in COLR.	72 hours A

3.9 REFUELING OPERATIONS

3.9.2 Unborated Water Source Isolation Valves

LCO 3.9.2 Each valve used to isolate unborated water sources shall be secured in the closed position.

APPLICABILITY: MODE 6.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each unborated water source isolation valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Required Action A.3 must be completed whenever Condition A is entered. --------------------- One or more valves not secured in closed position.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Initiate action~~s~~ to secure valve in closed position.	Immediately A
	AND	
	A.3 Perform SR 3.9.1.1.	4 hours

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by four bolts;

b. One door in each ~~airlock~~ air lock closed; and (A)

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE Containment Vent Isolation System.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS. AND A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately Immediately

3.9 REFUELING OPERATIONS

3.9.5 Residual Heat Removal (RHR) and Coolant Circulation—High Water Level

LCO 3.9.5 One RHR loop shall be OPERABLE and in operation.

---------------------------NOTE---------------------------
The required RHR loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause ~~dilution~~ reduction of the Reactor Coolant System boron concentration. A

APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RHR loop requirements not met.	A.1 Suspend operations involving a reduction in reactor coolant boron concentration.	Immediately
	<u>AND</u>	
	A.2 Suspend loading irradiated fuel assemblies in the core.	Immediately
	<u>AND</u>	
	A.3 Initiate action to satisfy RHR loop requirements.	Immediately
	<u>AND</u>	
		(continued)

3.9 REFUELING OPERATIONS

3.9.6 Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level

LCO 3.9.6 Two RHR loops shall be OPERABLE, and one RHR loop shall be in operation.

---------------------------NOTE---------------------------
Prior to initial criticality, only one RHR loop needs to be OPERABLE and in operation and the required RHR loop may be removed from operation for ≤ 1 hour per 8-hour period provided no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration.

APPLICABILITY: MODE 6 with the water level < 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Less than the required number of RHR loops OPERABLE.	A.1	Initiate action to restore required RHR loops to OPERABLE status.	Immediately
	<u>OR</u>		
	A.2	Initiate action~~s~~ to establish ≥ 23 ft of water above the top of reactor vessel flange.	Immediately A
B. No RHR loop in operation.	B.1	Suspend operations involving a reduction in reactor coolant boron concentration.	Immediately
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Initiate action to restore one RHR loop to operation. AND B.3 Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	Immediately 4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.6.1	Verify one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ 2000 gpm.	12 hours
SR 3.9.6.2	Verify correct breaker alignment and indicated power available to the required RHR pump that is not in operation.	7 days

A

3.9 REFUELING OPERATIONS

3.9.7 Refueling Cavity Water Level

LCO 3.9.7 Refueling cavity water level shall be maintained ≥ 23 ft above the top of reactor vessel flange.

-----NOTE-----
This LCO is not required to be met for the initial core loading.

APPLICABILITY: During CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	A.2	Suspend movement of irradiated fuel assemblies within containment.	Immediately
	<u>AND</u>		
	A.3	Initiate actions to restore refueling cavity water level to within limits.	Immediately

A

BASES

ACTIONS

A.3 (continued)

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures that the coolant boron concentration in the RCS, the refueling canal, and the refueling cavity is within the COLR limits. The boron concentration of the coolant in each volume is determined periodically by chemical analysis.

A minimum Frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, Section VI III, GDC 26, "Reactivity Control System Redundancy and Capability."

2. Watts Bar FSAR, Section 15, "Accident Analysis."

B 3.9 REFUELING OPERATIONS

B 3.9.3 Nuclear Instrumentation

BASES

BACKGROUND

The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed source range neutron flux monitors are part of the Neutron Monitoring System (NMS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core.

The installed primary source range neutron flux monitors are fission chambers. The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux (1E+6 cps) with a [5]% instrument accuracy. The detectors also provide continuous visual indication in the control room and an audible alarm to alert operators to a possible dilution accident. The NMS is designed in accordance with the criteria presented in Reference 1. B

APPLICABLE SAFETY ANALYSES

Two OPERABLE source range neutron flux monitors are required to provide a signal to alert the operator to unexpected changes in core reactivity such as with a boron dilution accident (Ref. 2) or an improperly loaded fuel assembly. The need for a safety analysis for an uncontrolled boron dilution accident is eliminated by isolating all unborated water sources as required by LCO 3.9.2, "Unborated Water Source Isolation Valves."

The source range neutron flux monitors satisfy Criterion 3 of the NRC Policy Statement.

LCO

This LCO requires that two source range neutron flux monitors be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1 (continued)

determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. In addition, during operation of the RHR loop with the water level in the vicinity of the reactor vessel nozzles, the RHR pump suction requirements must be met. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator for monitoring the RHR System in the control room.

A
1
INSERT

REFERENCES

1. Watts Bar FSAR, Section 5.5.7, "Residual Heat Removal System."

(continued)

INSERT

SR ~~3.4.8.2~~ 3.9.6.2

Verification that the required number of pumps are OPERABLE ensures that additional pumps can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pumps. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

BASES

REFERENCES (continued)

4. Title 10, Code of Federal Regulations, Part 20.1201(a), (a)(1), and (2)(2), "Occupational Dose Limits for Adults."

5. Malinowski, D. D., Bell, M. J., Duhn, E., and Locante, J., WCAP-~~828~~ 7828, Radiological Consequences of a Fuel Handling Accident, December 1971. C

JUSTIFICATION FOR CHANGES TO SECTION 4.0

GENERIC JUSTIFICATIONS

B. Removal of brackets and/or addition of plant specific parameter or information.

C. Correction of typographical error in the Proof and Review TS.

SPECIFIC JUSTIFICATIONS

1. Item c only provides a single information element from the description of the Watts Bar racks and is neither a criterion nor a limit. Other descriptive information of equal or greater importance which is not in the Technical Specifications is details of the neutron absorber poison, flux trap dimensions, etc. Center-to-center distance has been carried over from the first generation spent fuel storage racks for which it was the sole physical method for maintaining subcriticality. This is still true for new fuel racks and this criteria has been retained. Item b is the governing regulatory requirement for criticality control which must be addressed and satisfied by spent fuel storage design analyses and equipment. Item a establishes controls over the fuel to be stored. Together, these two criteria bound the design features and conditions for spent fuel storage. Other details of the spent fuel storage racks are appropriately described in the FSAR.

4.0 DESIGN FEATURES

4.1 Site

4.1.1 Site and Exclusion Area Boundaries

The site and exclusion area boundaries shall be as shown in Figure 4.1-1.

4.1.2 Low Population Zone (LPZ)

The LPZ shall be as shown in Figure 4.1-2 (within the 3-mile circle).

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of zirconium alloy fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 (Control Rod) Assemblies

B

The reactor core shall contain 57 control rod assemblies. The control material shall be boron carbide with silver indium cadmium tips as approved by the NRC.

(continued)

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

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

a. Fuel assemblies having a maximum U-235 enrichment of ~~3.15~~ 3.50 weight percent; and B

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 FSAR~~; and~~. C

c. A nominal 10.7 inch center to center distance between fuel assemblies placed in the high density fuel storage racks~~;~~. 2 1 Ø

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

a. Fuel assemblies having a maximum U-235 enrichment of 4.3 weight percent;

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 FSAR;

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 FSAR; and

d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 745′ - 1 1/2".

(continued)

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage (continued)

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than ~~[1312]~~ 1294 fuel assemblies.

B

(continued)

JUSTIFICATION FOR CHANGES TO SECTION 5.0

GENERIC JUSTIFICATIONS

A. This change reflects a change to NUREG-1431 submitted to the NRC by the Industry Owners groups via the generic STS change process.

B. Removal of brackets and/or addition of plant specific parameter or information.

C. Correction of typographical error in the Proof and Review TS.

SPECIFIC JUSTIFICATIONS

1. Change to be consistent with similar statement in paragraph 5.5.1.1.a.3.

2. The reference to snubbers in accordance with ASME Section XI is incorrect. 10 CFR 50.55a invokes ASME Section XI for the testing of pumps and valves, but not for the testing of snubbers. In the August 6, 1992 Federal Register, Volume 57, Number 152, page 34671 (attached), the NRC states categorically that snubber testing is not required by 10 CFR 50.55a and notes that plant technical specifications generally address snubber testing. Previous Technical Specifications which did address snubber testing have been relocated to the Watts Bar Technical Requirements Manual in accordance with the Commission's Policy Statement on Improved Technical Specifications.

   At the NRC/Industry Lead Plant Technical Specification Meeting in Chattanooga, Tennessee (April 20-22, 1993), the staff provided a draft copy of proposed changes to address the snubber issue. These changes are proposed to be added to the Watts Bar Tech Specs.

3. The Watts Bar design does not fully implement Regulatory Guide 1.52 nor ANSI-N510 as noted in the referenced FSAR table 6.5.

4. Change to reflect latest 10 CFR 20 changes.

5. The current STS schedule requirements are in conflict with 10 CFR 50.36 which requires that the time between reports shall not exceed 12 months. The STS would allow a floating 90 day period after January 1 of each year within which to provide a report. This would be more desirable than a schedule which could potentially retreat from year to year, however, it does not appear to be permitted by the CFR.

6. This represents a generic change in process by the NRC.

5.5 Reviews and Audits

5.5.1 Plant Reviews (continued)

The qualifications required to serve as a member or alternate member shall be specified with the minimum qualifications as recommended in Section 4 of ANSI N18.1-1971, except for the Site Radiological Control Manager who must meet the qualifications of Regulatory Guide 1.8, Revision 2.

2. Alternates

   All alternate members shall be appointed in writing by the PORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PORC activities at any one time.

3. Meeting Frequency

   The PORC shall meet on an as needed basis as convened by the PORC Chairman or his designated alternate.

4. Quorum

   The PORC quorum shall consist of the Chairman or his designated alternate and four members of which two may be alternates.

5. Reporting

   The PORC reports to the Plant Manager on all activities and findings. The meeting minutes shall serve as the official correspondence from PORC to the Plant Manager. PORC recommendations shall be recorded in the minutes and submitted to the Plant Manager by the PORC Chairman.

5.5.1.1 Functions

a. The PORC shall, as a minimum, incorporate functions that:

C

1. Advise the Plant Manager on all matters related to nuclear safety;

(continued)

5.5 Reviews and Audits

5.5.1.1 Functions (continued)

2. Recommend to the Plant Manager, or his designee, approval or disapproval of procedures that delegate review responsibilities of items considered under Specifications 5.5.1.2.a and 5.7.1.3;

3. Recommend to the Plant Manager, or his designee, approval or disapproval of items considered under Specifications 5.5.1.2.a.1 through 5.5.1.2.a.5 prior to their implementation, except as provided in Specification 5.7.1.3;

4. Determine whether each item considered under Specifications 5.5.1.2.a.1 through 5.5.1.2.a.4 constitutes an unreviewed safety question as defined in 10 CFR 50.59; and

5. Notify the Site Vice-President and the Nuclear Safety Review Board (NSRB) of any safety significant disagreement between the PORC and the Plant Manager within 24 hours. However, the Plant Manager shall have responsibility for resolution of such disagreements pursuant to Specification 5.1.1.

C

b. The Technical Reviewer(s) shall:

1. Advise his supervisor and/or PORC on all matters related to nuclear safety;

2. Determine the need for additional reviews by other disciplines and ensure that identified reviews are conducted for items considered under Specification 5.5.1.2.b prior to their implementation, except as provided in Specification 5.7.1.3;

3. Recommend to the designated Approval Authority, approval or disapproval of items considered under Specification 5.5.1.2.b.1 through 5.5.1.2.b.5 prior to their implementation, except as provided in Specification 5.7.1.3; and

4. Determine whether each item considered under Specifications 5.5.1.2.b.1 through 5.5.1.2.b.4 constitutes an unreviewed safety question as defined in 10 CFR 50.59. 1

5. Notify the PORC of any safety significant disagreement between reviewing organizations.

(continued)

5.0 ADMINISTRATIVE CONTROLS

5.6 Technical Specifications (TS) Bases Control

5.6.1 Changes to the Bases of the TS shall be made under appropriate administrative controls and reviewed according to Specification 5.5.1.

5.6.2 Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

a. A change in the TS incorporated in the license; or

b. A change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.

5.6.3 The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

5.6.4 Proposed changes that meet the criteria of Specification 5.6.2.a ~~and~~ or Specification 6.5.2.b shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71. A

5.0 ADMINISTRATIVE CONTROLS

5.7 Procedures, Programs, and Manuals

5.7.1 Procedures

5.7.1.1 Scope

Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in [Generic Letter 82-33];

c. Security plan implementation;

6 c ~~d~~. Site Radiological Emergency plan implementation;

d ~~e~~. Quality assurance for effluent and environmental monitoring; and

f. Fire Protection Program implementation; and

e ~~g~~. All programs specified in Specification 5.7.2.

5.7.1.2 Review and Approval

Each procedure of Specification 5.7.1.1, and changes thereto, shall be reviewed in accordance with Specification 5.5.1, approved by the Plant Manager or his designee in accordance with approved administrative procedures prior to implementation except as specified in Specification 5.7.1.3 and reviewed periodically as set forth in administrative procedures.

5.7.1.3 Temporarily Approved Changes

Temporarily approved changes to procedures of Specification 5.7.1.1 may be made provided:

a. The intent of the existing procedure is not altered;

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator license on the unit affected; and

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.3 Offsite Dose Calculation Manual (ODCM) (continued)

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

5.7.2.4 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Containment Spray, Safety Injection, Residual Heat Removal, Chemical and Volume Control, Reactor Coolant System, Sampling, and Waste Gas. The program shall include the following: C

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.7.2.5 In Plant Radiation Monitoring

This program provides controls to ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

a. Training of personnel;

b. Procedures for monitoring; and

c. Provisions for maintenance of sampling and analysis equipment.

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2 Programs and Manuals (continued)

5.7.2.6 Post Accident Sampling

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

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

5.7.2.7 Radioactive Effluent Controls Program

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

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 times the concentration values in 10 CFR 20.001-20.2401, Appendix B, Table 2, Column 2; C

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

(continued)

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5.7 Procedures, Programs, and Manuals

5.7.2.10 Inservice Inspection Program (continued)

INSERT 1

c ~~b~~. The provisions of SR 3.0.2 are applicable to the frequencies for performing inservice inspection activities;

d ~~c~~. Inspection of each reactor coolant pump flywheel per the recommendations of Regulation Position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975; and (2)

e ~~d~~. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.7.2.11 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following:

a. Provisions that inservice testing of ASME Code Class 1, 2, and 3 pumps, valves, and snubbers shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a; (2)

INSERT 2

c ~~b~~. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

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

(continued)

INSERT 1

b. Provisions for inservice inspection of all snubbers. The only snubbers excluded from this requirement are those installed on nonsafety-related systems and then only if their failure or the failure of the system on which they are installed, would not have an adverse effect on any safety-related system.

INSERT 2

b. Provisions for inservice testing of safety-related snubbers. Safety-related snubbers include those installed on safety-related components and those installed on nonsafety-related components if their failure or the failure of the component on which they are installed would have an adverse effect on any safety-related system.

5.7 Procedures, Programs, and Manuals

5.7.2.11 Inservice Testing Program (continued)

d ~~c~~. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

2

e ~~d~~. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

f ~~e~~. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.7.2.12 Steam Generator (SG) Tube Surveillance Program

Each SG shall be demonstrated OPERABLE by performance of an inservice inspection program. The program shall include the following:

a. SG tube sample size selection, sample size expansion, and inspection result classification criteria. Sample selection and testing shall be in accordance with Regulatory Guide 1.83, Revision 1, July 1975.

b. The establishment of SG tube inspection frequency dependent upon inspection result classification. Inspection frequency shall be in accordance with Regulatory Guide 1.83, Revision 1, July 1975.

c. SG tube plugging/repair limits. These limits shall be 40% of the nominal tube wall thickness consistent with Regulatory Guide 1.83, Revision 1, July 1975.

d. Specific definitions and limits for SG tube inservice inspection acceptance criteria consistent with Regulatory Guide 1.83, Revision 1, July 1975.

e. The minimum type testing to determine tube integrity.

The content and frequency of written reports shall be in accordance with Specification 5.9.2.

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Tube Surveillance Program (continued)

The provisions of SR 3.0.2 are applicable to SG Tube Surveillance Program inspection frequencies except those established by Category C-3 inspection results.

[Key elements to be discussed and provided.]

5.7.2.13 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.7.2.14 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in accordance with Regulatory Guide 1.52, Revision 2; ~~and~~ ASME N510-1989, and the exceptions noted for each ESF system in Table 6.5 of the FSAR. 3

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.14 Ventilation Filter Testing Program (VFTP) (continued)

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

ESF VENTILATION SYSTEM	ACCEPTANCE CRITERIA	FLOW RATE
Reactor Building Purge	< 1.00%	14,000 cfm ± 10%
Emergency Gas Treatment	< 0.05%	4,000 cfm ± 10%
Auxiliary Building Gas Treatment	< 0.05%	9,000 cfm ± 10%
Control Room Emergency	< 1.00%	4,000 cfm ± 10%

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass within acceptance criterion when tested in accordance with Regulatory Guide 1.52, Revision 2, ~~and~~ ASME N510-1989, and the exceptions noted for each ESF system in Table 6.5 of the FSAR, at the system flowrate specified below. (3)

ESF VENTILATION SYSTEM	ACCEPTANCE CRITERIA	FLOW RATE
Reactor Building Purge	< 1.00%	14,000 cfm ± 10%
Emergency Gas Treatment	< 0.05%	4,000 cfm ± 10%
Auxiliary Building Gas Treatment	< 0.05%	9,000 cfm ± 10%
Control Room Emergency	< 1.00%	4,000 cfm ± 10%

(continued)

5.7 Procedures, Programs, and Manuals

and the exceptions noted for each ESF system in Table 6.5 of the FSAR, (3)

5.7.2.14 Ventilation Filter Testing Program (VFTP) (continued)

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of ≤ 30°C and greater than or equal to the relative humidity specified below.

ESF VENTILATION SYSTEM	METHYL IODIDE PENETRATION	RELATIVE HUMIDITY	
Reactor Building Purge	10.0% < ~~0.7%~~ ~~1.0%~~	95%	B
Emergency Gas Treatment	< 0.175%	70%	
Auxiliary Building Gas Treatment	< 0.175%	70%	
Control Room Emergency	1.0% < ~~0.7%~~ ~~10.0%~~	70%	B

d. Demonstrate for each of the ESF systems that the pressure drop across the entire filtration unit is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989 at the system flowrate specified below.

the exceptions noted for each ESF system in Table 6.5 of the FSAR, (3)

ESF VENTILATION SYSTEM	PRESSURE DROP	FLOW RATE	
Reactor Building Purge	< 6.0 inches water	14,000 cfm ± 10%	B
Emergency Gas Treatment	< ~~[8.0]~~ 7.6 inches water	4,000 cfm ± 10%	B
Auxiliary Building Gas Treatment	< 8.0 inches water	9,000 cfm ± 10%	B
Control Room Emergency	< ~~[8.0]~~ 3.5 inches water	4,000 cfm ± 10%	B

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.14 Ventilation Filter Testing Program (VFTP) (continued)

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

ESF VENTILATION SYSTEM	AMOUNT OF HEAT
Emergency Gas Treatment	20 ± 2.0 kW
Auxiliary Building Gas Treatment	50 ± 5.0 kW

B

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

5.7.2.15 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure." The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures".

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Waste Gas Holdup System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); not B

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and > B

(continued)

5.0 ADMINISTRATIVE CONTROLS

5.8 Safety Function Determination Program (SFDP)

A

5.8.1 This program ensures loss of safety function is detected and appropriate actions taken. ~~Upon failure to meet two or more LCOs at the same time,~~ Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6.

5.8.2 The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

5.8.3 A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable (Case A); or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable (Case B); or

(continued)

5.9 Reporting Requirements

5.9.1 Routine Reports (continued)

5.9.1.2 Annual Reports

-----------------------------NOTE-----------------------------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted by ~~March 31~~ of each year. The initial report shall be submitted by ~~March 31~~ April 30 of the year following initial criticality. 4

Reports required on an annual basis include:

a. Occupational Radiation Exposure Report

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures > 100 mrem/yr and their associated man rem exposure according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance, waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.~~407~~ 2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions. 4

5.9.1.3 Annual Radiological Environmental Operating Report

-----------------------------NOTE-----------------------------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

(continued)

5.9 Reporting Requirements

5.9.1.3 Annual Radiological Environmental Operating Report (continued)

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements [in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979]. The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.9.1.4 Radioactive Effluent Release Report

------------------------------NOTE------------------------------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit during the previous 12 months of operations shall be submitted ~~within 90 days after January 1 of~~ each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (5)

(continued)

5.9 Reporting Requirements

5.9.1.4 Radioactive Effluent Release Report (continued)

consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a, and 10 CFR 50, Appendix I, Section IV.B.1, and 10 CFR 20.2107(a). (4)

5.9.1.5 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.9.1.6 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

LCO 3.1.4	Moderator Temperature Coefficient
LCO 3.1.6	Shutdown Bank Insertion Limit
LCO 3.1.7	Control Bank Insertion Limits
LCO 3.2.1	Heat Flux Hot Channel Factor
LCO 3.2.2	Nuclear Enthalpy Rise Hot Channel Factor
LCO 3.2.3	Axial Flux Difference
LCO 3.9.1	Boron Concentration

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary). (Methodology for Specifications 3.1.4 - Moderator Temperature Coefficient, 3.1.6 - Shutdown Bank Insertion Limit, 3.1.7 - Control Bank Insertion Limits, 3.2.1 - Heat Flux Hot Channel Factor, 3.2.2 - Nuclear Enthalphy Rise Hot Channel Factor, 3.2.3 - Axial Flux Difference, and 3.9.1 - Boron Concentration.

(continued)

5.10 Record Retention

5.10.3 (continued)

d. Records of gaseous and liquid radioactive material released to the environs;

e. Records of transient or operational cycles for those unit components identified in FSAR, Section 5.2.1.5;

f. Records of reactor tests and experiments;

g. Records of training and qualification for members of the unit staff;

h. Records of inservice inspections performed pursuant to the TS;

(A)

i. Records of quality assurance activities required by the Operational Quality Assurance (QA) Plan not listed in Specification 5.10.1 or Specification 5.10.2 and which are classified as permanent records by applicable regulations, codes, and standards;

j. Records of reviews performed for changes made to procedures, equipment, or reviews of tests and experiments pursuant to 10 CFR 50.59;

k. Records of the reviews and audits required by Specification 5.5.1 and Specification 5.5.2;

l. Records of the service lives of all hydraulic and mechanical snubbers required by Technical Requirement (TR) 3.7.3, "Snubbers", including the date at which the service life commences, and associated installation and maintenance records;

m. Records of secondary water sampling and water quality;

n. Records of analyses required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analysis at a later date (these records should include procedures effective at specified times and QA records showing that these procedures were followed);

o. Records of reviews performed for changes made to the Offsite Dose Calculation Manual and the Process Control Program; and

p. Records of steam generator tube surveillances.

5.0 ADMINISTRATIVE CONTROLS

5.11 High Radiation Area

1601(a)

1601(c)

5.11.1 Pursuant to 10 CFR 20, paragraph 20.~~203(c)(5)~~, in lieu of the requirements of 10 CFR 20.~~203(c)~~, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Radiological Control Technicians) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. (4)

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Radiological Controls Manager in the RWP.

5.11.2 In addition to the requirements of Specification 5.11.1, areas with radiation levels ≥ 1000 mrem/hr shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Operations Supervisor on duty or Radiological Controls supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP that shall specify the

(continued)