50-390 WATTS BAR 1 TVA

PROPOSED CHANGE TO TECH SPECS AND BASES, MARKUP

REC'D W/LTR DTD 8/27/92...9209020348

# 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 distributions, 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," GDC 26, "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.

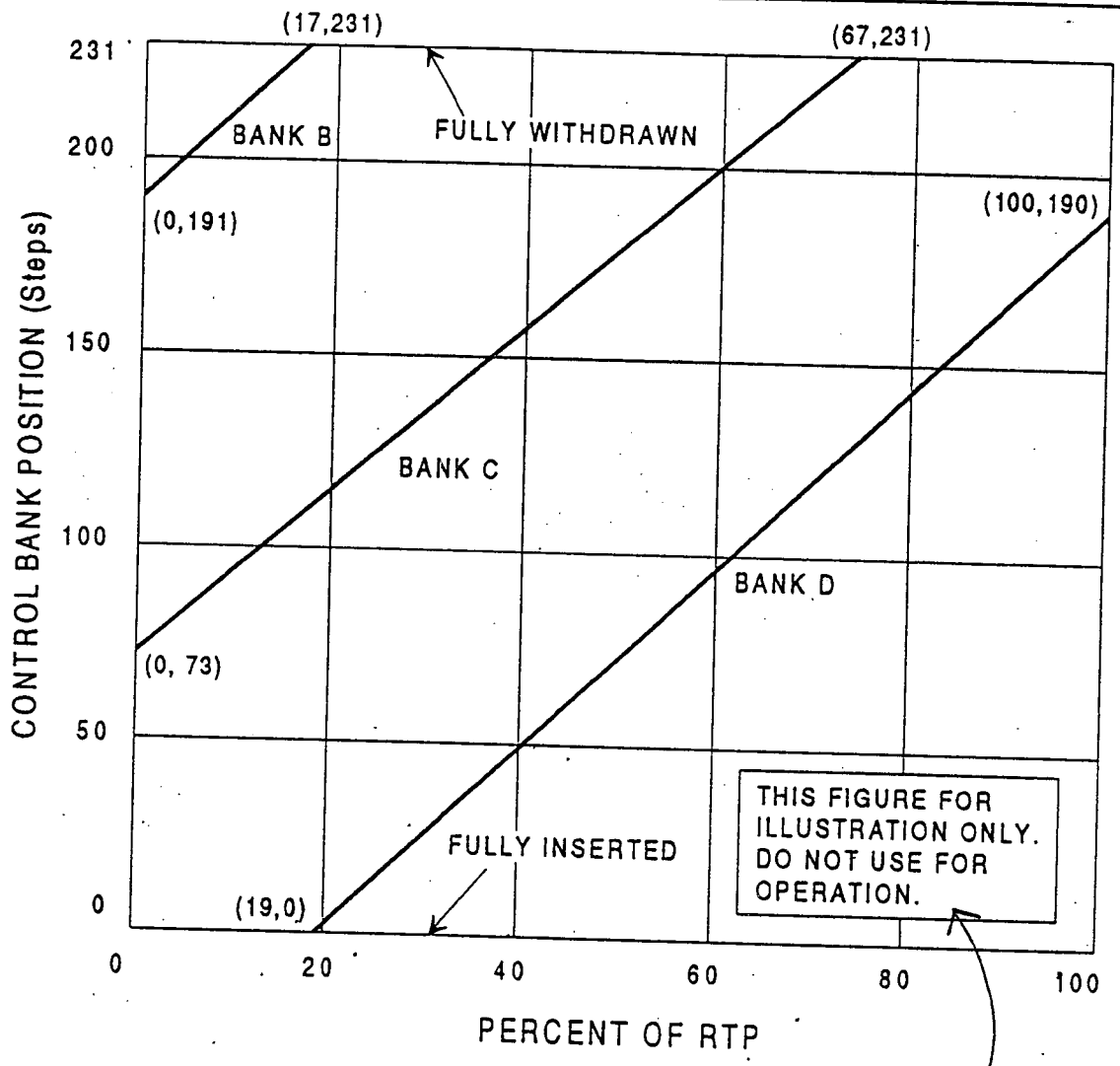
The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. ~~All~~ There are ~~plants have~~ four control banks and ~~at least two~~ 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 position of control bank C, at which control bank D will begin to move with bank C on a withdrawal, ~~will~~ as an example may be at

(continued)

BASES (continued)




Replace figure with INSERT FROM PAGE B 3.1-41A

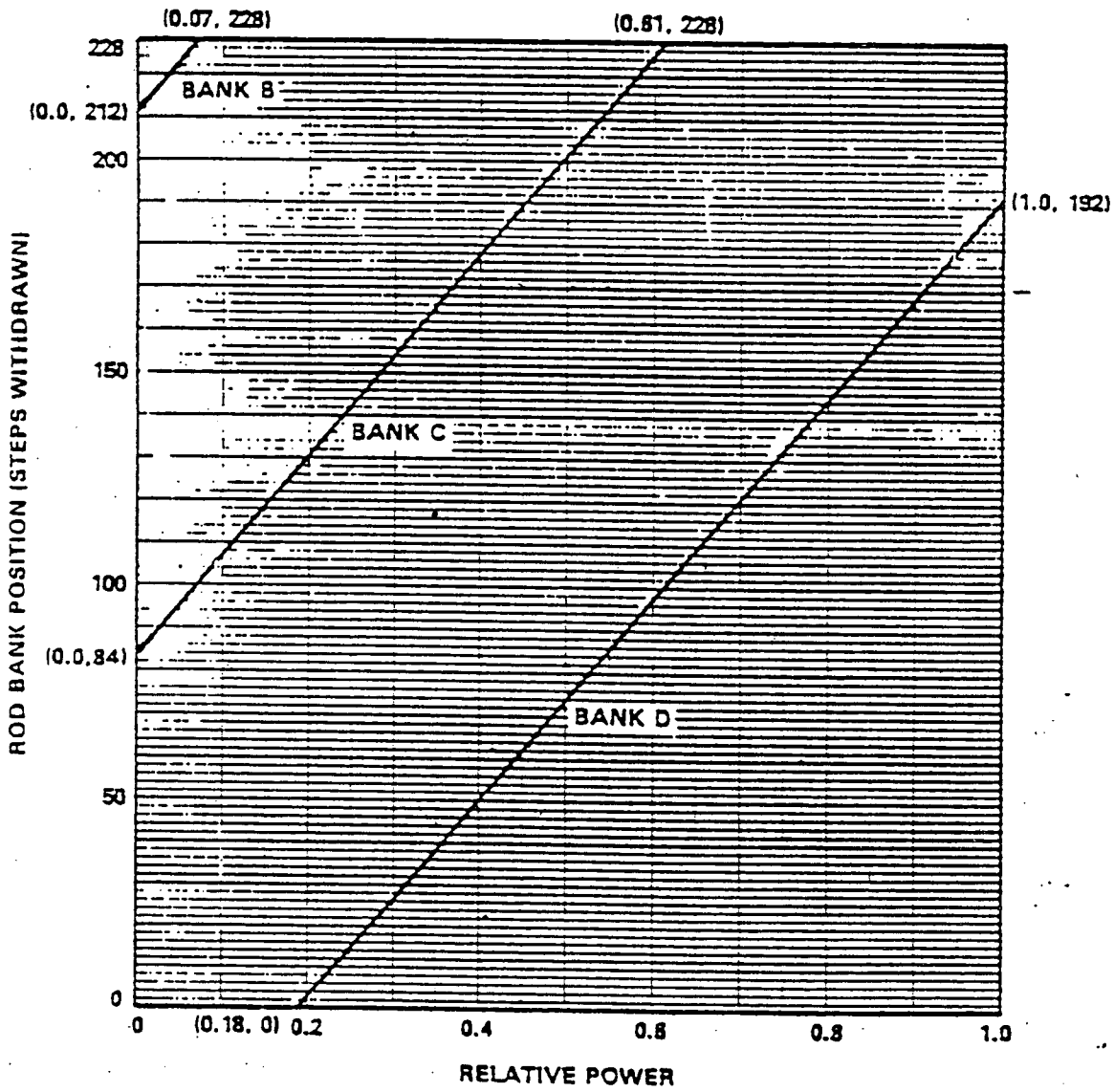
3

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

(continued)

③

# INSERT


(0.07, 228)
(0.61, 228)
228
BANK B
(0.0, 212)
200
(1.0, 192)
150
BANK C
100
(0.0,84)
BANK D
50
0
0
(0.18, 0)
0.2
0.4
0.6
0.8
1.0
ROD BANK POSITION (STEPS WITHDRAWN)
RELATIVE POWER

BASES

(4) Therefore, in this example, control bank C overlaps control bank D from 128 steps to the fully withdrawn position for control bank C.

BACKGROUND (continued)

(2) 128 steps. ~~for a fully withdrawn position of 231 steps~~. The fully withdrawn position ~~is~~ and predetermined overlap positions are defined in the COLR.

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally controlled automatically by the Rod Control System, but can also be manually controlled. They are capable of adding reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited, so that the fuel design criteria are maintained. Together, LCO 3.1.5, LCO 3.1.6, "Shutdown Bank Insertion Limits," LCO 3.1.7, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," provide limits on control component operation and on monitored process variables, which ensure that the core operates within the fuel design criteria.

The shutdown and control bank insertion and alignment limits, AFD, and QPTR are process variables that together characterize and control the three-dimensional power distribution of the reactor core. Additionally, the control bank insertion limits control the reactivity that could be added in the event of a rod ejection accident, and the shutdown and control bank insertion limits ensure the required SDM is maintained.

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission-product barrier and release fission products to the reactor coolant in the event of a loss-of-coolant accident (LOCA), loss of flow, ejected rod, or other accident requiring termination by a Reactor Trip System (RTS) trip function.

APPLICABLE SAFETY ANALYSES

The shutdown and control bank insertion limits, AFD, and QPTR LCOs are required to prevent power distributions that could result in fuel cladding failures in the event of a LOCA, loss of flow, ejected rod, or other accident requiring termination by an RTS trip function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The acceptance criteria for addressing shutdown and control bank insertion limits and inoperability or misalignment are that:

a. There be no violations of:

1. specified acceptable fuel design limits, or
2. Reactor Coolant System pressure boundary integrity; and

b. The core remains subcritical after accident transients, other than a main steam line break (MSLB). (9)

As such, the shutdown and control bank insertion limits affect safety analysis involving core reactivity and power distributions (Ref. ~~3~~). 3 through 13 (5)

The SDM requirement is ensured by limiting the control and shutdown bank insertion limits so that allowable inserted worth of the RCCAs is such that sufficient reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin that assumes the maximum worth RCCA remains fully withdrawn upon trip (Ref. ~~4~~). (5) 5,6,8 and 11

Operation at the insertion limits or AFD limits may approach the maximum allowable linear heat generation rate or peaking factor with the allowed QPTR present. Operation at the insertion limit may also indicate the maximum ejected RCCA worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected RCCA worths.

The control and shutdown bank insertion limits ensure that safety analyses assumptions for SDM, ejected rod worth, and power distribution peaking factors are preserved (Ref. ~~3~~). 3 through 13

The insertion limits satisfy Criterion 2 of the NRC Policy Statement, in that they are initial conditions assumed in the safety analysis. (5)

LCO

The limits on control banks sequence, overlap, and physical insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected rod worth is maintained, and ensuring adequate

(continued)

BASES

LCO (continued)

negative reactivity insertion is available on trip. The overlap between control banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during control bank motion.

APPLICABILITY

The control bank sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2 with $k_{eff} \geq 1.0$. These limits must be maintained, since they preserve the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions. Applicability in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES.

The applicability requirements have been modified by a Note indicating the LCO requirements are suspended during the performance of SR 3.1.5.2. This SR verifies the freedom of the rods to move, and requires the control bank to move below the LCO limits, which would violate the LCO.

ACTIONS

A.1.1, A.1.2, A.2, B.1.1, B.1.2, and B.2

When the control banks are outside the acceptable insertion limits, they must be restored to within those limits. This restoration can occur in two ways:

a. Reducing power to be consistent with rod position; or

b. Moving rods to be consistent with power.

Also, verification of SDM or initiation of boration to regain SDM is required within 1 hour, since the SDM in MODES 1 and 2 normally ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1, "SHUTDOWN 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 SR 3.1.1.1.

(continued)

BASES

ACTIONS

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

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 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.

6

In the event that a rod group is found to be inoperable, the rod group is considered to be not within limits, and Required Actions A.2 or B.2 and LCO 3.1.5 apply.

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.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits.

The estimated critical position (ECP) depends upon a number of factors, one of which is xenon concentration. If the ECP was calculated long before criticality, xenon concentration

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1 (continued)

could change to make the ECP substantially in error. Conversely, determining the ECP immediately before criticality could be an unnecessary burden. There are a number of unit parameters requiring operator attention at that point. Performing the ECP calculation within 4 hours prior to criticality avoids a large error from changes in xenon concentration, but allows the operator some flexibility to schedule the ECP calculation with other startup activities.

SR 3.1.7.2

With an OPERABLE bank insertion limit monitor, verification of the control bank insertion limits at a Frequency of 12 hours is sufficient to ensure OPERABILITY of the bank insertion limit monitor and to detect control banks that may be approaching the insertion limits since, normally, very little rod motion occurs in 12 hours. If the insertion limit monitor becomes inoperable, verification of the control bank position at a Frequency of 4 hours is sufficient to detect control banks that may be approaching the insertion limits. 1

SR 3.1.7.3

When control banks are maintained within their insertion limits as checked by SR 3.1.7.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. A Frequency of 12 hours is consistent with the insertion limit check above in SR 3.1.7.2.

7 REFERENCES

INPUT INSERT FROM Page B 3.1-46A

~~1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.~~

~~2. 10 CFR 50.46.~~

~~3. FSAR, Chapter [15].~~

~~4. FSAR, Chapter [15].~~

~~5. FSAR, Chapter [15].~~

7

# INSERT

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

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 Locker Rotor."

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

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 3.1.7

1. Change to correct error in STS.

2. Change to reflect WBN specific parameter values.

3. Change to reflect WBN specific control bank insertion figure.

4. Change in text to more clearly describe control bank overlap for control banks C and D at WBN.

5. Change in reference numbers to reflect addition of several more specific FSAR section references.

6. Deleted text that was considered confusing. LCO 3.1.8 provides actions for inoperable rod position indication which allows determining rod position using the incore detectors once per 8 hours. Actions A.2 and B.2 of this LCO apply to insertion limit and overlap. LCO 3.1.8 applies to alignment. It is unclear what this paragraph intends with respect to "inoperability" of a rod group. The other important issues are trippability and position which are already addressed by LCOs 3.1.5 and 3.1.8, respectively.

7. Change to specify WBN specific reference information and format.

8. This change reflects comments made by the industry to the NRC at the Proof and Review meeting on the RSTS in Irvine, CA July 13 - July 20, 1992.

9. This statement is incorrect for the MSLB event when the core briefly returns critical.

## 3.1 REACTIVITY CONTROL SYSTEMS

### 3.1.8 Rod Position Indication

LCO 3.1.8 The ~~[Digital]~~ Analog (1) Rod Position Indication (~~Ø~~A RPI) (1) System and the Demand Position Indication System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each inoperable rod position indicator per group and each demand position indicator per bank.
---

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. (1) One ~~Ø~~A RPI per group inoperable for one or more groups. | A.1 | Verify the position of the rods with inoperable position indicators by using movable incore detectors. | Once per 8 hours |
| | OR | | |
| | A.2 | Reduce THERMAL POWER to ≤ 50% RTP. | 8 hours |
| B. One or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction since the last determination of the rod's position. | B.1 | Verify the position of the rods with inoperable position indicators by using movable incore detectors. | 8 hours |
| | OR | | |

(continued)

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| B. (continued) | B.2 Reduce THERMAL POWER to ≤ 50% RTP. | 8 hours |
| C. One demand position indicator per bank inoperable for one or more banks. | C.1.1 Verify by administrative means all A DRPIs for the affected banks are OPERABLE. | Once per 8 hours |
| | AND | |
| | C.1.2 Verify the most withdrawn rod and the least withdrawn rod of the affected banks are ≤ 12 steps apart. | Once per 8 hours |
| | OR | |
| | C.2 Reduce THERMAL POWER to ≤ 50% RTP. | 8 hours |
| D. Required Action and associated Completion Time of Condition A, Condition B, or Condition C not met. | D.1 Be in MODE 3. | 6 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.1.8.1 | Verify each A DRPI agrees within [12] steps of the group demand position for the full indicated range of rod travel. | [18 months] |

B 3.1 REACTIVITY CONTROL SYSTEMS (6)

B 3.1.8 Rod Position Indication

BASES

BACKGROUND

According to GDC 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1.8 is required to ensure OPERABILITY of the control rod position indicators to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The OPERABILITY, including position indication, of the shutdown and control rods are initial assumptions 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. Rod position indication is required to assess OPERABILITY and misalignment.

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 and OPERABILITY 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 out of the core (up or withdrawn) or into the core (down or inserted) by their control rod drive mechanisms. 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.

(continued)

BASES

BACKGROUND (continued)

The axial position of shutdown rods and control rods are determined by two separate and independent systems: the Bank Demand Position Indication System (commonly called group step counters) and the ~~[Digital]~~ ANAlog Rod Position Indication (ØRPI) System. (1)

The Bank Demand Position Indication System counts the pulses from the Rod Control System that move the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± $\frac{5}{8}$ inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

(1) The ØRPI System provides a ~~highly~~ (3) accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube with a center-to-center distance of 3.75 inches, which is 6 steps. ~~To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one system fails, the DRPI will go on half accuracy with an effective coil spacing of 7.5 inches, which is 12 steps. Therefore,~~ (4) (1) The normal indication accuracy of the ØRPI System is ± 6 steps (± 3.75 inches), and the maximum uncertainty is ± 12 steps (± 7.5 inches). With an indicated deviation of 12 steps between the group (1) step counter and ØRPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches.

APPLICABLE SAFETY ANALYSES

(6) Control rod and shutdown position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2), with control or shutdown rods operating outside their limits undetected. Therefore, the acceptance criteria for rod position indication is that rod positions must be known with sufficient accuracy in order to verify the core is operating within the group sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO 3.1.6, "Shutdown Bank Insertion Limits," and

2 through 12

(5)

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

LCO 3.1.7, "Control Bank Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.5, "Rod Group Alignment Limits"). Control rod positions are continuously 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.~~7~~ 8 specifies that ~~one~~ the ~~Ø~~A RPI System and ~~one~~ the Bank Demand Position Indication System be OPERABLE for ~~each~~ control rods. For the control rod position indicators to be OPERABLE requires meeting the SR of the LCO and the ALL following:

a. The ~~Ø~~A RPI System has passed a CHANNEL FUNCTIONAL CHECK within the prescribed interval;

b. For the ~~Ø~~A RPI 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 ~~Ø~~A RPI System.

The agreement between the Bank Demand Position Indication System and the ~~Ø~~A RPI System is within the limit, indicating that the Bank Demand Position Indication System is adequately calibrated for measurement of control rod bank position.

A deviation of less than the allowable limit, given in the COLR, 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).

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)

BASES

LCO (continued)

inoperable, misaligned, or mispositioned control rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

APPLICABILITY

The requirements on the ØRPI and step counters are only applicable in MODES 1 and 2 (consistent with LCO 3.1.5, LCO 3.1.6, and LCO 3.1.7), because these are the only MODES in which power is generated, and the OPERABILITY and alignment of rods have the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System.

ACTIONS

The Actions table is modified by a NOTE indicating that a separate Condition entry is allowed for each inoperable rod position indicator per group and each demand position indicator per bank. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator.

A.1

When one ØRPI channel per group fails, the position of the rod can still be determined by use of the incore movable detectors. Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, the Required Action of B.1 or B.2 below is required. Therefore, verification of RCCA position within the Completion Time of 8 hours is adequate for allowing continued full-power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

(continued)

BASES

ACTIONS (continued)

A.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 13). 5

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to ≤ 50% RTP from full-power conditions without challenging plant systems and allowing for rod position determination by Required Action A.1 above.

B.1 and B.2

These Required Actions clarify that when one or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction, since the position was last determined, the Required Actions of A.1 and A.2 are still appropriate but must be initiated promptly under Required Action B.1 to begin verifying that these rods are still properly positioned, relative to their group positions.

If, within 8 hours, the rod positions have not been determined, THERMAL POWER must be reduced to ≤ 50% RTP to avoid undesirable power distributions that could result from continued operation at > 50% RTP, if one or more rods are misaligned by more than 24 steps. The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions.

C.1.1 and C.1.2

A 1

With one demand position indicator per bank inoperable, the rod positions can be determined by the ~~D~~RPI System. Since normal power operation does not require excessive movement of rods, verification by administrative means that the rod position indicators are OPERABLE and the most withdrawn rod and the least withdrawn rod are ≤ 12 steps apart within the allowed Completion Time of once every 8 hours is adequate. 9

(continued)

BASES (continued)

ACTIONS (continued)

C.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factor limits (Ref. 13). The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions per Required Actions C.1.1 and C.1.2 or reduce power to ≤ 50% RTP.

D.1

If the Required Actions cannot be completed within the associated Completion Time, 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. The allowed Completion Time 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 3.1.8.1

1 A A

2 Verification that the ~~D~~RPI agrees with the demand position within [12] steps ensures that the ~~D~~RPI is operating correctly. ~~Since the DRPI does not display the actual shutdown rod positions between 18 and 210 steps, only points within the indicated ranges are required in comparison.~~ 4

2 2

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 unnecessary plant transients if the SR were performed with the reactor at power. Operating experience has shown these components virtually always pass the SR when performed at a Frequency of once every [18 months.] Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. 2

8 REFERENCES

Input Insert from Page B 3.1-52A

~~1. 10 CFR 50, Appendix A, GDC 13.~~

~~2. FSAR, Section [15].~~

~~3. FSAR, Section [15].~~

(8)

# INSERT

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 13, "Instrumentation and Control."
2. WATTS BAR FSAR, Section 15.2.1, "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition."
3. WATTS BAR FSAR, Section 15.2.2, "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal At Power."
4. WATTS BAR FSAR, Section 15.2.3, "Rod Cluster Control Assembly Misalignment."
5. WATTS BAR FSAR, Section 15.2.4, "Uncontrolled Boron Dilution."
6. WATTS BAR FSAR, Section 15.2.5, "Partial Loss of Forced Reactor Coolant Flow."
7. WATTS BAR FSAR, Section 15.2.13, "Accidental Depressurization of the Main Steam System."
8. WATTS BAR FSAR, Section 15.3.4, "Complete Loss of Forced Reactor Coolant Flow."
9. WATTS BAR FSAR, Section 15.3.6, "Single Rod Cluster Control Assembly Withdrawal At Full Power."
10. WATTS BAR FSAR, Section 15.4.2.1, "Major Rupture of Main Steam Line."
11. WATTS BAR FSAR, Section 15.4.4, "Single Reactor Coolant Pump Locker Rotor."
12. WATTS BAR FSAR, Section 15.4.6, "Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)."
13. Watts Bar FSAR, Section 4.3, "Nuclear Design."

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 3.1.8

1. WBN utilizes an Analog Rod Position Indication System, not a Digital Rod Position Indication System.

2. Format change to delete brackets that identify plant specific information/values.

3. WBN does not consider the ARPI System to be "highly" accurate.

4. Deleted text that does not apply to WBN's ARPI System.

5. Change in reference numbers to reflect addition of several more specific FSAR section references.

6. Change to correct error in STS.

7. Editorial change of STS text to more clearly specify LCO requirements for WBN.

8. Change to specify WBN specific reference information and format.

9. This change reflects comments made by the industry to the NRC at the Proof and Review meeting on the RSTS in Irvine, CA July 13 - July 20, 1992.

## 3.1 REACTIVITY CONTROL SYSTEMS

### 3.1.9 PHYSICS TESTS Exceptions – MODE 1

LCO 3.1.9 During the performance of PHYSICS TESTS, the requirements of

LCO 3.1.5, "Rod Group Alignment Limits";
LCO 3.1.6, "Shutdown Bank Insertion Limits";
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)"

may be suspended, provided:

a. THERMAL POWER is maintained ≤ 85% RTP;

b. Power Range Neutron Flux – High trip setpoints are ≤ 10% RTP above the THERMAL POWER at which the test is performed, with a maximum setting of 90% RTP; and

c. SDM is ≥ 1.6% Δk/k.

1

APPLICABILITY: MODE 1 during PHYSICS TESTS.

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. SDM not within limit. | A.1 | Initiate boration to restore SDM to within limit. | 15 minutes |
| | AND | | |
| | A.2 | Suspend PHYSICS TESTS exceptions. | 1 hour |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| B. THERMAL POWER not within limit. | B.1 | Reduce THERMAL POWER to within limit. | 1 hour |
| | OR | | |
| | B.2 | Suspend PHYSICS TESTS exceptions. | 1 hour |
| C. Power Range Neutron Flux – High trip setpoints > 10% RTP above the PHYSICS TEST power level.<br><br>OR<br><br>Power Range Neutron Flux – High trip setpoints > 90% RTP. | C.1 | Restore Power Range Neutron Flux – High trip setpoints to ≤ 10% above the PHYSICS TEST power level, or to ≤ 90% RTP, whichever is lower. | 1 hour |
| | OR | | |
| | C.2 | Suspend PHYSICS TESTS exceptions. | 1 hour |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.1.9.1 | Verify THERMAL POWER is ≤ 85% RTP. | 1 hour |
| SR 3.1.9.2 | Verify Power Range Neutron Flux – High trip setpoints are ≤ 10% above the PHYSICS TESTS power level, and ≤ 90% RTP. 2 | Within 8 hours prior to initiation of PHYSICS TESTS |
| SR 3.1.9.3 | Perform SR 3.2.1.1 and SR 3.2.2.1. | 12 hours |
| SR 3.1.9.4 | Verify SDM is ≥ 1.6% Δk/k. 1 | 24 hours |

## B 3.1 REACTIVITY CONTROL SYSTEMS

## B 3.1.9 PHYSICS TESTS Exceptions – MODE 1

BASES

BACKGROUND

The primary purpose of the MODE 1 PHYSICS TESTS exceptions is to permit relaxations of existing LCOs to allow the performance of instrumentation calibration tests and special PHYSICS TESTS. The exceptions to LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)" are most often appropriate for xenon stability tests. The exceptions to LCO 3.1.5, "Rod Group Alignment Limits"; LCO 3.1.6, "Shutdown Bank Insertion Limit"; and LCO 3.1.7, "Control Bank Insertion Limits," may be required in the event that it is necessary or desirable to do special PHYSICS TESTS involving abnormal rod or bank configurations.

Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

The key objectives of a test program are to (Ref. 3):

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in the design and analysis;

c. Verify the assumptions used to predict unit response;

d. Ensure that installation of equipment at the facility has been accomplished, in accordance with the design; and

e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality; during startup, low power, power

(continued)

BASES

BACKGROUND (continued)

ascension, and at-power operation; and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions, and that the core can be operated as designed (Ref. 4).

PHYSICS TESTS procedures are written and approved, in accordance with established formats. The procedures include all information necessary to permit a detailed execution of the testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures, and test results are approved prior to continued power escalation and long-term power operation.

The PHYSICS TESTS required for reload fuel cycles (Ref. 4) in MODE 1 are listed below:

a. Neutron Flux Symmetry;

b. Power Distribution – Intermediate Power;

c. Power Distribution – Full Power; and

d. Critical Boron Concentration – Full Power.

The first test can be performed in either MODE 1 or 2, and the last three tests are performed in MODE 1. These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance. The last two tests are performed at ≥ 90% RTP.

1

a. The Neutron Flux Symmetry Test measures the degree of azimuthal symmetry of the core neutron flux at as low a power level as practical, depending on the method used. The Flux Distribution Method uses incore flux detectors to measure the azimuthal flux distribution at selected locations with the core at ≤ 30% RTP.

b. The Power Distribution – Intermediate Power Test measures the power distribution of the reactor core at intermediate power levels between 40% and 75% RTP. This test uses the incore flux detectors to measure core power distribution.

(continued)

BASES (continued)

BACKGROUND (continued)

c. The Power Distribution – Full Power Test measures the power distribution of the reactor core at ≥ 90% RTP using incore flux detectors.

d. The Critical Boron Concentration – Full Power Test simply measures the critical boron concentration at > 90% RTP, with all rods fully withdrawn, the lead control bank being at or near its fully withdrawn position, and with the core at equilibrium xenon conditions.

1 6

6

it may be necessary to perform special PHYSICS TESTS involving abnormal rod or bank configurations

For initial startups, there are two currently required tests that violate the referenced LCO. The pseudo-ejected rod test, performed at approximately 30% RTP, and the pseudo-dropped rod test, performed at approximately 50% RTP, require individual rod misalignments that exceed the limits specified in the relevant LCO.

APPLICABLE SAFETY ANALYSES

The fuel is protected by an LCO, which preserves the initial conditions of the core assumed during the safety analyses. The methods for development of the LCO, which are superseded by this LCO, are described in the Westinghouse Reload Safety Evaluation Methodology Report (Ref. 5). The above-mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational problems, may require the operating controls or process variables to deviate from their LCO limitations.

3

Reference 6 defines requirements for initial testing of the facility, including PHYSICS TESTS. Tables ~~[14.1-1 and~~ 14.1-2 (Ref. 6) summarize the zero, low power, and power tests. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits for all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in:

LCO 3.1.5, "Rod Group Alignment Limits";
LCO 3.1.6, "Shutdown Bank Insertion Limits";
LCO 3.1.7, "Control Bank Insertion Limits";
LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; or

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)"

are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the requirements of LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)," and LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," are satisfied, power level is maintained ≤ 85% RTP, and SDM is ≥ 1.6% Δk/k. Therefore, LCO 3.1.9 requires surveillance of the hot channel factors and SDM to verify that their limits are not being exceeded.

1

PHYSICS TESTS include measurements of core nuclear parameters or the exercise of control components that affect process variables. Among the process variables involved are AFD and QPTR, which represent initial conditions of the unit safety analyses. Also involved are the movable control components (control and shutdown rods), which are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR.

PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the component and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of the NRC Policy Statement.

Reference 7 allows special test exceptions to be included as part of the LCO that they affect. However, it was decided to retain this special test exception as a separate LCO because it was less cumbersome and provided additional clarity.

LCO

This LCO allows selected control rods and shutdown rods to be positioned outside their specified alignment limits and insertion limits to conduct PHYSICS TESTS in MODE 1, to verify certain core physics parameters. The power level is limited to ≤ 85% RTP and the power range neutron flux trip setpoint is set at 10% RTP above the PHYSICS TESTS power level with a maximum setting of 90% RTP. Violation of LCO 3.1.5, LCO 3.1.6, LCO 3.1.7, LCO 3.2.3, or LCO 3.2.4, during the performance of PHYSICS TESTS does not pose any threat to the integrity of the fuel as long as the

(continued)

BASES

LCO (continued)

requirements of LCO 3.2.1 and LCO 3.2.2, are satisfied, and provided:

a. THERMAL POWER is maintained ≤ 85% RTP;

b. Power Range Neutron Flux – High trip setpoints are ≤ 10% RTP above the THERMAL POWER at which the test is performed, with a maximum setting of 90% RTP; and

c. SDM is ≥ 1.6% Δk/k.

Operation with THERMAL POWER ≤ 85% RTP during PHYSICS TESTS provides an acceptable thermal margin when one or more of the applicable LCOs is out of specification. The Power Range Neutron Flux – High trip setpoint is reduced so that a similar margin exists between the steady-state condition and the trip point that exists during normal operation at RTP.

APPLICABILITY

This LCO is applicable in MODE 1 when performing PHYSICS TESTS. The applicable PHYSICS TESTS are performed at ≤ 85% RTP. Other PHYSICS TESTS are performed at full power but do not require violation of any existing LCO, and therefore do not require a PHYSICS TESTS exception. The PHYSICS TESTS performed in MODE 2 are covered by LCO 3.1.10, "PHYSICS TESTS Exceptions – MODE 2."

ACTIONS

A.1 and A.2

If the SDM requirement is 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. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

When THERMAL POWER is > 85% RTP, the only acceptable actions are to reduce THERMAL POWER to ≤ 85% RTP or to suspend the PHYSICS TESTS exceptions. With the PHYSICS TESTS exceptions suspended, the PHYSICS TESTS may proceed if all other LCO requirements are met. Fuel integrity may be challenged with control rods or shutdown rods misaligned and THERMAL POWER > 85% RTP. The allowed Completion Time of 1 hour is reasonable, based on operating experience, for completing the Required Actions in an orderly manner and without challenging plant systems. This Completion Time is also consistent with the Required Actions of the LCOs that are suspended by the PHYSICS TESTS.

C.1 and C.2

When the Power Range Neutron Flux – High trip setpoints are > 10% RTP above the PHYSICS TESTS power level or > 90% RTP, the Reactor Trip System (RTS) may not provide the required degree of core protection if the trip setpoint is greater than the specified value.

The only acceptable actions are to restore the trip setpoint to the allowed value or to suspend the performance of the PHYSICS TESTS exceptions. The Completion Time of 1 hour is based on the practical amount of time it may take to restore the Neutron Flux – High trip setpoints to the correct value, consistent with operating plant safety. This Completion Time is consistent with the Required Actions of the LCOs that are suspended by the PHYSICS TESTS.

SURVEILLANCE REQUIREMENTS

SR 3.1.9.1

Verification that the THERMAL POWER level is ≤ 85% RTP will ensure that the required core protection is provided during the performance of PHYSICS TESTS. Control of the reactor power level is a vital parameter and is closely monitored during the performance of PHYSICS TESTS. A Frequency of 1 hour is sufficient for ensuring that the power level does not exceed the limit.

(continued)

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. Verifying the trip setpoint at a frequency of 8 hours during the performance of the PHYSICS TESTS ensures that the RTS will provide the required core protection.

4

SR 3.1.9.3

the 2

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, 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

3 f. Design isothermal temperature coefficient (ITC).

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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.9.4 (continued)

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.

REFERENCES

5

1. ~~10 CFR 50, Appendix B, Section XI, 1988.~~
2. ~~10 CFR 50.59.~~
3. ~~Regulatory Guide 1.68, Revision 2, August 1978.~~
4. ~~ANSI/ANS 19.6.1 1985, December 13, 1985.~~
5. ~~WCAP 9273 NP A, "Westinghouse Reload Safety Evaluation Methodology Report", July 1985.~~
6. ~~FSAR, Section [14].~~
7. ~~WCAP 11618, November 1987, and Addendum 1, April 1989.~~

1. Title 10, Code of Federal Regulations, Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," 1988.
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.
4. ANSI/ANS-19.6.1-1985, "Reload Startup PHYSICS TESTS for Pressurized Water Reactors," American National Standards Institute, December 13, 1985.
5. WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology Report," July 1985.
6. WATTS BAR FSAR, Section ~~14~~ 14.2, "~~Initial~~ Test ~~and~~ Program."
7. WCAP-11618, "MERITS Program — Phase II, Task 5, Criteria Application," dated November 1987, including Addendum 1, April 1989.

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 3.1.9

1. Format change to delete brackets that identify plant specific information/values.

2. Change to correct error in the STS.

3. Change to reflect WBN specific information.

4. Deleted text due to an editorial error in the STS. SR 3.1.9.2 is only required 8 hours prior to physics tests, not once per 8 hours.

5. Change to specify WBN specific reference information and format.

6. Watts Bar FSAR Amendment 69 takes exception to these listed tests and therefore, the wording has been changed to address generic abnormal rod configurations during physics testing that require this exception.

## 3.1 REACTIVITY CONTROL SYSTEMS

### 3.1.10 PHYSICS TESTS Exceptions – MODE 2

LCO 3.1.10 During the performance of PHYSICS TESTS, the requirements of

LCO 3.1.4, "Moderator Temperature Coefficient (MTC)";
LCO 3.1.5, "Rod Group Alignment Limits";
LCO 3.1.6, "Shutdown Bank Insertion Limits";
LCO 3.1.7, "Control Bank Insertion Limits"; and
LCO 3.4.2, "RCS Minimum Temperature for Criticality"

may be suspended, provided:

5 ~~a. THERMAL POWER is maintained ≤ 5% RTP;~~ 1

a ~~b~~. RCS lowest loop average temperature is ≥ ~~[531]~~ 541 °F; and

b ~~c~~. SDM is ≥ [1.6]% Δk/k. 2

APPLICABILITY: MODE 2 during PHYSICS TESTS.

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. SDM not within limit. | A.1 | Initiate boration to restore SDM to within limit. | 15 minutes |
| | AND | | |
| | A.2 | Suspend PHYSICS TESTS exceptions. | 1 hour |
| B. THERMAL POWER not within limit. | B.1 | Open reactor trip breakers. | Immediately |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| C. RCS lowest loop average temperature not within limit. | C.1 Restore RCS lowest loop average temperature to within limit. | 15 minutes |
| | OR | |
| | C.2 Be in MODE 3. | 30 minutes |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.1.10.1 | Perform ~~an ANALOG~~ a CHANNEL OPERATIONAL TEST on power-range and intermediate-range channels per SR 3.3.1.7, SR 3.3.1.8, and Table 3.3.1-1. | Within 12 hours prior to initiation of PHYSICS TESTS |
| SR 3.1.10.2 | Verify the RCS lowest loop average temperature is ~~within limit~~. ≥ 541°F. | 30 minutes |
| SR 3.1.10.3 | Verify THERMAL POWER is ≤ 5% RTP. | 1 hour |
| SR 3.1.10.~~4~~4 | Verify SDM is ≥ 1.6% Δk/k. | 24 hours |

# B 3.1 REACTIVITY CONTROL SYSTEMS

## B 3.1.10 PHYSICS TESTS Exceptions – MODE 2

BASES

BACKGROUND

The primary purpose of the MODE 2 PHYSICS TESTS exceptions is to permit relaxations of existing LCOs to allow certain PHYSICS TESTS to be performed.

Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

The key objectives of a test program are to (Ref. 3):

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in the design and analysis;

c. Verify the assumptions used to predict unit response;

d. Ensure that installation of equipment in the facility has been accomplished, in accordance with the design; and

e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality, during startup, during low power operations, during power ascension, at high power, and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed (Ref. 4).

PHYSICS TESTS procedures are written and approved, in accordance with established formats. The procedures include

(continued)

BASES

BACKGROUND (continued)

all information necessary to permit a detailed execution of the testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long-term power operation.

The PHYSICS TESTS required for reload fuel cycles (Ref. 4) in MODE 2 are listed below:

a. Critical Boron Concentration – Control Rods Withdrawn;

b. Critical Boron Concentration – Control Rods Inserted;

c. Control Rod Group Worth;

d. Isothermal Temperature Coefficient (ITC); and

e. Neutron Flux Symmetry.

The first four tests are performed in MODE 2, and the last test can be performed in either MODE 1 or 2. These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance.

a. The Critical Boron Concentration – Control Rods Withdrawn Test measures the critical boron concentration at hot zero power (HZP). With all rods out, the lead control bank is at or near its fully withdrawn position. HZP is where the core is critical ($k_{eff}$ = 1.0), and the Reactor Coolant System (RCS) is at design temperature and pressure for zero power. Performance of this test should not violate any of the referenced LCOs.

b. The Critical Boron Concentration – Control Rods Inserted Test measures the critical boron concentration at HZP, with a bank having a worth of at least 1% $\Delta k/k$ when fully inserted into the core. This test is used to measure the boron reactivity coefficient. With the core at HZP and all banks fully withdrawn, the boron concentration of the reactor coolant is gradually lowered in a continuous manner. The selected bank is

(continued)

BASES

BACKGROUND (continued)

then inserted to make up for the decreasing boron concentration until the selected bank has been moved over its entire range of travel. The reactivity resulting from each incremental bank movement is measured with a reactivity computer. The difference between the measured critical boron concentration with all rods fully withdrawn and with the bank inserted is determined. The boron reactivity coefficient is determined by dividing the measured bank worth by the measured boron concentration difference. Performance of this test could violate LCO 3.1.5, "Rod Group Alignment Limits"; LCO 3.1.6, "Shutdown Bank Insertion Limit"; or LCO 3.1.7, "Control Bank Insertion Limits."

c. The Control Rod Group Worth Test is used to measure the reactivity worth of selected control banks. This test is performed at HZP and has three alternative methods of performance. The first method, the Boron Exchange Method, varies the reactor coolant boron concentration and moves the selected control bank in response to the changing boron concentration. The reactivity changes are measured with a reactivity computer. This sequence is repeated for the remaining control banks. The second method, the Rod Swap Method, measures the worth of a predetermined reference bank using the Boron Exchange Method above. The reference bank is then nearly fully inserted into the core. The selected bank is then inserted into the core as the reference bank is withdrawn. The HZP critical conditions are then determined with the selected bank fully inserted into the core. The worth of the selected bank is inferred, based on the position of the reference bank with respect to the selected bank. This sequence is repeated as necessary for the remaining control banks. The third method, the Boron Endpoint Method, moves the selected control bank over its entire length of travel and then varies the reactor coolant boron concentration to achieve HZP criticality again. The difference in boron concentration is the worth of the selected control bank. This sequence is repeated for the remaining control banks. Performance of this test could violate LCO 3.1.5, LCO 3.1.6, or LCO 3.1.7.

d. The ITC Test measures the ITC of the reactor. This test is performed at HZP and has two methods of

(continued)

BASES

BACKGROUND (continued)

performance. The first method, the Slope Method, varies RCS temperature in a slow and continuous manner. The reactivity change is measured with a reactivity computer as a function of the temperature change. The ITC is the slope of the reactivity versus the temperature plot. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. The second method, the Endpoint Method, changes the RCS temperature and measures the reactivity at the beginning and end of the temperature change. The ITC is the total reactivity change divided by the total temperature change. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. Performance of this test could violate LCO 3.4.2, "RCS Minimum Temperature for Criticality."

e. The Flux Symmetry Test measures the degree of azimuthal symmetry of the neutron flux at as low a power level as practical, depending on the test method employed. This test can be performed at HZP (Control Rod Worth Symmetry Method) or at ≤ 30% RTP (Flux Distribution Method). The Control Rod Worth Symmetry Method inserts a control bank, which can then be withdrawn to compensate for the insertion of a single control rod from a symmetric set. The symmetric rods of each set are then tested to evaluate the symmetry of the control rod worth and neutron flux (power distribution). A reactivity computer is used to measure the control rod worths. Performance of this test could violate LCO 3.1.5, LCO 3.1.6, or LCO 3.1.7. The Flux Distribution Method uses the incore flux detectors to measure the azimuthal flux distribution at selected locations with the core at ≤ 30% RTP.

APPLICABLE SAFETY ANALYSES

The fuel is protected by LCOs that preserve the initial conditions of the core assumed during the safety analyses. The methods for development of the LCOs that are excepted by this LCO are described in the Westinghouse Reload Safety Evaluation Methodology Report (Ref. 5). The above-mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

problems, may require the operating control or process variables to deviate from their LCO limitations.

The COLR defines requirements for initial testing of the facility, including PHYSICS TESTS. Table~~s~~ ~~[14.1-1 and 14.1-2]~~ 14.2.2 summarize the zero, low power, and power tests. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits for all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in LCO 3.1.4, "Moderator Temperature Coefficient (MTC)," LCO 3.1.5, LCO 3.1.6, LCO 3.1.7, and LCO 3.4.2 are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to ≤ 5% RTP, the reactor coolant temperature is kept ≥ 531°F, and SDM is ≥ [1.6]% Δk/k.

The PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. Among the process variables involved are AFD and QPTR, which represent initial conditions of the unit safety analyses. Also involved are the movable control components (control and shutdown rods), which are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR. PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the components and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of the NRC Policy Statement.

Reference 6 allows special test exceptions (STEs) to be included as part of the LCO that they affect. It was decided, however, to retain this STE as a separate LCO because it was less cumbersome and provided additional clarity.

LCO

This LCO allows the reactor parameters of MTC and minimum temperature for criticality to be outside their specified limits. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. Operation beyond specified

(continued)

BASES

LCO (continued)

limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

The requirements of LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.1.7, and LCO 3.4.2 may be suspended during the performance of PHYSICS TESTS provided:

5 ~~a. THERMAL POWER is maintained ≤ 5% RTP;~~ 1

a ~~b~~. RCS lowest loop average temperature is ≥ ~~[531]~~ 541 °F; and

b ~~c~~. SDM is ≥ [1.6]% Δk/k. 2

APPLICABILITY

This LCO is applicable in MODE 2 when performing low power PHYSICS TESTS. The applicable PHYSICS TESTS are performed in MODE 2 at HZP. Other PHYSICS TESTS are performed in MODE 1 and are addressed in LCO 3.1.9, "PHYSICS TESTS Exceptions – MODE 1."

ACTIONS

A.1 and A.2

If the SDM requirement is 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. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

B.1

When THERMAL POWER is > 5% RTP, the only acceptable action is to open the reactor trip breakers to prevent operation of the reactor beyond its design limits. Immediately opening the reactor trip breakers will shut down the reactor and prevent operation of the reactor outside of its design limits.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

When the RCS lowest $T_{avg}$ is < 5~~3~~41°F, the appropriate action is to restore $T_{avg}$ to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring $T_{avg}$ to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with temperature below 5~~3~~41°F could violate the assumptions for accidents analyzed in the safety analyses. If the Required Actions cannot be completed within the associated Completion Time, 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 an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, for reaching MODE 3 from full-power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.10.1

The power-range and intermediate-range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." ~~An ANALOG~~ A CHANNEL OPERATIONAL TEST is performed on each power-range and intermediate-range channel within 12 hours prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS. The 12-hour time limit is sufficient to ensure that the instrumentation is OPERABLE shortly before initiating PHYSICS TESTS.

SR 3.1.10.2

Verification that the RCS lowest loop $T_{avg}$ is ≥ 5~~3~~41°F will ensure that the unit is not operating in a Condition that could invalidate the safety analyses. Verification of the RCS temperature at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

5

SR 3.1.10.3

Verification that the power level is ≤ 5% RTP will ensure that the fuel design criteria are not violated during the performance of the PHYSICS TESTS. The Frequency of once per hour is adequate to ensure that the power level does not exceed the limit. Unit operations are conducted slowly during the performance of PHYSICS TESTS, and monitoring the power level at a Frequency of 1 hour is sufficient for ensuring that the THERMAL POWER does not exceed the limit.

SR 3.1.10.~~4~~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

3

f. Design ~~I~~sothermal 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.

4

REFERENCES

INPUT INSERT FROM PAGE B 3.1-69

1. ~~10 CFR 50, Appendix B, Section XI.~~
2. ~~10 CFR 50.59.~~
3. ~~Regulatory Guide 1.68, Revision 2, August, 1978.~~

(continued)

BASES

REFERENCES (continued)

4. ~~ANSI/ANS-19.6.1-1985, December 13, 1985.~~

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

6. ~~WCAP-11618, including Addendum 1, April 1989.~~

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.

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

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 3.1.10

1. Change to reflect WBN specific parameter value.

2. Format change to delete brackets that identify plant specific information/values.

3. Change to reflect WBN specific information.

4. Change to specify WBN specific reference information and format.

5. These comments reflect changes identified by the industry to the NRC at the Proof and Review meeting on the RSTS in Irvine, CA July 13 - July 20, 1992.

## 3.1 REACTIVITY CONTROL SYSTEMS

### 3.1.11 SHUTDOWN MARGIN (SDM) Test Exceptions

LCO 3.1.11 The SDM requirements in MODE 2 may be suspended, provided the reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2 when measuring control rod worth and SDM.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more control rods not fully inserted.<br><br>AND<br><br>Available trip reactivity from OPERABLE control rods less than the highest estimated control rod worth. | A.1 Initiate boration to restore SDM to within limit. | 15 minutes |
| B. All control rods fully inserted.<br><br>AND<br><br>Reactor subcritical by less than the highest estimated control rod worth. | B.1 Initiate boration to restore SDM to within limits. | 15 minutes |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.1.11.1 | ----------NOTE----------<br>Only required for control rods not fully inserted.<br>----------<br><br>Determine the position of each control rod. | 2 hours |
| SR 3.1.11.2 | ----------NOTE----------<br>Only required for control rods not fully inserted.<br>----------<br><br>Trip each control rod from ≥ the 50% withdrawn position, and verify full control rod insertion. | Within 24 hours prior to reducing SDM outside limits |

1

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.11 SHUTDOWN MARGIN (SDM) Test Exceptions 1

BASES

BACKGROUND

The primary purpose of the SDM test exception is to permit relaxation of the SDM requirements during the measurement of control rod worths in MODE 2 during PHYSICS TESTS.

Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

The key objectives of a test program are to (Ref. 3):

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in the design and analysis;

c. Verify the assumptions used to predict unit response;

d. Ensure that installation of equipment at the facility has been accomplished, in accordance with the design; and

e. Verify that operating and emergency procedures are adequate.

To achieve these objectives, testing is performed prior to initial criticality, during startup, low power, power ascension, and at-power operation, and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed (Ref. 4).

PHYSICS TEST procedures are written and approved, in accordance with established formats. The procedures include

(continued)

BASES

BACKGROUND (continued)

all information necessary to permit a detailed execution of the testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures, and test results are approved prior to continued power escalation and long-term power operation.

During the PHYSICS TESTS measurements of control rod worth, it may be necessary to align individual rods and banks in certain configurations and utilize boron concentrations that do not provide sufficient SDM to meet the normal requirements. In this situation, it is necessary to invoke special test exceptions (STEs) to allow the necessary PHYSICS TESTS to be completed.

APPLICABLE SAFETY ANALYSES

Special PHYSICS TESTS may require operating the core under controlled conditions for short periods of time with less than the normally required SDM. As such, these tests are not covered by any safety analysis calculations.

Under the acceptance criteria to allow suspension of certain LCOs for PHYSICS TESTS, fuel damage criteria are not to be exceeded. Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because adequate limits on power distribution and shutdown capability are maintained during PHYSICS TESTS.

Reference 5 defines the requirements for initial testing of the facility, including PHYSICS TESTS. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). PHYSICS TESTS for reload fuel cycles are given in Table 1 of ANSI/ANS-19-6.1-1985. Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, Conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. As long as the linear heat rate remains within its limit, fuel design criteria are preserved.

PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the components and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

This LCO provides an exemption to the SDM requirements under controlled conditions. These conditions require that at least the highest estimated control rod worths be available for trip insertion. It is assumed that this available negative reactivity will be sufficient to shut down the core if required, assuming there is not a concurrent boron dilution or cooldown event. This exemption is allowed even though there are no bounding safety analyses because the tests are performed under close supervision and provide valuable information on control rod worth and core SDM.

APPLICABILITY

This LCO is only applicable in MODE 2, and then only during actual measurement of control rod worths because this is the only time the exception is required.

ACTIONS

A.1

If one or more control rods are not fully inserted and the available trip reactivity from OPERABLE control rods is less than the highest estimated control rod worth, the SDM, assumed for the test conditions, may not be available. Under these conditions, it is necessary to promptly restore the SDM to within limits.

The allowed Completion Time of 15 minutes ensures prompt action and provides an acceptable time for initiating boration to restore SDM, without allowing the core to remain in an unacceptable condition for an extended period of time.

B.1

If all control rods are fully inserted, and the reactor is subcritical by less than the highest estimated control rod worth, the SDM, assumed for the test conditions, may not be available. Under these conditions, it is necessary to promptly restore the SDM to within limits.

The allowed Completion Time of 15 minutes provides an acceptable time for initiating boration to restore SDM, without allowing the core to remain in an unacceptable condition for an extended period of time.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.11.1

In order to establish an acceptable SDM during the measurement of control rod worths, it is necessary to know the position of each control rod. A test Frequency of 2 hours is reasonable, based on normal control rod motion during control rod worth measurements.

SR 3.1.11.1 has been modified by a Note establishing that the position of only those control rods not fully inserted must be determined. It is assumed that the position and worth of fully inserted control rods is known.

SR 3.1.11.2

One of the assumptions made in granting an STE for SDM, is that all control rods not fully inserted will fully insert when tripped. This Surveillance is performed to verify that fact.

The Frequency of 24 hours prior to reducing the plant SDM below the normal requirements is acceptable, based on the assumption that the control rods will remain OPERABLE and trippable for 24 hours and during the performance of the test.

SR 3.1.11.2 has been modified by a Note establishing that this Surveillance is only required for control rods not fully inserted. During the performance of control rod worth measurements, certain control rods remain fully inserted. Since these rods are not relied on to trip, there is no need to demonstrate that they will fully insert when tripped.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
2. 10 CFR 50.59.
3. Regulatory Guide 1.68, Revision 2, August 1978.
4. ANSI/ANS-19.6.1-1985, December 13, 1985.
5. FSAR, Section [14].

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 3.1.11

1. Change to correct error in the STS.

2. Change to specify WBN specific reference information and format.

3. The TS 3.1.11, Shutdown Margin (SDM) Test Exception, is not needed at WBN and should be deleted for the following reasons:

   (1) The test to measure the highest worth control (i.e. the configuration of N-1 control rods inserted with Keff equal to 1.0) is not scheduled to be performed as part of the WBN Unit 1 initial test program.

   (2) The TS LCO 3.1.10 already provides the exceptions needed for other control rod worth measurements.

   (3) Other plants have conducted, with NRC acceptance, initial test programs without establishing the N-1 configuration.

   (4) Chapter 14 of the Final Safety Analysis Report (FSAR) does not, specifically, address the Physics Test that needed the Test Exception. This testing was removed from the initial test program in 1985. The specific test (Formally part of SU-1.4 and SU-3.8A) was the measurement of the highest worth control rod. The testing was generally referred to as the N-1 rod measurement and involved the rod configuration where all rods were inserted except the rod with the highest worth while the reactor remained critical.

## 3.2 POWER DISTRIBUTION LIMITS

### 3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$) ($F_Q$ Methodology)

3

LCO 3.2.1 $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. $F_Q^C(Z)$ not within limit. 6 | A.1 Reduce THERMAL POWER ≥ ~~at least~~ 1% RTP for each 1% $F_Q^C(Z)$ exceeds limit. | 15 minutes |
| | AND | |
| 6 | A.2 Reduce Power Range Neutron Flux – High trip setpoints at ≥ ~~least~~ 1% for each 1% $F_Q^C(Z)$ exceeds limit. | 8 hours |
| | AND | |
| 6 | A.3 Reduce Overpower ΔT trip setpoints at ≥ ~~least~~ 1% for each 1% $F_Q^C(Z)$ exceeds limit. | 72 hours |
| | AND | |
| | A.4 Perform SR 3.2.1.1. | Prior to increasing THERMAL POWER above the limit of Required Action A.1 |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| B. $F_Q^W(Z)$ not within limits. | B.1 Reduce AFD limits ~~at least~~ (6) ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit. | 2 hours |
| C. Required Action and associated Completion Time of Condition A or B not met. | C.1 Be in MODE 2. | 6 hours |

SURVEILLANCE REQUIREMENTS

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

1. During power escalation at the beginning of each cycle, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained. (6)

~~2. These SRs are not required to be performed prior to entry into MODE 1.~~

---

| SURVEILLANCE | FREQUENCY |
|---|---|
| SR 3.2.1.1 Verify $F_Q^C(Z)$ is within limit. (6) | Once After each refueling prior to THERMAL POWER exceeding 75% RTP<br><br>AND<br>12 hours after ~~Upon~~ achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^C(Z)$ was last verified<br><br>AND<br><br>31 EFPD thereafter |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.2.1.2 (4) | ----------NOTE---------- If $F_Q^C(Z)$ (W) is within limits and measurements indicate $\text{maximum over } z \left[ \frac{F_Q^C(Z)}{K(Z)} \right]$ has increased since the previous evaluation of $F_Q^C(Z)$:<br>a. Increase $F_Q^W(Z)$ by a factor of [1.02] and reverify $F_Q^W(Z)$ is within limits; or<br>b. Repeat SR 3.2.1.2 once per 7 EFPD until two successive flux maps indicate $\text{maximum over } z \left[ \frac{F_Q^C(Z)}{K(Z)} \right]$ has not increased.<br>---------- | |
| (4) | Verify $F_Q^C(Z)$ (W) is within limit. | After each refueling prior to THERMAL POWER exceeding 75% RTP<br><br>AND<br><br>(continued) |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| SR 3.2.1.2 (continued) (6) | 12 hours after ~~Upon~~ achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^W(Z)$ was last verified<br><br>AND<br><br>31 EFPD thereafter |

B 3.2 POWER DISTRIBUTION LIMITS

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

(3)

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.

$F_Q(Z)$ 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 non-steady-state conditions are accomplished by operating the core within the

(continued)

BASES

BACKGROUND (continued)

limits of 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 input to the fuel must not exceed 280 cal/gm (Ref. 2); and

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:

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

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

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 ~~[2.32]~~ 2.4, and K(Z) is a function that looks like the one provided in Figure B 3.2.1B-1. 3

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

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

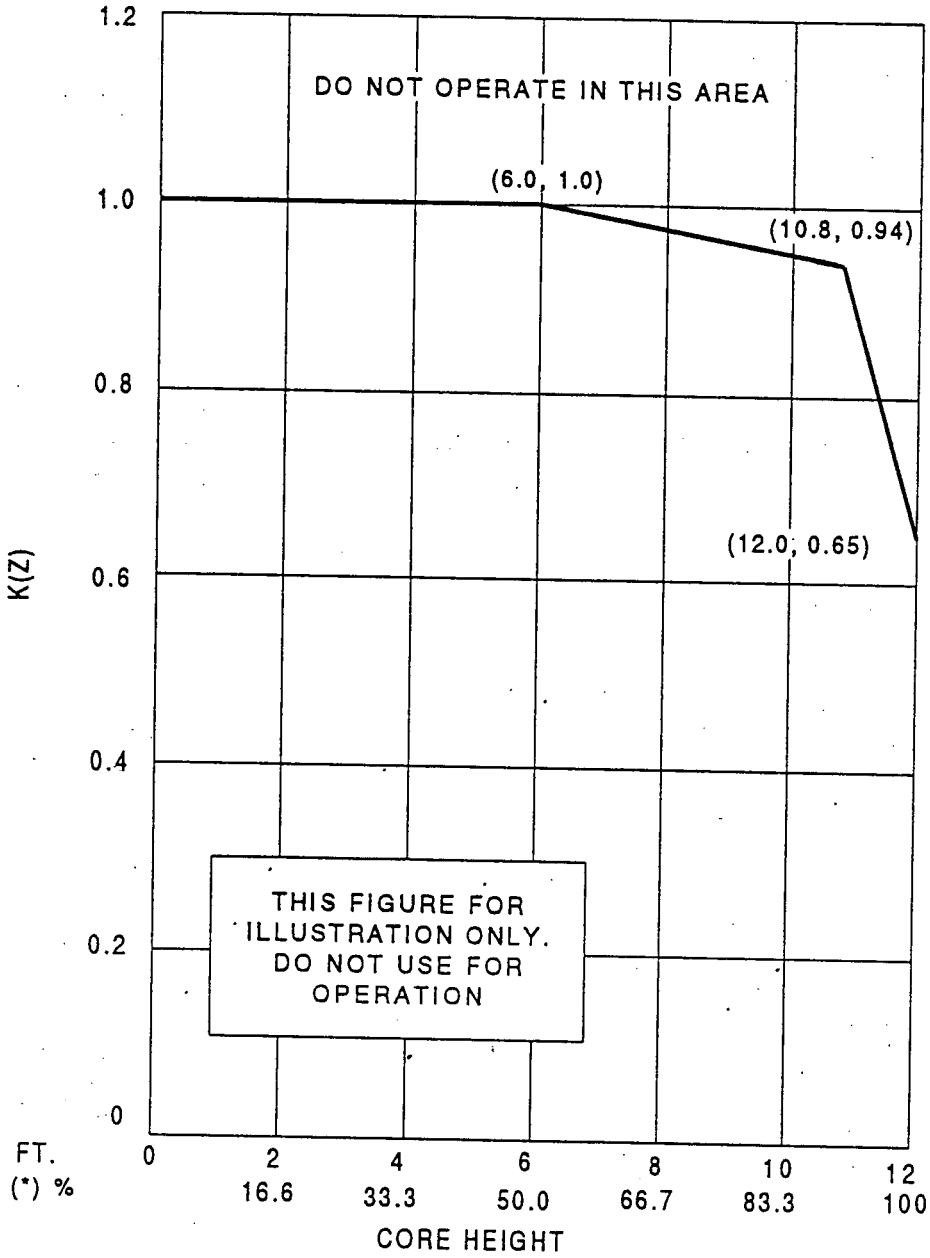
$$F_Q^C(Z) = F_Q^M(Z)\ [1.0815]$$

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

$F_Q^C(Z)$ 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)

BASES (continued)




*For core height of 12 feet

Figure B 3.2.1B-1 (page 1 of 1) (3)
K(Z) - Normalized $F_Q(Z)$ as a Function of Core Height

(continued)

BASES

LCO (continued)

The expression for $F_Q^W(Z)$ is:

$$F_Q^W(Z) = F_Q^C(Z)\ W(Z)$$

where W(Z) is a cycle-dependent function that accounts for power distribution transients encountered during normal operation. W(Z) is included in the COLR.

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large- or small-break LOCA.

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for $F_Q(Z)$ produces unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

APPLICABILITY

The $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

A.1

⑥

Reducing THERMAL POWER by ~~at least~~ ≥ 1% RTP for each 1% by which $F_Q^C(Z)$ exceeds its limit maintains an acceptable absolute power density. $F_Q^C(Z)$ is $F_Q^M(Z)$ multiplied by a factor accounting for manufacturing tolerances and measurement uncertainties. $F_Q^M(Z)$ is the measured value of $F_Q(Z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

(continued)

BASES

ACTIONS (continued)

A.2 (6)

A reduction of the Power Range Neutron Flux–High trip setpoints by ~~at least~~ ≥ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 8 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.3

Reduction in the Overpower ΔT trip setpoints by ≥ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1. (6)

A.4

Verification that $F_Q^C(Z)$ has been restored to within its limit by performing SR 3.2.1.1 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1 ensures that core conditions during operation at higher power levels are consistent with safety analyses assumptions.

B.1

If it is found that the maximum calculated value of $F_Q(Z)$ that can occur during normal maneuvers, $F_Q^W(Z)$, exceeds its specified limits, there exists a potential for $F_Q^C(Z)$ to become excessively high if a normal operational transient occurs. Reducing the AFD by ~~at least~~ ≥ 1% for each 1% by which $F_Q^W(Z)$ exceeds its limit within the allowed Completion Time of 2 hours restricts the axial flux distribution such that even if a transient occurred, core peaking factors are not exceeded. (6)

(continued)

BASES

ACTIONS (continued)

C.1

If Required Actions A.1 through A.4 or B.1 are not met within their associated Completion Times, the plant must be placed in a mode or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 and SR 3.2.1.2 are modified by ~~two~~ Notes. The ~~first~~ Note applies during the first power ascension after a refueling. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that $F_Q^C(Z)$ and $F_Q^W(Z)$ are within their specified limits after a power rise of more than 10% RTP over the THERMAL POWER at which they were last verified to be within specified limits. Because $F_Q^C(Z)$ and $F_Q^W(Z)$ could not have previously been measured in this reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of before exceeding 75% RTP. This ensures that some determination of is made at a lower power level at which adequate margin is available before going to 100% RTP. Also, this frequency condition, together with the frequency condition requiring verification of following a power increase of more than 10%, ensures that ~~is~~ verified as soon as RTP (or any other level 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 $F_Q(Z)$. The Frequency condition is not intended to require ~~$F_Q(Z)$~~ verification 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 at which was last measured.

~~The second Note states that SR 3.2.1.1 and SR 3.2.1.2 are not required to be performed prior to entry into MODE 1~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

because the plant must be in MODE 1 before the Surveillance can be performed. (6)

SR 3.2.1.1

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

The limit with which $F_Q^C(Z)$ is compared varies inversely with power and directly with a function called K(Z) provided in the COLR.

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 technical specifications.

(6)

Performing this Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q^C(Z)$ limit is met when RTP is achieved because Peaking Factors generally decrease as power level is increased.

If THERMAL POWER has been increased by (3) 10% RTP ~~or more~~ since the last determination of $F_Q^C(Z)$, another evaluation of this factor is required ~~upon~~ 12 hours after achieving equilibrium conditions at this higher power level (to ensure that $F_Q^C(Z)$ values are being reduced sufficiently with power increase to stay within the LCO limits).

SR 3.2.1.2

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_Q(Z)$ limits. Because flux maps are taken in steady-state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.2 (continued)

factor increase over steady-state values, calculated as a function of core elevation, Z, is called W(Z). Multiplying the measured total peaking factor, $F_Q^C(Z)$, by W(Z) gives the maximum $F_Q(Z)$ calculated to occur in normal operation, $F_Q^W(Z)$.

The limit with which $F_Q^W(Z)$ is compared varies inversely with power and directly with the function K(Z) provided in the COLR.

The W(Z) curve is provided in the COLR for discrete core elevations. Flux map data are typically taken for 30 to 75 core elevations. $F_Q^W(Z)$ evaluations are not applicable for the following axial core regions, measured in percent of core height:

a. Lower core region, from 0 to 15% inclusive; and

b. Upper core region, from 85 to ~~10~~ 100% inclusive. (4)

The top and bottom 15% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed. If $F_Q^W(Z)$ is evaluated and found to be within its limit, an evaluation of the expression below is required to account for any increase to $F_Q^M(Z)$ that may occur and cause the $F_Q(Z)$ limit to be exceeded before the next required $F_Q(Z)$ evaluation.

If the two most recent $F_Q(Z)$ evaluations show an increase in the expression

$$\text{maximum over } z \left[ \frac{F_Q^C(Z)}{K(Z)} \right],$$

it is required to meet the $F_Q(Z)$ limit with the last $F_Q^W(Z)$ increased by a factor of [1.02], or to evaluate $F_Q(Z)$ more

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.2 (continued)

frequently, each 7 EFPD. These alternative requirements prevent $F_Q(Z)$ from exceeding its limit for any significant period of time without detection.

The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_Q(Z)$ evaluations.

(6) The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with Technical Specifications, to preclude adverse peaking factors between 31-day surveillances.

$F_Q(Z)$ is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification (2) 12 hours after achieving equilibrium conditions to ensure that $F_Q(Z)$ is within its limit at higher power levels.

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

REFERENCES

1. 10 CFR 50.46, 1974.
2. Regulatory Guide 1.77, Rev. 0, May 1974.
3. 10 CFR 50, Appendix A, GDC 26.
4. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.

INSERT B-10 (ATTACHED) (2)

INSERT B-10

REFERENCES


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

2. Regulatory Guide 1.77, Rev. [ ], "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized water Reactors," [date].

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.]

JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.2.1

1. Changes provided by John Molinda, Westinghouse, May 13, 1992 to correct omissions in the STS. Westinghouse has notified NRC of these omissions.

2. Watts Bar prefers to use expanded references.

3. Watts Bar will utilize the $F_Q$ methodology.

4. Change to correct error in the STS. SR 3.2.1.2 applies to $F^W(Z)$ not $F^C(Z)$.

5. Change to include plant specific values.

6. These changes reflect comments identified by the industry to the NRC at the Proof and Review meeting on the RSTS in Irvine, CA July 13 - July 20, 1992.

## 3.2 POWER DISTRIBUTION LIMITS

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

LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. ---------NOTE--------- Required Actions A.2 and A.3 must be completed whenever Condition A is entered. ---------- $F^N_{\Delta H}$ not within limit. | A.1.1 Restore $F^N_{\Delta H}$ to within limit. | 4 hours |
| | OR | |
| | A.1.2.1 Reduce THERMAL POWER to < 50% RTP. | 4 hours |
| | AND | |
| | A.1.2.2 Reduce Power Range Neutron Flux – High trip setpoints to ≤ 55% RTP. | 8 hours |
| | AND | |
| | A.2. Perform SR 3.2.2.1. | 24 hours |
| | AND | |
| | | (continued) |

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. (continued) | A.3 --------NOTE--------- THERMAL POWER does not have to be reduced to comply with this Required Action. ---------- | |
| | Perform SR 3.2.2.1. | Prior to THERMAL POWER exceeding 50% RTP<br><br>AND<br><br>Prior to THERMAL POWER exceeding 75% RTP<br><br>AND<br><br>24 hours after THERMAL POWER reaching ≥ 95% RTP |
| B. Required Action and associated Completion Time not met. | B.1 Be in MODE 2. | 6 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.2.2.1 | ----------NOTE---------- Not required to be performed prior to entry into MODE 1. ---------- Verify $F^N_{\Delta H}$ is within limits. | e (3) Once After each refueling prior to THERMAL POWER exceeding 75% RTP AND 31 EFPD thereafter (3) |

# 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.

1

WRB1

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 [W3] CHF correlation. All DNB-limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements.

(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;

c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm [Ref. 1]; and

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 [W3] CHF correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB.

1

WRB1

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)

BASES

APPLICABLE SAFETY ANALYSES (continued)

this variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation.

The LOCA safety analysis indirectly models $F^N_{\Delta H}$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature [Ref. 3].

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "Axial Flux Difference (AFD)," LCO 3.2.4, "Quadrant Power Tilt Ratio (QPTR)," LCO 3.1.7, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)."

$F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady-state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of the NRC Policy Statement.

LCO.

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for a DNB.

The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses.

A power multiplication factor in this equation includes an additional margin for higher radial peaking from reduced

(continued)

BASES

LCO (continued)

thermal feedback and greater control rod insertion at low power levels. The limiting value of $F^N_{\Delta H}$ is allowed to increase 0.3% for every 1% RTP reduction in THERMAL POWER.

APPLICABILITY

The $F^N_{\Delta H}$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power. Specifically, the design bases events that are sensitive to $F^N_{\Delta H}$ in other modes (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict $F^N_{\Delta H}$ in these modes.

ACTIONS

A.1.1

With $F^N_{\Delta H}$ exceeding its limit, the unit is allowed 4 hours to restore $F^N_{\Delta H}$ to within its limits. This restoration may, for example, involve realigning any misaligned rods or reducing power enough to bring $F^N_{\Delta H}$ within its power-dependent limit. When the $F^N_{\Delta H}$ limit is exceeded, the DNBR limit is not likely violated in steady-state operation, because events that could significantly perturb the $F^N_{\Delta H}$ value (e.g., static control rod misalignment) are considered in the safety analyses. However, the DNBR limit may be violated if a DNB limiting event occurs. Thus, the allowed Completion Time of 4 hours provides an acceptable time to restore $F^N_{\Delta H}$ to within its limits without allowing the plant to remain in an unacceptable condition for an extended period of time.

Condition A is modified by a Note that requires that Required Actions A.2 and A.3 must be completed whenever Condition A is entered. Thus, if power is not reduced because this Required Action is completed within the 4-hour time period, Required Action A.2 nevertheless requires another measurement and calculation of $F^N_{\Delta H}$ within 24 hours in accordance with SR 3.2.2.1.

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

(continued)

BASES

ACTIONS

A.1.1 (continued)

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.2.2.1 are not additive.

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, an incore flux map (SR 3.2.2.1) must be obtained and the measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit at the lower power level. The unit is provided 20 additional hours to perform this task over and above the 4 hours allowed by either Action A.1.1 or Action A.1.2.1. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which

(continued)

BASES

ACTIONS

A.2 (continued)

is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24-hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate $F^N_{\Delta H}$.

A.3

Verification that $F^N_{\Delta H}$ is within its specified limits after an out-of-limit occurrence ensures that the cause that led to the $F^N_{\Delta H}$ exceeding its limit is corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the $F^N_{\Delta H}$ limit is within the LCO limits prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is $\geq$ 95% RTP.

This Required Action is modified by a Note that states that THERMAL POWER does not have to be reduced prior to performing this Action.

B.1

When Required Actions A.1.1 through A.3 cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1 (continued)

measurement uncertainty before making comparisons to the $F^N_{\Delta H}$ limit.

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.

3

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.

This Surveillance is modified by a Note that states that SR 3.2.2.1 is not required to be performed for entry into MODE 1 because the unit must be in MODE 1 to perform surveillances that demonstrate that the LCO is met. 3

2

REFERENCES

1. Regulatory Guide 1.77, Rev [0], May 1974.
2. 10 CFR 50, Appendix A, GDC 26.
3. 10 CFR 50.46.

REFERENCES

1. Regulatory Guide 1.77, Rev [0], "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized water Reactors," [date]. 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," [~~date~~].

JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.2.2

1. Added brackets to heat flux ratio due to results of new analysis possibly changing this value. Changed W3 CHF correlation to WRB1 which is current for Watts Bar.

2. Watts Bar prefers to use expanded references.

3. This change reflects comments made by the industry to the NRC at the Proof and Review meeting on the RSTS in Irvine, CA July 13 - July 20, 1992.

## 3.2 POWER DISTRIBUTION LIMITS

### 3.2.3B AXIAL FLUX DIFFERENCE (AFD) (Relaxed Axial Offset Control) (RAOC) Methodology (4)

LCO 3.2.3 The AFD in %-flux-difference units shall be maintained within the limits specified in the COLR.

----------NOTE----------
The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.
---

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

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. AFD not within limits. | A.1 Reduce THERMAL POWER to < 50% RTP. | 30 minutes |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.2.3.1 | Verify AFD within limits for each OPERABLE excore channel, as follows: | |
| | a. With the AFD monitor alarm inoperable, or | 1 hour |
| | b. With AFD monitor alarm OPERABLE. | 7 days |

7 days
AND
Once within 1 hour and every 1 hour thereafter with the AFD monitor alarm inoperable.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3B AXIAL FLUX DIFFERENCE (AFD) (Relaxed Axial Offset Control) (RAOC) 4
Methodology)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the amount of axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

RAOC is a calculational procedure that defines the allowed operational space of the AFD versus THERMAL POWER. The AFD limits are selected by considering a range of axial xenon distributions that may occur as a result of large variations of the AFD. Subsequently, power peaking factors and power distributions are examined to ensure that the loss-of-coolant accident (LOCA), loss-of-flow accident, and anticipated transient limits are met. Violation of the AFD limits invalidate the conclusions of the accident and transient analyses with regard to fuel cladding integrity.

Although the RAOC defines limits that must be met to satisfy safety analyses, typically an operating scheme, Constant Axial Offset Control (CAOC), is used to control axial power distribution in day-to-day operation (Ref. 1). CAOC requires that the AFD be controlled within a narrow tolerance band around a burnup-dependent target to minimize the variation of axial peaking factors and axial xenon distribution during unit maneuvers.

The CAOC operating space is typically smaller and lies within the RAOC operating space. Control within the CAOC operating space constrains the variation of axial xenon distributions and axial power distributions. RAOC calculations assume a wide range of xenon distributions and then confirm that the resulting power distributions satisfy the requirements of the accident analyses.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The AFD is a measure of the axial power distribution skewing to either the top or bottom half of the core. The AFD is sensitive to many core-related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution, and, to a lesser extent, reactor coolant temperature and boron concentration.

The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

(6) (1) The RAOC methodology (Ref. 2) establishes a xenon distribution library with tentatively wide AFD limits. These limits are labeled tentative because [ ]. One-dimensional axial power distribution calculations are then performed to demonstrate that normal operation power shapes are acceptable for the LOCA and loss-of-flow accident, and for initial conditions of anticipated transients. The tentative limits are adjusted as necessary to meet the safety analysis requirements.

The limits on the AFD ensure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on the AFD also restrict the range of power distributions that are used as initial conditions in the analyses of Condition 2, 3, or 4 events. This ensures that the fuel cladding integrity is maintained for these postulated accidents. The most important Condition 4 event is the LOCA. The most important Condition 3 event is the loss-of-flow accident. The most important Condition 2 events are uncontrolled bank withdrawal and boration or dilution accidents. Condition 2 accidents simulated to begin from within the AFD limits are used to confirm the adequacy of the Overpower $\Delta T$ and Overtemperature $\Delta T$ trip setpoints.

The limits on the AFD satisfy Criterion 2 of the NRC Policy Statement.

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator through the manual operation of the control banks

(continued)

BASES

LCO (continued)

or automatic motion of control banks. The automatic motion 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 or %ΔI.

The AFD limits are provided in the COLR. Figure B 3.2.3B-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 above 50% RTP when the combination of THERMAL POWER and core peaking factors are of primary importance in safety analysis.

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

(continued)

BASES

ACTIONS

A.1 (continued)

30 minutes is reasonable, based on operating experience, to reach 50% RTP without challenging plant systems.

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.

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

Watts Bar 3. FSAR, Chapter ~~[15]~~ 7.7, "Control Systems Not Required for Safety"

JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.2.3

1. Deleted sentence per discussion with Westinghouse on May 14, 1992. This sentence was to have been deleted from the RSTS but inadvertently was not. Westinghouse has notified NRC of this correction.

2. Inserted representative figure for RAOC AFD limits to correct omission in RSTS. Westinghouse has notified NRC of this correction.

3. Corrected FSAR reference for Watts Bar.

4. Watts Bar will utilize the RAOC methodology to determine AFD limits.

5. Change to delete information not applicable to Watts Bar.

6. This change reflects comments made by the industry to the NRC at the Proof and Review meeting on the RSTS in Irvine, CA July 13 - July 20, 1992.

## 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 at least 3% from RTP for each 1% of QPTR > 1.00. | 2 hours<br>AND<br>Once per 12 hours thereafter |
| | AND | | |
| | A.2 | Perform SR 3.2.1.1 and SR 3.2.2.1. | 24 hours<br>AND<br>Once per 7 days thereafter |
| | AND | | |
| | A.3.1 | Reevaluate safety analyses and confirm results remain valid for duration of operation under this condition. | Prior to increasing THERMAL POWER ~~to RTP~~ above the limit of Required Action A.1 (7) |
| | AND | | |
| | | | (continued) |

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. (continued) | A.3.2 --------NOTE--------- Perform Required Action A.3.2 only after Required Action A.3.1 is completed. ---------- | |
| | Calibrate excore detectors to show zero QPTR. | Prior to increasing THERMAL POWER ~~to RTP~~ above the limit of Required Action A.1 (7) |
| | AND | |
| | A.3.3 --------NOTE--------- Perform Required Action A.3.3 only after Required Action A.3.2 is completed. ---------- | |
| | Perform SR 3.2.1.1 and SR 3.2.2.2. | Within 24 hours after reaching RTP<br>OR<br>Within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1. (7) |
| B. Required Action and associated Completion Time not met. | B.1 Reduce THERMAL POWER to ≤ 50% RTP. | 4 hours |

NOTE
With one power range channel inoperable and ~~the~~ THERMAL POWER < 75% RTP, the remaining three power range channels can be used for calculating QPTR.

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.2.4.1 | Verify QPTR is within limit by calculati~~ng~~on. QPTR as follows:<br>a. With QPTR alarm inoperable, or<br>b. With QPTR alarm OPERABLE. | 7 days<br>AND<br>Once within 12 hours and every 12 hours thereafter with the QPTR alarm inoperable<br>12 hours<br>7 days |
| SR 3.2.4.2 | NOTE<br>Only required to be performed if one power range channel is inoperable with THERMAL POWER ≥ 75% RTP.<br><br>Verify QPTR is within limit using ~~with~~ the movable incore detectors ~~by a~~ equivalent.<br>a. Using two sets of four-thimble locations with quarter-core symmetry;<br>OR<br>b. Taking a power distribution flux map. | 7<br>Once within 12 hours<br>AND<br>12 hours thereafter |

# 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 input to the fuel must not exceed 280 cal/gm (Ref. 2); and

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).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

The QPTR limits ensure that $F^N_{\Delta H}$ and $F_Q(Z)$ remain below their limiting values by preventing an undetected change in the gross radial power distribution.

In MODE 1, the $F^N_{\Delta H}$ and $F_Q(Z)$ limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses.

The QPTR satisfies Criterion 2 of the NRC Policy Statement.

LCO

The QPTR limit of 1.02, at which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in $F_Q(Z)$ and ($F^N_{\Delta H}$) is possibly challenged.

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits.

Applicability in MODE 1 $\leq$ 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F^N_{\Delta H}$ and $F_Q(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.

ACTIONS

A.1

With the QPTR exceeding its limit, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.00 is a

(continued)

BASES

ACTIONS

A.1 (continued)

conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition. Because the QPTR alarm is already in its alarmed state, any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. If the QPTR continues to increase, THERMAL POWER has to be reduced accordingly. ~~[For this facility, this 12 hour Completion Time is acceptable because:]~~ A 12-hour Completion Time is sufficient because any additional rate of change in QPTR would be relatively slow, and the cause for the initial QPTR excursion will, in all likelyhood, have been identified and controlled. 1 7

A.2

The peaking factors $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F^N_{\Delta H}$ and $F_Q(Z)$ within the Completion Time of 24 hours ensures that these primary indicators of power distribution are within their respective limits. A Completion Time of 24 hours takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F^N_{\Delta H}$ and $F_Q(Z)$ with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

A.3.1

Although $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists.

(continued)

BASES

ACTIONS

A.3.1 (continued)

It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This reevaluation is required to ensure that, before ~~returning~~ increasing THERMAL POWER ~~to RTP~~ above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

7

A.3.2

If the QPTR has exceeded the 1.02 limit and a reevaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are recalibrated to show a zero QPTR prior to increasing THERMAL POWER. This is done to detect any subsequent significant changes in QPTR.

6 indicated

Required Action A.3.2 is modified by a Note that states that the QPT is not zeroed out until after the reevaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.3.1). This Note is intended to prevent any ambiguity about the required sequence of actions.

A.3.3

2

Once the flux tilt is zeroed out (i.e., Required Action A.3.2 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution at RTP is consistent with the safety analysis assumptions, Action A.2.3 requires verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their specified limits within 24 hours of reaching RTP. As an added precaution, if the core power does not reach RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours of the time when the ascent to power was begun. These Completion Times are intended to allow adequate time to ~~return the unit to its RTP~~ increase THERMAL POWER to above the limit of Required Action A.1 while not permitting the core to remain with unconfirmed power

A.3.3

(continued)

BASES

ACTIONS

A.3.3 (continued)

distributions for extended periods of time. ~~[For this facility, these Completion Times are acceptable because:]~~ 3 7

Action A.3.3 is modified by a Note that states that the peaking factor surveillances may only be done after the excore detectors have been calibrated to show zero tilt (i.e., Required Action A.3.2). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are calibrated to show zero tilt and the core returned to power.

B.1

If Required Actions A.1 through A.3.3 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to < 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

This Surveillance verifies that the QPTR as indicated by the Nuclear Instrumentation System (NIS) excore channels is within its limits. The Frequency of 7 days when the QPTR alarm is OPERABLE is acceptable because of the low probability that this alarm can remain inoperable without detection.

When the QPTR alarm is inoperable, the Frequency is increased to 12 hours. This Frequency is adequate to detect any relatively slow changes in QPTR because for those causes of QPT that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

7 SR 3.2.4.1 is modified by a Note that allows QPTR to be calculated with three power range channels, if THERMAL POWER < 75% RTP and one power range channel is inoperable.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.4.2

This Surveillance is modified by Note a ①, which states that it is required only when one power range channel is inoperable and the THERMAL POWER is ≥ 75% RTP.

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.

For purposes of monitoring the QPTR when one power range channel is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. The incore detector monitoring is performed with a full incore flux map or two sets of four-thimble locations with quarter-core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8. ~~for three- and four-loop cores.~~ ④

⑦ The symmetric thimble flux can be used to generate symmetric thimble "tilt". This can be compared to a reference symmetric thimble tilt, from the most recent full core flux map, to generate an incore QPTR. The incore QPTR can be used to confirm that QPTR is within limits.

With one NIS channel inoperable, the indicated tilt may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has actually occurred, which might cause the QPTR limit to be exceeded, the incore result may be compared against previous flux maps either using the symmetric thimbles as described above or a complete flux map. Nominally, quadrant tilt from the Surveillance should be within 2% of the tilt shown by the most recent flux map data. ⑦

REFERENCES

1. 10 CFR 50.46.
2. Regulatory Guide 1.77, Rev [0], May 1974.
3. 10 CFR 50, Appendix A, GDC 26.

⑤

Replace with INSERT B-6 (Attached)

INSERT B-6

REFERENCES


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

2. Regulatory Guide 1.77, Rev [0], "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," ~~[date]~~. May 1974

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."

JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.2.4

1. This statement should be deleted from the STS and replaced by the wording shown which was provided by Westinghouse to explain the acceptability of the 12-hour completion time.

2. Change to correct Action A.2.3 to A.3.3 typographical error.

3. Deleted justification for completion time because this is provided by preceding text. This should be deleted from STS. Westinghouse has notified NRC of this correction.

4. Deleted "for three - and four - loop cores" because Watts Bar is a four - loop core and no need to mention applicability for 3 loop core.

5. Watts Bar prefers to use expanded references.

6. Editorial change for clarity.

7. This change reflects comments made by the industry to the NRC at the Proof and Review meeting on the RSTS in Irvine, CA July 13 - July 20, 1992.

NOTE: In order to meet submital schedules, the actual typed Watts Bar instrumentation section was based on a second round draft copy of NUREG-1431 (May 1992) which addressed industry comments on the first round draft NUREG submitted in July 1991. The editorial review by the industry and the NRC (June 1992) resulted in additional changes for inclusion into the Proof & Review section (July 1992). These changes were also incorporated into the Watts Bar instrumentation section. However, due to the unreadability of the editorial copy due to the number of changes, a decision was made to use the Proof & Review copy (July 1992) as the platform for the Watts Bar markup. Not all of the editorial comments from the June meeting were actually included into the Proof & Review section, however, so there are minor differences between the Watts Bar typed copy and the Proof & Review copy which are identified by a greek delta symbol.

## 3.3 INSTRUMENTATION

### 3.3.1 Reactor Trip System (RTS) Instrumentation

LCO 3.3.1 The RTS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

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

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. One or more Functions with one or more required channels inoperable. | A.1 | Enter the Condition referenced in Table 3.3.1-1 for the channel(s). | Immediately |
| B. One Manual Reactor Trip channel inoperable. | B.1 | Restore channel to OPERABLE status. | 48 hours |
| | OR | | |
| | B.2.1 | Be in MODE 3. | 54 hours |
| | AND | | |
| | B.2.2 | Open reactor trip breakers (RTBs). | 55 hours |

(continued)

(12) INSERT 1 (see next pg.)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| C. One channel or train inoperable. | C.1 Restore channel or train to OPERABLE status. | 48 hours |
| | OR | |
| | C.2 Open RTBs. | 49 hours |
| D. One Power Range Neutron Flux–High channel inoperable. (A) (13) (C) | Any single (C) ----------NOTE---------- ~~The inoperable~~ channel may be bypassed for up to 4 hours for surveillance testing. ~~of the other channels. and setpoint adjustment~~. ---------- | setpoint adjustment was deleted. |
| | D.1.~~1~~ 2 Reduce THERMAL POWER to ≤ 75% RTP. | ~~4~~ 12 hours (26) |
| | AND | |
| | D.1.~~2~~ 1 Place channel in trip. | 6 hours |
| | OR | |
| | D.2.1 Place channel in trip. | 6 hours |
| | AND | |
| | D.2.2 Perform SR 3.2.4.2. | Once per 12 hours |
| | OR | |
| | D.3 Be in MODE 3. | 12 hours |

(continued)

INSERT 1

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

One train of automatic trip logic may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE.

---

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| E. One channel inoperable. | Any Single (C)<br>----------NOTE----------<br>~~The inoperable~~ channel may be bypassed for up to 4 hours for surveillance testing. ~~of other channels~~. (C)<br>---------- | | |
| | E.1 | Place channel in trip. | 6 hours |
| | OR | | |
| | E.2 | Be in MODE 3. | 12 hours |
| F. THERMAL POWER between P-6 and P-10, one Intermediate Range Neutron Flux channel inoperable. | F.1 | Reduce THERMAL POWER to < P-6. | 2 hours |
| | OR | | |
| | F.2 | Increase THERMAL POWER to > P-10. | 2 hours |
| G. THERMAL POWER between P-6 and P-10, two Intermediate Range Neutron Flux channels inoperable. | G.1 | Suspend operations involving positive reactivity additions. | Immediately |
| | AND | | |
| | G.2 | Reduce THERMAL POWER to < P-6. | 2 hours |
| H. THERMAL POWER below P-6, one or two Intermediate Range Neutron Flux channels inoperable. | H.1 | Restore channel(s) to OPERABLE status. | Prior to increasing THERMAL POWER above P-6 |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| I. One Source Range Neutron Flux channel inoperable. | I.1 | Suspend operations involving positive reactivity additions. | Immediately |
| J. Two Source Range Neutron Flux channels inoperable. (11) | J.1 | Suspend operations involving positive reactivity additions. | Immediately |
| | AND | | |
| | J.~~X~~ 1 | Open RTBs. | Immediately |
| K. One Source Range Neutron Flux channel inoperable. | K.1 | Restore channel to OPERABLE status. | 48 hours |
| | OR | | |
| | K.2.1 | Open RTBs. | 49 hours |
| (11) | AND | | |
| | K.2.2 | Suspend operations involving positive reactivity additions. | 49 hours |
| | AND | | |
| | K.2.3 | Close unborated water source isolation valves. | 49 hours |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| L. Required (G) (29) ~~All~~ Source Range Neutron Flux channels inoperable. (H) | L.1 | Suspend operations involving positive reactivity additions. | Immediately |
| | AND | | |
| | L.2 | Close unborated water source isolation valves. | 1 hour |
| | AND | | |
| | L.3 | Perform SR 3.1.1.1. | 1 hour<br>AND<br>Once per 12 hours thereafter |
| Suspend all operations involving positive reactivity additions that would reduce the SDM to less than the limits specified in LCO 3.1.1 and LCO 3.1.2. | | | |
| M. One channel inoperable. (C) | Any single (C)<br>----------NOTE----------<br>~~The inoperable~~ channel may be bypassed for up to 4 hours for surveillance testing. ~~of other channels~~<br>---------- | | |
| | M.1 | Place channel in trip. | 6 hours |
| | OR | | |
| | M.2 | Reduce THERMAL POWER to < P-7. | 12 hours |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| N. One channel inoperable. | Any single (C)<br>----------NOTE----------<br>~~The inoperable~~ channel may be bypassed for up to 4 hours for surveillance testing. ~~of other channels.~~<br>---------- | |
| (14) Reactor Coolant Flow -- Low (single loop) (C) | N.1 Place channel in trip. | 6 hours |
| | OR | |
| | N.2 Reduce THERMAL POWER to < P-8. | 10 hours |
| O. One Reactor Coolant Pump Breaker Position channel inoperable. | ----------NOTE----------<br>~~The inoperable~~ channel may be bypassed for up to 4 hours for surveillance testing of other channels.<br>---------- | |
| (1) | O.1 Restore channel to OPERABLE status. | 6 hours |
| | OR | |
| | O.2 Reduce THERMAL POWER to < P-8. | 10 hours |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| O ~~P~~. One Turbine Trip channel inoperable. (C) | Any single (C)<br>----------NOTE----------<br>~~The inoperable~~ channel may be bypassed for up to 4 hours for surveillance testing. ~~of the other channels.~~<br>----------<br>O ~~P~~.1 Place channel in trip.<br>OR<br>O ~~P~~.2 Reduce THERMAL POWER to < [P-9]. (F) | Turbine Trip (Low Fluid Oil Pressure) (15)<br>6 hours<br>10 hours |
| P ~~O~~. One ~~Safety Injection~~ SI Input from ESFAS train inoperable. (17) | ----------NOTE----------<br>One train may be bypassed for up to 40 hours for (F) surveillance testing provided the other train is OPERABLE.<br>----------<br>P ~~O~~.1 Restore train to OPERABLE status.<br>OR<br>P ~~O~~.2 Be in MODE 3. | 6 hours<br>12 hours |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| Q ~~R~~. One RTB train inoperable. | ----------NOTES----------<br>1. One train may be bypassed for up to 2 hours for surveillance testing, provided the other train is OPERABLE.<br>2. One RTB may be bypassed for up to 2 hours for maintenance on undervoltage or shunt trip mechanisms, provided the other train is OPERABLE.<br>---------- | |
| | Q ~~R~~.1 Restore train to OPERABLE status. | 1 hour |
| | OR | |
| | ~~R~~.2 Be in MODE 3. Q | 7 hours |
| R ~~S~~. One channel inoperable. | R ~~S~~.1 Verify interlock is in required state for existing unit conditions. | 1 hour |
| | OR | |
| | ~~S~~.2 Be in MODE 3. R | 7 hours |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| S ~~T~~. One channel inoperable. | ~~T~~.1 S Verify interlock is in required state for existing unit conditions. | 1 hour |
| | OR | |
| | ~~T~~.2 S Be in MODE 2. | 7 hours |
| T ~~U~~. One trip mechanism inoperable for one RTB. | ~~U~~.1 T Restore inoperable trip mechanism to OPERABLE status. | 48 hours |
| | OR | |
| | ~~U~~.2.1 T Be in MODE 3. | 54 hours |
| | AND | |
| | ~~U~~.2.2 T Open RTB. | 55 hours |

Insert Conditions U & V
(Attached)
(EAGLE-21 Impact)

©

INSERT
Conditions U & V

| | | | | |
|---|---|---|---|---|
| | | ----------NOTE---------- Any single channel may be bypassed for up to 4 hours for surveillance testing. ---------- | | |
| U. | One Steam Generator Water Level--Low - Low channel inoperable. | U.1.1 | Place channel in trip. | 6 hours |
| | | AND | | |
| | | U.1.2 | For the affected protection set, set the Trip Time Delay ($T_s$) to match the Trip Time Delay ($T_m$). | 4 hours |
| | | OR | | |
| | | U.2 | Be in MODE 3. | 12 hours |
| V. | One Vessel $\Delta T$ channel inoperable. | V.1 | Set the Trip Time Delay threshold power level for ($T_s$) and ($T_m$) to 0% power. | 6 hours |
| | | OR | | |
| | | V.2 | Be in MODE 3. | 12 hours |

SURVEILLANCE REQUIREMENTS

----------NOTE----------
Refer to Table 3.3.1-1 to determine which SRs apply for each RTS Function.
---

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.3.1.1 | Perform CHANNEL CHECK. | 12 hours |
| SR 3.3.1.2 | ----------NOTES----------<br>1. Adjust NIS channel if absolute difference is > 2%.<br><br>2. Not required to be performed until F ~~[12]~~ 120 hours after THERMAL POWER is ≥ 15% RTP.<br>---<br><br>Compare results of calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) channel output. | 24 hours |
| SR 3.3.1.3 | ----------NOTES----------<br>1. Adjust NIS channel if absolute difference is ≥ 3%.<br><br>2. Not required to be performed until ~~[24]~~ 28 96 hours after THERMAL POWER is ≥ ~~[15]~~ 150% RTP. F<br>---<br><br>Compare results of the incore detector measurements to NIS AFD. | 31 effective full power days (EFPD) |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.3.1.4 | ----------NOTE---------- This Surveillance must be performed on the reactor trip bypass breaker prior to placing the bypass breaker in service. ---------- Perform TADOT. | 31 days on a STAGGERED TEST BASIS |
| SR 3.3.1.5 | Perform ACTUATION LOGIC TEST. | 31 days on a STAGGERED TEST BASIS |
| SR 3.3.1.6 | ----------NOTES---------- 1. Adjust excore channels to agree with incore detector measurements. 2. Not required to be performed until ~~[24] hours~~ after THERMAL POWER is ≥ 50% RTP. ---------- ~~Compare~~ Calibrate excore channels to agree with incore detector measurements. | 92 EFPD |
| SR 3.3.1.7 | Perform ACOT. | 92 days |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.3.1.8 | ----------NOTE----------<br>This Surveillance shall include verification that interlocks P-6 and P-10 are in their required state for existing unit conditions.<br>----------<br><br>Perform ~~A~~ C COT. | F<br>~~[92]~~ days |
| SR 3.3.1.9 | H ----------NOTE----------<br>Verification of setpoint is not required.<br>Perform TADOT. | F<br>~~[92]~~ days |
| SR 3.3.1.10 | ----------NOTE----------<br>This Surveillance shall include verification that the time constants are adjusted to the prescribed values.<br>----------<br><br>Perform CHANNEL CALIBRATION. | F<br>~~[18]~~ months |
| SR 3.3.1.11 | ----------NOTE----------<br>Neutron detectors are excluded from CHANNEL CALIBRATION.<br>----------<br><br>Perform CHANNEL CALIBRATION. | F<br>~~[18]~~ months |
| 7 ~~SR 3.3.1.12~~ | ~~----------NOTE----------~~<br>~~This Surveillance shall include verification of Reactor Coolant System resistance temperature detector bypass loop flow rate.~~<br>~~----------~~<br><br>~~Perform CHANNEL CALIBRATION.~~ | ~~[18] months~~ |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | FREQUENCY |
|---|---|
| ~~SR 3.3.1.17 Perform ACOT.~~ | ~~18 months~~ |
| SR 3.3.1.14 Perform TADOT. | 18 months |
| SR 3.3.1.15 ----------NOTE---------- Verification of setpoint is not required. ---------- Perform TADOT. | -----NOTE------ Only required when not performed within previous 31 days ---------- Prior to reactor startup |
| SR 3.3.1.16 ----------NOTE---------- Neutron detectors are excluded from response time testing. ---------- Verify RTS RESPONSE TIME is within limits. | 18 months on a STAGGERED TEST BASIS |

Table 3.3.1-1 (page 1 of 8)
Reactor Trip System Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS | CONDITIONS | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE | TRIP SETPOINT (D) |
|---|---|---|---|---|---|---|
| 1. Manual Reactor Trip | 1,2 | 2 | B | SR 3.3.1.~~13~~ 12 | N/A | N/A |
| | 3(~~b~~)[a], 4(~~b~~)[a], 5(~~b~~)[a] (F) | 2 | C | SR 3.3.1.~~13~~ 12 | N/A | N/A |
| 2. Power Range Neutron Flux | | | | | | |
| a. High | 1,2 | 4 | D | SR 3.3.1.1<br>SR 3.3.1.2<br>SR 3.3.1.7<br>SR 3.3.1.11<br>SR 3.3.1.~~16~~ ~~15~~ 14 | (F) ≤ ~~111.8~~ 111.4% RTP (A) | (F) ≤ ~~109~~% RTP |
| b. Low | 1(~~c~~)[b], 2 (F) | 4 | E | SR 3.3.1.1<br>SR 3.3.1.7<br>SR 3.3.1.11<br>SR 3.3.1.~~16~~ ~~15~~ 14 | (F) ≤ ~~27.2~~ 27.4% RTP | (F) ≤ ~~25~~% RTP |
| c. f(ΔI) | 1,2 | 4 | E | SR 3.3.1.3<br>SR 3.3.1.6 | Refer to Note 1 (Page 3.3-20) | Refer to Note 1 (Page 3.3-20) |
| 3. Power Range Neutron Flux Rate | | | | | | |
| a. High Positive Rate | 1,2 | 4 | E | SR 3.3.1.7<br>SR 3.3.1.11 | (A) ≤ ~~16.8~~ 6.3% RTP with time constant ≥ ~~12~~ 2 sec | (F) ≤ ~~15~~% RTP with time constant ≥ ~~12~~ 2 sec |
| b. High Negative Rate | 1,2 | 4 | E | SR 3.3.1.7<br>SR 3.3.1.11<br>SR 3.3.1.~~16~~ ~~15~~ 14 | (A) 6.3% ≤ ~~16.8~~ RTP with time constant ≥ ~~12~~ 2 sec | (F) ≤ ~~15~~% RTP with time constant ≥ ~~12~~ 2 sec |
| 4. Intermediate Range Neutron Flux | 1(~~c~~)[b], 2(~~d~~)[c] (F) | 2 | F,G | SR 3.3.1.1<br>SR 3.3.1.8<br>SR 3.3.1.11 | ≤ ~~31.3~~ 31.2% RTP (A) | (F) ≤ ~~25~~% RTP |
| | 2(~~e~~)[d] (F) | 2 | H | SR 3.3.1.1<br>SR 3.3.1.8<br>SR 3.3.1.11 | ≤ ~~31.3~~ 31.2% RTP | (F) ≤ ~~25~~% RTP |

(continued)

(D) ~~(a) Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~

(F) (a) ~~(b)~~ With Reactor Trip Breakers (RTBs) closed and Rod Control System capable of rod withdrawal.

(b) ~~(c)~~ Below the P-10 (Power Range Neutron Flux) interlocks.

(c) ~~(d)~~ Above the P-6 (Intermediate Range Neutron Flux) interlocks.



INSERT (e) from next page. (F)

(~~e~~ d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

move to previous page

Table 3.3.1-1 (page 2 of 8)
Reactor Trip System Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS | CONDITIONS | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE | TRIP SETPOINT ~~(a)~~ |
|---|---|---|---|---|---|---|
| 5. Source Range Neutron Flux | 2 ~~(e)~~ d (F) | 2 | I,J | SR 3.3.1.1<br>SR 3.3.1.8<br>SR 3.3.1.11<br>SR 3.3.1.~~16~~ ~~15~~ 14 | ≤ [1.4 E5] cps | ≤ [1.0 E5] cps |
| | 3 ~~(b)~~ a, 4 ~~(b)~~ a, 5 ~~(b)~~ a (F) | 2 | J,K | SR 3.3.1.1<br>SR 3.3.1.8<br>SR 3.3.1.11<br>SR 3.3.1.~~16~~ ~~15~~ 14 | ≤ [1.4 E5] cps | ≤ [1.0 E5] cps |
| | 3 ~~(f)~~ e, 4 ~~(f)~~ e, 5 ~~(f)~~ e (F) | (A) ~~2~~ 1 | L | SR 3.3.1.1<br>SR 3.3.1.8<br>SR 3.3.1.11<br>(H) ~~SR 3.3.1.16~~ ~~15~~ 14 | ≤ [1.4 E5] cps | ≤ [1.0 E5] cps |
| 6. Overtemperature ΔT | 1,2 | [4] | E | SR 3.3.1.1<br>SR 3.3.1.7<br>SR 3.3.1.~~12~~ 10<br>SR 3.3.1.~~16~~ ~~15~~ 14 | Refer to Note 1 (Page 3.3-20) | Refer to Note 1 (Page 3.3-20) |
| 7. Overpower ΔT | 1,2 | [4] | E | SR 3.3.1.1<br>SR 3.3.1.7<br>SR 3.3.1.~~12~~ 10<br>SR 3.3.1.~~16~~ ~~15~~ 14 | Refer to Note 2 (Page 3.3-20) | Refer to Note 2 (Page 3.3-20) |

16

(continued)

D

~~(a) Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~

(F)
~~(b)~~ a With RTBs closed and Rod Control System capable of rod withdrawal.

~~(e)~~ d Below the P-6 (Intermediate Range Neutron Flux) interlocks.

~~(f)~~ e With the RTBs open. In this condition, source range Function does not provide reactor trip but does provide ~~[input to the Boron Dilution Protection System (LCO 3.3.9), and]~~ indication.

D

Table 3.3.1-1 (page 3 of 8)
Reactor Trip System Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS | CONDITIONS | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE | TRIP SETPOINT[a] |
|---|---|---|---|---|---|---|
| 8. Pressurizer Pressure | | | | | | |
| a. Low | 1 (~~g~~ f) F | [4] | M | SR 3.3.1.1<br>SR 3.3.1.7<br>SR 3.3.1.10<br>SR 3.3.1.~~16~~ ~~15~~ 14 | ≥ ~~[1886]~~ 1964.2 psig | ≥ ~~[1900]~~ 1970 psig |
| b. High | 1,2 | [4] | E | SR 3.3.1.1<br>SR 3.3.1.7<br>SR 3.3.1.10<br>SR 3.3.1.~~16~~ ~~15~~ 14 | ≤ ~~[2396]~~ 2390.2 psig | ≤ ~~[2385]~~ 2385 psig F |
| 9. Pressurizer Water Level – High | 1 (~~g~~ f) F | 3 | M | SR 3.3.1.1<br>SR 3.3.1.7<br>SR 3.3.1.10 | ≤ ~~[93.8]~~ 92.7 % | ≤ ~~[92]~~ 92 % F |
| 10. Reactor Coolant Flow – Low | | | | | | |
| a. Single Loop | 1 (~~h~~ g) F | 3 per loop | M | SR 3.3.1.1<br>SR 3.3.1.7<br>SR 3.3.1.10<br>SR 3.3.1.~~16~~ ~~15~~ 14 | ≥ ~~[89.2]~~ 89.6 % (i) | ≥ ~~[90]~~ 90 % F |
| b. Two Loops | 1 (~~j~~ h) F | 3 per loop | M | SR 3.3.1.1<br>SR 3.3.1.7<br>SR 3.3.1.10<br>SR 3.3.1.~~16~~ 14 | ≥ ~~[89.2]~~ 89.6 % (i) | ≥ ~~[90]~~ 90 % F |

A 16 D

(continued)

~~(a) Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~

F (~~g~~ f) Above the P-7 (Low Power Reactor Trips Block) interlock.

(~~h~~ g) Above the P-8 (Power Range Neutron Flux) interlock.

(~~j~~ h) Above the P-7 (Low Power Reactor Trips Block) interlock and below the P-8 (Power Range Neutron Flux) interlock.

(i) Percent of Thermal Design Flow 8

Table 3.3.1-1 (page 4 of 8)
Reactor Trip System Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS | CONDITIONS | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE | TRIP SETPOINT[a] |
|---|---|---|---|---|---|---|
| ~~11. Reactor Coolant Pump (RCP) Breaker Position~~ | | | | | | |
| ~~a. Single Loop~~ | ~~1[h]~~ | ~~1 per RCP~~ | ~~O~~ | ~~SR 3.3.1.14~~ | ~~N/A~~ | ~~N/A~~ |
| ~~b. Two Loops~~ | ~~1[i]~~ | ~~1 per RCP~~ | ~~M~~ | ~~SR 3.3.1.14~~ | ~~N/A~~ | ~~N/A~~ |
| 11 ~~12~~. Undervoltage RCPs | 1[~~g~~] f | A ~~[3]~~ 1 per bus | M | SR 3.3.1.9 (J) 3<br>SR 3.3.1.10<br>SR 3.3.1.~~16~~ ~~15~~ 14 | ≥ [4760] volts | ≥ [4830] volts |
| 12 ~~13~~. Underfrequency RCPs | 1[~~g~~] f | A ~~[3]~~ 1 per bus | M | SR 3.3.1.9 (J)<br>SR 3.3.1.10<br>SR 3.3.1.~~16~~ ~~15~~ 14 | ≥ [57.1] Hz | ≥ [57.5] Hz |
| ~~14. Steam Generator (SG) Water Level – Low Low~~ | ~~1,2~~ | ~~[4 per SG]~~ | ~~E~~ | ~~SR 3.3.1.1~~<br>~~SR 3.3.1.7~~<br>~~SR 3.3.1.10~~<br>~~SR 3.3.1.16~~ | ~~≥ [30.4]%~~ | ~~≥ [32.3]%~~ |
| ~~15. SG Water Level – Low~~ | ~~1,2~~ | ~~2 per SG~~ | ~~E~~ | ~~SR 3.3.1.1~~<br>~~SR 3.3.1.7~~<br>~~SR 3.3.1.10~~<br>~~SR 3.3.1.16~~ | ~~≥ [30.4]%~~ | ~~≥ [32.3]%~~ |
| ~~Coincident with Steam Flow/ Feedwater Flow Mismatch~~ | ~~1,2~~ | ~~2 per SG~~ | ~~E~~ | ~~SR 3.3.1.1~~<br>~~SR 3.3.1.7~~<br>~~SR 3.3.1.10~~<br>~~SR 3.3.1.16~~ | ~~≤ [42.5]% full steam flow at RTP~~ | ~~≤ [40]% full steam flow at RTP~~ |

Replace with Insert A

(continued)

~~(a) Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~

(~~g~~) f Above the P-7 (Low Power Reactor Trips Block) interlock.

(~~h~~) g Above the P-8 (Power Range Neutron Flux) interlock.

(~~i~~) h Above the P-7 (Low Power Reactor Trips Block) interlock and below the P-8 (Power Range Neutron Flux) interlock.

(j) Setpoint Verification not required

t A

C

| | FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS | CONDITIONS | SURVEILLANCE REQUIREMENTS | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|---|---|
| 13. | SG Water Level--Low-Low | 1, 2 | 3/SG | U | SR 3.3.1.1<br>SR 3.3.1.7<br>SR 3.3.1.10<br>SR 3.3.1.~~15~~ 14 | ≥ 17% of narrow range instrument span | ≥ 15.4% of narrow range instrument span |
| | Coincident with:<br><br>a) Vessel ΔT Equivalent to power ≤ 50% RTP<br><br>With a time delay (Ts) if one Steam Generator is affected<br><br>or<br><br>A time delay (Tm) if two or more Steam Generators are affected | 1, 2 | 1/loop | V | SR 3.3.1.1<br>SR 3.3.1.7<br>SR 3.3.1.10<br>SR 3.3.1.~~25~~ 14 | Vessel ΔT variable input ≤ 50% RTP<br><br>≤ Ts (Refer to Note 5, page 3.3-19)<br><br>≤ Tm (Refer to Note 5, page 3.3-19) | Vessel ΔT variable input ≤ 51.8% RTP<br><br>≤ [ ] Ts (Refer to Note 5, page 3.3-19)<br><br>≤ [ ] Tm (Refer to Note 5, page 3.3-19) |
| | OR<br><br>b) Vessel ΔT equivalent to power > 50% RTP | 1, 2 | 1/loop | V | SR 3.3.1.1<br>SR 3.3.1.7<br>SR 3.3.1.10<br>SR 3.3.1.~~25~~ 14 | Vessel ΔT variable input ≤ 50% RTP | Vessel ΔT variable input ≤ 51.8% RTP |

16

Table 3.3.1-1 (page 5 of 8)
Reactor Trip System Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS | CONDITIONS | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE | TRIP SETPOINT ~~(a)~~ |
|---|---|---|---|---|---|---|
| 14 ~~13~~. Turbine Trip | | | | | | |
| a. Low Fluid Oil Pressure | 1 ~~(j)~~ k | 3 | ~~P~~ O | SR 3.3.1.10<br>SR 3.3.1.~~15~~ ~~14~~ 13 | ≥ ~~[750]~~ 43 psig | ≥ ~~[800]~~ 45 psig |
| b. Turbine Stop Valve Closure | 1 ~~(j)~~ k | 4 | ~~P~~ O | SR 3.3.1.10<br>SR 3.3.1.~~15~~ ~~14~~ 13 | ≥ ~~[1]~~ b % open | ≥ ~~[1]~~ b % open |
| 15 ~~14~~. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS) | 1,2 | 2 trains | ~~P~~ P | SR 3.3.1.~~14~~ ~~13~~ 12 | N/A | N/A |
| 16 ~~15~~. Reactor Trip System Interlocks | | | | | | |
| a. Intermediate Range Neutron Flux, P-6 | 2 ~~(e)~~ d | 2 | ~~S~~ R | SR 3.3.1.11<br>~~SR 3.3.1.~~~~13~~ ~~12~~ | ≥ [6E-11] amp | ≥ [1E-10] amp |
| b. Low Power Reactor Trips Block, P-7 | 1 | 1 per train | ~~T~~ S | SR 3.3.1.11<br>~~SR 3.3.1.~~~~13~~ ~~12~~ | N/A | N/A |
| c. Power Range Neutron Flux, P-8 | 1 | 4 | ~~T~~ S | SR 3.3.1.11<br>~~SR 3.3.1.~~~~13~~ ~~12~~ | ≤ ~~[50.2]~~ 50.4 % RTP | ≤ ~~[48]~~ 48 % RTP |
| d. Power Range Neutron Flux, P-9 | 1 | 4 | ~~T~~ S | SR 3.3.1.11<br>~~SR 3.3.1.~~~~13~~ ~~12~~ | ≤ ~~[52.2]~~ 52.4 % RTP | ≤ ~~[50]~~ b % RTP |
| e. Power Range Neutron Flux, P-10 | 1,2 | 4 | ~~S~~ R | SR 3.3.1.11<br>~~SR 3.3.1.~~~~13~~ ~~12~~ | ≥ ~~[12.2]~~ 7.6% RTP and ≤ ~~[75]~~ 12.4 % RTP | ≥ ~~[10]~~ 9 % RTP |
| f. Turbine Impulse Pressure, P-13 | 1 | 2 | ~~T~~ S | ~~[SR 3.3.1.1]~~ 9<br>SR 3.3.1.10<br>~~SR 3.3.1.~~~~13~~ ~~12~~ | ≤ ~~[12.2]~~ 12.4 % turbine power | ≤ ~~[10]~~ 9 % turbine power |

(continued)

~~(a) Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~

~~(e)~~ d Below the P-6 (Intermediate Range Neutron Flux) interlocks.

~~(j)~~ k Above the P-9 (Power Range Neutron Flux) interlock.

Table 3.3.1-1 (page 6 of 8)
Reactor Trip System Instrumentation

| | FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS | CONDITIONS | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE | TRIP SETPOINT ~~(a)~~ |
|---|---|---|---|---|---|---|---|
| 17 ~~18~~. | Reactor Trip Breakers ~~(k)~~ ℓ | 1,2 | 2 trains | ~~B~~ a | SR 3.3.1.4 | N/A | N/A |
| | | 3 ~~(b)~~ a, 4 ~~(b)~~ a, 5 ~~(b)~~ a | 2 trains | C | SR 3.3.1.4 | N/A | N/A |
| 18 ~~19~~. | Reactor Trip Breaker Undervoltage and Shunt Trip (m) Mechanisms | 1,2 | 1 per RTB each | ~~B~~ T | SR 3.3.1.4 | N/A | N/A |
| | | 3 ~~(b)~~ a, 4 ~~(b)~~ a, 5 ~~(b)~~ a | 1 per RTB each | C | SR 3.3.1.4 | N/A | N/A |
| 19 ~~20~~. | Automatic Trip Logic | 1,2 | 2 trains | ~~B~~ ~~C~~ P | SR 3.3.1.5 | N/A | N/A |
| | | 3 ~~(b)~~ a, 4 ~~(b)~~ a, 5 ~~(b)~~ a | 2 trains | C | SR 3.3.1.5 | N/A | N/A |

~~(a) Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~

~~(b)~~ a With RTBs closed and Rod Control System capable of rod withdrawal.

~~(k)~~ ℓ Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB.

(m) Shunt Trip mechanism does not apply for Reactor Trip Bypass Breakers

A

Table 3.3.1-1 (page 7 of 8)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

lower case

The Overtemperature ΔT Function ALLOWABLE VALUE shall not exceed the following trip setpoint by more than [3.8]% of ΔT span.

2.0

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)} \left(\frac{1}{1+\tau_3 s}\right) \le \Delta T_o \left\{ K_1 - K_2 \frac{(1+\tau_4 s)}{(1+\tau_5 s)} \left[ T \left(\frac{1}{1+\tau_6 s}\right) - T' \right] + K_3 (P - P') - f_1(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT, °F.
$\Delta T_0$ is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, $sec^{-1}$.
T is the measured RCS average temperature, °F.
T' is the ~~nominal~~ indicated $T_{avg}$ at RTP, ≤ [588] °F.
P is the measured pressurizer pressure, psig.
P' is the nominal RCS operating pressure, ≤ [2235] psig.

$K_1$ ≤ [1.09] 1.0952 $K_2$ ≥ [0.0138] 3 /°F $K_3$ = [0.00071] 47 psig.
$\tau_1$ ≥ [8] 33 sec. $\tau_2$ ≤ [3] sec. 4 $\tau_3$ ≤ [2] sec. 1
$\tau_4$ ≥ [33] 12 sec. $\tau_5$ ≤ [4] 3 sec. $\tau_6$ ≤ [2] sec.

$f_1(\Delta I)$ = 1.26{35 + ($q_t$ - $q_b$)} when $q_t - q_b$ ≤ -[35] % RTP
0% of RTP when -[35]% RTP < $q_t - q_b$ ≤ [7] % RTP
-1.05{($q_t$ - $q_b$) - 7} when $q_t - q_b$ > [7] % RTP

Where $q_t$ and $q_b$ are percent RTP in the upper and lower halves of the core respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

(A)

Table 3.3.1-1 (page 8 of 8)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

lower case

The Overpower ΔT Function ALLOWABLE VALUE shall not exceed the following trip setpoint by more than [3]% of ΔT span.

1.8

$$\Delta T\left[\frac{(1+\tau_4 s)(1+\tau_5 s)}{(1+\tau_x s)}\right]\left[\frac{1}{1+\tau_3 s}\right] \le \Delta T_0\left\{K_4 - K_5\left[\frac{\tau_x s}{1+\tau_x s}\right]\left[\frac{1}{1+\tau_6 s}\right]T - K_6\left[T\left(\frac{1}{1+\tau_6 s}\right) - T''\right] - f_2(\Delta I)\right\}$$

Where: ΔT is measured RCS ΔT, °F.
$\Delta T_0$ is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, $sec^{-1}$.
T is the measured RCS average temperature, °F.
T" is the ~~nominal~~ indicated $T_{avg}$ at RTP, ≤ [588]°F.

$K_4 \le [1.09]$ $K_5 \ge [0.02]/°F$ for increasing $T_{avg}$
$[0]/°F$ for decreasing $T_{avg}$
$K_6 \ge [0.00128]/°F$ when $T > T''$
$[0]/°F$ when $T \le T''$

$\tau_1 \ge [8]$ sec. $\tau_2 \le [3]$ sec.
$\tau_6 \le [2]$ sec. $\tau_7 \ge [10]$ sec.
$\tau_3 \le [2]$ sec.

$f_2(\Delta I) = 0\%$ RTP for all ΔI.

$\tau_4 \ge 12$ sec.
$\tau_5 \le 3$ sec.

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Note 3 Steam Generator Water Level Low-Low Trip Time Delay

$$T_s = \{ A_1 (P)^3 + A_2 (P)^2 + A_3 (P) + A_4 \} \{ 0.99 \}$$

$$T_m = \{ B_1 (P)^3 + B_2 (P)^2 + B_3 (P) + B_4 \} \{ 0.99 \}$$

Where:

$P$ = Vessel $\Delta T$ Equivalent to power (% RTP), $P \leq 50\%$ RTP.

$T_s$ = Time Delay for Steam Generator Water Level --Low-Low Reactor Trip, one Steam Generator affected.

$T_m$ = Time Delay for Steam Generator Water Level --Low-Low Reactor Trip, two or more Steam Generators affected.

| | |
|---|---|
| $A_1$ = [ TBD ] | $B_1$ = [ TBD ] |
| $A_2$ = [ TBD ] | $B_2$ = [ TBD ] |
| $A_3$ = [ TBD ] | $B_3$ = [ TBD ] |
| $A_4$ = [ TBD ] | $B_4$ = [ TBD ] |

TBD = To Be Determined

Global ACOT To CDT



# B 3.3 INSTRUMENTATION

## B 3.3.1 Reactor Trip System (RTS) Instrumentation

BASES

BACKGROUND

The RTS initiates a unit shutdown, based upon the values of selected unit parameters, to protect against violating the core fuel design limits and Reactor Coolant System (RCS) pressure boundary during anticipated operational occurrences (AOOs) and to assist the Engineered Safety Features (ESF) Systems in mitigating accidents.

The protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RTS, as well as LCOs on other reactor system parameters and equipment performance.

The LSSS, defined in this specification as the Trip Setpoints, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs).

During AOOs, which are those events expected to occur one or more times during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit value to prevent departure from nuclear boiling (DNB);

2. Fuel centerline melt shall not occur; and

3. The RCS pressure Safety Limit (SL) of 2750 psia shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits ~~(SLs)~~," also maintains the above values and assures that offsite dose will be within the 10 CFR ~~5~~0 and 10 CFR 100 criteria during AOOs. 2 27

Accidents are events which are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 limits. Different accident categories ~~are~~ allowed a

F

(continued)

BASES

BACKGROUND (continued)

different fraction of these limits, based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The RTS instrumentation is segmented into four distinct but interconnected modules as illustrated in Figure ~~1~~ 7.1-1, FSAR, Chapter ~~[7]~~ 7 (Ref. ~~1~~ 2), and as identified below:

1. Field transmitters or process or contact actuation sensors: provide a measurable electronic signal based upon the physical characteristics of the parameter being measured;

2. Signal Process Control and Protection System, including ~~Analog~~ Protection System, Nuclear Instrumentation System (NIS), and field contacts, ~~and protection channel sets~~: provides analog to digital conversion, signal conditioning, ~~bistable~~ setpoint comparison, process algorithm actuation (Digital Protection System), compatible electrical signal output to protection system devices, and control board/control room/miscellaneous indications;

3. Solid State Protection System (SSPS), including input, logic, and output bays: initiates proper unit shutdown and/or ESF actuation in accordance with the defined logic which is based on the ~~bistable outputs~~ bistable, setpoint comparators, or contact outputs from the signal process control and protection system; and ~~or CPU trip~~

4. Reactor trip switchgear, including reactor trip breakers (RTBs) and bypass breakers: provides the means to interrupt power to the control rod drive mechanisms (CRDM) and allow the rod cluster control assemblies (RCCAs), or "rods," to fall into the core and shut down the reactor. The bypass breakers allow testing of the RTBs at power.

Field Transmitters or Process Sensors

In order to meet the design demands for redundancy and reliability, more than one, and often as many as ~~four~~ five field transmitters or sensors are used to measure unit parameters. To account for the calibration tolerances and instrument drift, which are assumed to occur between and during calibrations, statistical allowances are provided in the

(continued)

BASES

BACKGROUND

Field Transmitters or Process Sensors (continued)

Trip Setpoint and Allowable Values. The OPERABILITY of each transmitter or sensor can be evaluated when its "as found" calibration data are compared against its documented acceptance criteria.

(18) (Digital Protection System) analog to digital conversion

Signal Process Control and Protection System

Generally, three or four channels of process control equipment are used for the signal processing of unit parameters measured by the field instruments. The process control equipment provides signal conditioning, comparable output signals for instruments located on the main control board, and comparison of measured input signals with setpoints established by safety analyses. These setpoints are defined in References 1, 2, and 3. If the measured value of a unit parameter exceeds the predetermined setpoint, an output from a bistable is forwarded to the SSPS for decision evaluation. Channel separation is maintained up to and through the input bays. However, not all unit parameters require four channels of sensor measurement and signal processing. Some unit parameters provide input only to the SSPS while others provide input to the SSPS, main control board, unit computer, and one or more control systems.

(C)

.g setpoint Comparator or contact (18)

Generally, if a parameter is used only for input to the protection circuits, three channels with a two out of three logic are sufficient to provide the required reliability and redundancy. If one channel fails in the nonconservative direction, the Function is still OPERABLE with a two out of two logic. If one channel fails in the conservative direction, a trip will not occur because of the single failure, and the Function is still OPERABLE with a one out of two logic.

Generally, if a parameter is used for input to the SSPS and a control Function, four channels with a two out of four logic are sufficient to provide the required reliability and redundancy. The circuit must be able to withstand both an input failure to the control system, which may then require the protection Function actuation, and a single failure in the other channels providing the protection Function actuation. Again, a single failure will neither cause nor

(continued)

BASES

BACKGROUND

Signal Process Control and Protection System (continued)

prevent the protection Function actuation. These requirements are described in IEEE-279-1971 (Ref. 4). The actual number of channels required for each unit parameter is specified in Reference 1.

Two logic ~~channels~~ trains are required to ensure no single random failure of a logic ~~channel~~ train will disable the RTS. The logic ~~channels~~ trains are designed such that testing required while the reactor is at power may be accomplished without causing a trip. ~~Provisions to allow removing logic channels from service during maintenance are unnecessary because of the logic systems designed reliability.~~ (A) (A)

(18) , Setpoint Comparators, or Contact

Trip Setpoints and Allowable Values (C)

The Trip Setpoints are the nominal or trip output values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION accuracy ~~(i.e., ± rack calibration + comparator setting accuracy)~~. (C) (21)

~~or CPU~~ trip outputs

~~and CPU~~ trip outputs

The Trip Setpoints used in the bistables are based on the analytical limits stated in Reference 1. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors, for those RTS channels which must function in harsh environments as defined by 10 CFR 50.49 (Ref. 5), the Trip Setpoints and Allowable Values specified in Table 3.3.1-1 are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the Trip Setpoints, including their explicit uncertainties, is provided in the "~~RTS/ESFAS Setpoint Methodology Study~~" (Ref. 6). The actual nominal Trip Setpoint entered into the bistable/comparator is more conservative than that specified by the Allowable Value, to account for changes in random measurement errors detectable by ~~an ACOT~~. a COT One example of such a change in measurement error is drift during the Surveillance interval. If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE. (C)

(20) Westinghouse Setpoint Methodology for Protection Systems, Watts Bar 1 and 2

(continued)

BASES

BACKGROUND

Trip Setpoints and Allowable Values (continued)

Setpoints in accordance with the Allowable Value will ensure that SLs are not violated during AOOs (and that the consequences of DBAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed). Note that in the accompanying LCO 3.3.1, the Trip Setpoints of Table 3.3.1-1 are the LSSS.

Each channel of the process control equipment can be tested on line to verify that the signal or setpoint accuracy is within the specified allowance requirements of Reference 2. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated. SRs for the channels are specified in the SRs section.

The Trip Setpoints and Allowable Values listed in Table 3.3.1-1 are based upon the methodology described in Reference 5, which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each Trip Setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

Solid State Protection System

The SSPS equipment is used for the decision logic processing of bistable outputs from the signal processing equipment. To meet the redundancy requirements, two trains of SSPS, each performing the same Functions, are provided. In the event that one train is taken out of service for maintenance or test purposes, the second train will provide reactor trip and/or ESF actuation for the unit. In the event that both trains are taken out of service or placed in test, a reactor trip will result. Each train is packaged in its own cabinet for physical and electrical separation to satisfy separation and independence requirements. The system has been designed to trip in the event of a loss of power, directing the unit to a safe shutdown condition.

(continued)

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BACKGROUND

Setpoint comparator

C

~~CPU~~ Trip outputs and

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F and

Solid State Protection System (continued)

The SSPS provides the decision logic for actuating a reactor trip or ESF actuation, provides the electrical output signal that will initiate the required trip or actuation, and provides the status, interlock, and annunciator output signals to the main control room of the unit.

contact outputs

The bistable outputs from the signal processing equipment are sensed by the SSPS equipment and combined into logic matrices that represent combinations indicative of various unit upset and accident transients. If a required logic matrix combination is completed, the system will initiate a reactor trip or send actuation signals via master and slave relays to those components whose aggregate Function best serves to alleviate the condition and restore the unit to a safe condition. Examples are given in the Applicable Safety Analyses, LCO, and Applicability sections of these Bases.

F

Reactor Trip Switchgear

The RTBs are in the electrical power supply line from the control rod drive motor generator set power supply to the CRDMs. Opening of the RTBs interrupts power to the CRDMs, which allows the shutdown rods and control rods to fall into the core by gravity. Each RTB is equipped with a bypass breaker to allow testing of the RTB while the unit is at power. During normal operation the output from the SSPS is a voltage signal that energizes the undervoltage coils in the RTBs, and bypass breakers if in use. When the required logic matrix combination is completed, the SSPS output voltage signal is removed, the undervoltage coils are de-energized, the breaker trip lever is actuated by the de-energized undervoltage coil, and the RTBs and bypass breakers are tripped open. This allows the shutdown rods and control rods to fall into the core. In addition to the de-energization of the undervoltage coils, each breaker is also equipped with a shunt trip device that is energized to trip the breaker open upon receipt of a reactor trip signal from the SSPS. Either the undervoltage coil or the shunt trip mechanism is sufficient by itself, thus providing a diverse trip mechanism.

The decision logic matrix Functions are described in the functional diagrams included in Reference 2. In addition to

(continued)

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BACKGROUND

Reactor Trip Switchgear (continued)

the reactor trip or ESF, these diagrams also describe the various "permissive interlocks" that are associated with unit conditions. Each train has a built in testing device that can automatically test the decision logic matrix Functions and the actuation devices while the unit is at power. When any one train is taken out of service for testing, the other train is capable of providing unit monitoring and protection until the testing has been completed. The testing device is semiautomatic to minimize testing time.

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

The RTS functions to maintain the SLs during all AOOs and mitigates the consequences of DBAs in all MODES in which the RTBs are closed.

Each of the analyzed accidents and transients can be detected by one or more RTS Functions. The accident analysis described in Reference 3 takes credit for most RTS trip Functions. RTS trip Functions not specifically credited in the accident analysis were qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the unit. These RTS trip Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. They may also serve as backups to RTS trip Functions that were credited in the accident analysis.

The LCO requires all instrumentation performing an RTS Function, listed in Table 3.3.1-1 in the accompanying LCO, to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

The LCO generally requires OPERABILITY of four or three channels in each instrumentation Function and two channels in each logic Function. Four OPERABLE instrumentation channels in a two out of four configuration are required when one RTS channel is also used as a control system input. This configuration accounts for the possibility of the shared channel failing in such a manner that it creates a transient that requires RTS action. In this case, the RTS will still provide protection, even with random failure of

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

one of the other three protection channels. Three operable instrumentation channels in a two out of three configuration are generally required when there is no potential for control system and protection system interaction that could simultaneously create a need for RTS trip and disable one RTS channel. The two out of three and two out of four configurations allow one channel to be tripped during maintenance or testing without causing a reactor trip. Specific exceptions to the above general philosophy exist and are discussed below.

Reactor Trip System Functions

The safety analyses and OPERABILITY requirements applicable to each RTS Function are discussed below:

1. Manual Reactor Trip

The Manual Reactor Trip ensures that the control room operator can initiate a reactor trip at any time by using either of two reactor trip switches in the control room. A Manual Reactor Trip accomplishes the same results as any one of the automatic trip Functions. It is used by the reactor operator to shut down the reactor whenever any parameter is rapidly trending toward its Trip Setpoint.

The LCO requires two Manual Reactor Trip channels, one per train, to be OPERABLE. Two independent channels are required to be OPERABLE so that no single random failure will disable the manual reactor trip Function.

In MODE 1 or 2, manual initiation of a reactor trip must be OPERABLE. These are the MODES in which the shutdown rods and/or control rods are partially or fully withdrawn from the core. In MODE 3, 4, or 5, the manual initiation Function must also be OPERABLE if the shutdown rods or control rods are withdrawn or the Control Rod Drive (CRD) System is capable of withdrawing the shutdown rods or the control rods. In this condition, inadvertent control rod withdrawal is possible. In MODE 3, 4, or 5, manual initiation of a reactor trip does not have to be OPERABLE if the CRD System is not capable of withdrawing the shutdown rods or control rods. If the rods cannot be withdrawn from

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCOs, and APPLICABILITY

1. Manual Reactor Trip (continued)

the core, there is no need to be able to trip the reactor because all of the rods are inserted. In MODE 6, neither the shutdown rods nor the control rods are permitted to be withdrawn and the CRD Mechanisms are disconnected from the control rods and shutdown rods. Therefore, the manual initiation Function is not required.

2. Power Range Neutron Flux

The NIS power range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS power range detectors provide input to the Rod Control System and the Steam Generator (SG) Water Level Control System. Therefore, the actuation logic must be able to withstand an input failure to the control system (which may then require the protection Function actuation) and a single failure in the other channels providing the protection Function actuation. Note that this Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

a. Power Range Neutron Flux–High

The Power Range Neutron Flux–High trip Function ensures that protection is provided, from all power levels, against a positive reactivity excursion leading to DNB during power operations. These can be caused by rod withdrawal or reductions in RCS temperature.

The LCO requires all four of the Power Range Neutron Flux–High channels to be OPERABLE.

In MODE 1 or 2, when a positive reactivity excursion could occur, the Power Range Neutron Flux–High trip must be OPERABLE. This Function will terminate the reactivity excursion and shut down the reactor prior to reaching a power level that could damage the fuel. In MODE 3, 4, 5, or 6, the NIS power range detectors cannot detect

(continued)

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APPLICABLE SAFETY ANALYSES LCOs, and APPLICABILITY

a. Power Range Neutron Flux—High (continued)

neutron levels in this range. (22) In these MODES, the Power Range Neutron Flux—High does not have to be OPERABLE because the reactor is shut down and reactivity excursions into the power range are extremely unlikely. Other RTS Functions and administrative controls provide protection against reactivity additions when in MODE 3, 4, 5, or 6.

b. Power Range Neutron Flux—Low

The LCO requirement for the Power Range Neutron Flux—Low trip Function ensures that protection is provided against a positive reactivity excursion from low power or subcritical conditions.

The LCO requires all four of the Power Range Neutron Flux—Low channels to be OPERABLE.

In MODE 1, below the Power Range Neutron Flux (P-10 setpoint), and in MODE 2, the Power Range Neutron Flux—Low trip must be OPERABLE. This Function may be manually blocked by the operator when two out of four power range channels are greater than approximately 10% of RTP (P-10 setpoint). This Function is automatically unblocked when three out of four power range channels are below the P-10 setpoint. Above the P-10 setpoint, positive reactivity additions are mitigated by the Power Range Neutron Flux—High trip Function.

In MODE 3, 4, 5, or 6, the Power Range Neutron Flux—Low trip Function does not have to be OPERABLE because the reactor is shut down and the NIS power range detectors cannot detect neutron levels in this range. Other RTS trip Functions and administrative controls provide protection against positive reactivity additions or power excursions in MODE 3, 4, 5, or 6.

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCOs, and APPLICABILITY (continued)

c. Power Range Neutron Flux – f(ΔI)

The f(ΔI) Function is used in the calculation of the Overtemperature ΔT trip. It is a Function of the indicated difference between the upper and lower NIS power range detectors. This Function measures the axial power distribution. The Overtemperature ΔT Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced in accordance with Note 1 of Table 3.3.1-1.

The LCO requires all four channels of f(ΔI) to be OPERABLE.

This Function acts only as an input to the Overtemperature ΔT Function; therefore, no LSSS are specifically applied to the f(ΔI) trip Function.

In MODE 1 or 2, when the Overtemperature ΔT trip is required to be OPERABLE, the f(ΔI) Function must be OPERABLE because the f(ΔI) Function provides one of the inputs to the Overtemperature ΔT trip.

3. Power Range Neutron Flux Rate

The Power Range Neutron Flux Rate trips use the same channels as discussed for Function 2 above.

a. Power Range Neutron Flux – High Positive Rate

The Power Range Neutron Flux – High Positive Rate trip Function ensures that protection is provided against rapid increases in neutron flux which are characteristic of an RCCA drive rod housing rupture and the accompanying ejection of the RCCA. This Function compliments the Power Range Neutron Flux – High and Low Setpoint trip

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

a. Power Range Neutron Flux—High Positive Rate (continued)

Functions to ensure that the criteria are met for a rod ejection from the power range.

The LCO requires all four of the Power Range Neutron Flux—High Positive Rate channels to be OPERABLE.

In MODE 1 or 2, when there is a potential to add a large amount of positive reactivity from a rod ejection accident (REA), the Power Range Neutron Flux—High Positive Rate trip must be OPERABLE. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux—High Positive Rate trip Function does not have to be OPERABLE because other RTS trip Functions and administrative controls will provide protection against positive reactivity additions. Also, since only the shutdown banks may be withdrawn in MODE 3, 4, or 5, the remaining complement of control bank worth ensures a sufficient degree of SDM in the event of an REA. In MODE 6, no rods are withdrawn and the SDM is increased during refueling operations. The reactor vessel head is also removed or the closure bolts are detensioned preventing any pressure buildup. In addition, the NIS power range detectors cannot detect neutron levels present in this mode.

b. Power Range Neutron Flux—High Negative Rate

The Power Range Neutron Flux—High Negative Rate trip Function ensures that protection is provided for multiple rod drop accidents. At high power levels, a multiple rod drop accident could cause local flux peaking which would result in an unconservative local DNBR. DNBR is defined as the ratio of the heat flux required to cause a DNB at a particular location in the core to the local heat flux. The DNBR is indicative of the margin to DNB. No credit is taken for the operation of this Function for those rod drop accidents in which the local DNBRs will be greater than the limit.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

b. Power Range Neutron Flux–High Negative Rate (continued)

The LCO requires all four Power Range Neutron Flux–High Negative Rate channels to be OPERABLE.

In MODE 1 or 2, when there is potential for a multiple rod drop accident to occur, the Power Range Neutron Flux–High Negative Rate trip must be OPERABLE. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux–High Negative Rate trip Function does not have to be OPERABLE because the core is not critical and DNBR is not a concern. Also, since only the shutdown banks may be withdrawn in MODE 3, 4, or 5, the remaining complement of control bank worth ensures a sufficient degree of SDM in the event of an REA. In MODE 6, no rods are withdrawn and the required SDM is increased during refueling operations. In addition, the NIS power range detectors cannot detect neutron levels present in this MODE.

4. Intermediate Range Neutron Flux

The Intermediate Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank withdrawal accident from a subcritical condition during startup. This trip Function provides redundant protection to the Power Range Neutron Flux–Low Setpoint trip Function. The NIS intermediate range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS intermediate range detectors do not provide any input to control systems. Note that this Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

The LCO requires two channels of Intermediate Range Neutron Flux to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

4. Intermediate Range Neutron Flux (continued)

Because this trip Function is important only during startup, there is generally no need to disable channels for testing while the Function is required to be OPERABLE. Therefore, a third channel is unnecessary.

In MODE 1 below the P-10 setpoint and in MODE 2 when there is a potential for an uncontrolled rod withdrawal accident during reactor startup, the Intermediate Range Neutron Flux trip must be OPERABLE. Above the P-10 setpoint, the Power Range Neutron Flux–High Setpoint trip ~~and the Power Range Neutron Flux–High Positive Rate trip~~ provides core protection for a rod withdrawal accident. In MODE 3, 4, or 5, the Intermediate Range Neutron Flux trip does not have to be OPERABLE because the control rods must be fully inserted and only the shutdown rods may be withdrawn. The reactor cannot be started up in this condition. The core also has the required SDM to mitigate the consequences of a positive reactivity addition accident. In MODE 6, all rods are fully inserted and the core has a required increased SDM. Also, the NIS intermediate range detectors cannot detect neutron levels present in this MODE.

Protection from an uncontrolled rod withdrawl accident is provided by the source range neutron flux monitor.

5. Source Range Neutron Flux

The LCO requirement for the Source Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled bank rod withdrawal accident from a subcritical condition during startup. This trip Function provides redundant protection to the Power Range Neutron Flux–Low Setpoint and Intermediate Range Neutron Flux trip Functions. In MODES 3, 4, and 5, administrative controls also prevent the uncontrolled withdrawal of rods. The NIS source range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS source range detectors do not provide any inputs to control systems. The source range trip is the only RTS automatic protective Function required in MODES 3, 4, and 5. Therefore, the functional capability at the specified Trip Setpoint is assumed to be available.

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

5. Source Range Neutron Flux (continued)

The LCO requires two channels of Source Range Neutron Flux to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function. The LCO also requires one channel of the Source Range Neutron Flux to be OPERABLE in MODE 3, 4, or 5 with RTBs open. ~~In this case, the source range Function is to provide control room indication and input to the Boron Dilution Protection System (BDPS). The outputs of the Function to RTS logic are not required OPERABLE when the RTBs are open.~~ (9)

The Source Range Neutron Flux Function provides protection for control rod withdrawal from subcritical, boron dilution and control rod ejection events. The Function also provides visual neutron flux indication in the control room.

In MODE 2 when below the P-6 setpoint during a reactor startup, the Source Range Neutron Flux trip must be OPERABLE. Above the P-6 setpoint, the Intermediate Range Neutron Flux trip and the Power Range Neutron Flux-Low Setpoint trip will provide core protection for reactivity accidents. Above the P-6 setpoint, the NIS source range ~~detectors are de-energized and~~ Neutron Flux trip Function is disabled and inoperable. (A)

In MODE 3, 4, or 5 with the reactor shut down, the Source Range Neutron Flux trip Function must also be OPERABLE. If the CRD System is capable of rod withdrawal, the Source Range Neutron Flux trip must be OPERABLE to provide core protection against a rod withdrawal accident. If the CRD System is not capable of rod withdrawal, the source range detectors are not required to trip the reactor. However, their monitoring Function must be OPERABLE to monitor core neutron levels and provide indication of reactivity changes that may occur as a result of events like a boron dilution. ~~These inputs are provided to the BDPS.~~ (9) The requirements for the NIS source range detectors in MODE 6 are addressed in LCO 3.9.3, "Nuclear Instrumentation".

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

6. Overtemperature ΔT

The Overtemperature ΔT trip Function is provided to ensure that the design limit DNBR is met. This trip Function also limits the range over which the Overpower ΔT trip Function must provide protection. The inputs to the Overtemperature ΔT trip include ~~alle~~ ~~pressure~~ pressurizer pressure, A,F reactor coolant temperature, axial power distribution, and reactor power as indicated by loop ΔT assuming full reactor coolant flow. Protection from violating the DNBR limit is assured for those transients that are slow with respect to delays from the core to the measurement system. The Function monitors both variation in power and flow since a decrease in flow has the same effect on ΔT as a power increase. The Overtemperature ΔT trip Function uses each loop's ΔT as a measure of reactor power and is compared with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature – the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature;
- pressurizer pressure – the Trip Setpoint is varied to correct for changes in system pressure;
- axial power distribution – discussed under Function 2.c, f(ΔI).

Dynamic compensation is included for system piping delays from the core to the temperature measurement system.

The Overtemperature ΔT trip Function is calculated for each loop as described in Note 1 of Table 3.3.1-1. Trip occurs if Overtemperature ΔT is indicated in two loops. ~~At some units,~~ The pressure and temperature signals are used for other control Functions. ~~For those units,~~ The actuation logic must be able to withstand an input failure to the control system, which may then require the protection Function actuation, and a single failure in the other channels providing the protection Function actuation. Note that this Function also provides a signal to generate

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BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

6. Overtemperature ΔT (continued)

a turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overtemperature ΔT condition and may prevent a reactor trip.

The LCO requires all four channels of the Overtemperature ΔT trip Function to be OPERABLE in MODE 1 or 2. (D) ~~for two and four loop units (the LCO requires all three channels on the Overtemperature ΔT trip Function to be OPERABLE for three loop units).~~ Note that the Overtemperature ΔT Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overtemperature ΔT trip must be OPERABLE to prevent DNB. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

7. Overpower ΔT

The Overpower ΔT trip Function ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions. This trip Function also limits the required range of the Overtemperature ΔT trip Function and provides a backup to the Power Range Neutron Flux—High Setpoint trip. The Overpower ΔT trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. It uses the ΔT of each loop as a measure of reactor power ~~and~~ is automatically varied with the following parameters: with a setpoint that (H)

- reactor coolant average temperature—the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature; and

- rate of change of reactor coolant average temperature—including dynamic compensation for

(continued)

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

7. Overpower ΔT (continued)

the delays between the core and the temperature measurement system.

Therefore,

The Overpower ΔT trip Function is calculated for each loop as per Note 2 of Table 3.3.1-1. Trip occurs if Overpower ΔT is indicated in two loops. ~~At some units,~~ The temperature signals are used for other control Functions. ~~At those units,~~ The actuation logic must be able to withstand an input failure to the control system which may then require the protection Function actuation and a single failure in the remaining channels providing the protection Function actuation. Note that this Function also provides a signal to generate a turbine runback prior to reaching the ~~ALLOWABLE VALUE.~~ TRIP SETPOINT A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overpower ΔT condition and may prevent a reactor trip.

STGT

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The LCO requires four channels ~~for two and four loop units (three channels for three loop units)~~ of the Overpower ΔT trip Function to be OPERABLE. Note that the Overpower ΔT Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

A

In MODE 1 or 2, the Overpower ΔT trip Function must be OPERABLE. These are the only times that enough heat is generated in the fuel to be concerned about the heat generation rates and overheating of the fuel. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

8. Pressurizer Pressure

The same sensors provide input to the Pressurizer Pressure-High and -Low trips and the Overtemperature ΔT trip. ~~At some units,~~ The Pressurizer Pressure channels are also used to provide input to the Pressurizer Pressure Control System. ~~For those units,~~ Therefore, the actuation logic must be able to withstand an input

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(continued)

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

8. Pressurizer Pressure (continued)

failure to the control system (which may then require the protection Function actuation) and a single failure in the other channels providing the protection Function actuation.

a. Pressurizer Pressure – Low

The Pressurizer Pressure – Low trip Function ensures that protection is provided against violating the DNBR limit due to low pressure.

D The LCO requires all four channels ~~for two and four loop units, (three channels for three loop units)~~ of Pressurizer Pressure – Low to be OPERABLE in MODE 1 above P-7. ~~This trip Function ensures that protection is provided against violating the DNBR limit due to low pressure.~~ 4

In MODE 1, when DNB is a major concern, the Pressurizer Pressure – Low trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock (NIS power range P-10 or turbine impulse pressure greater than approximately 10% of full power equivalent (P-13)). On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 setpoint, no conceivable power distributions can occur that would cause DNB concerns.

b. Pressurizer Pressure – High

The Pressurizer Pressure – High trip Function ensures that protection is provided against overpressurizing the RCS. This trip Function operates in conjunction with the pressurizer relief and safety valves to prevent RCS overpressure conditions.

D The LCO requires all four channels ~~for two and four loop units (three channels for three loop units)~~ of the Pressurizer Pressure – High to be OPERABLE.

The Pressurizer Pressure – High LSSS is selected to be below the pressurizer safety valve

(continued)

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

b. Pressurizer Pressure–High (continued)

actuation pressure and above the power operated relief valve (PORV) setting. This setting minimizes challenges to safety valves while avoiding unnecessary reactor trip for those pressure increases which can be controlled by the PORVs.

In MODE 1 or 2, the Pressurizer Pressure–High trip must be OPERABLE to help prevent RCS overpressurization and minimize challenges to the relief and safety valves. In MODE 3, 4, 5, or 6, the Pressurizer Pressure–High trip Function does not have to be OPERABLE because transients which could cause an overpressure condition will be slow to occur. Therefore, the operator will have sufficient time to evaluate unit conditions and take corrective actions. Additionally, low temperature overpressure protection systems provide overpressure protection when below MODE 4.

9. Pressurizer Water Level–High

The Pressurizer Water Level–High trip Function provides a backup signal for the Pressurizer Pressure–High trip and also provides protection against water relief through the pressurizer safety valves. These valves are designed to pass steam in order to achieve their design energy removal rate. A reactor trip is actuated prior to the pressurizer becoming water solid. The LCO requires three channels of Pressurizer Water Level–High to be OPERABLE. The pressurizer level channels are used as input to the Pressurizer Level Control System. A fourth channel is not required to address control/protection interaction concerns. The level channels do not actuate the safety valves, and the high pressure reactor trip is set below the safety valve setting. Therefore, with the slow rate of charging available, pressure overshoot due to level channel failure cannot cause the safety valve to lift before reactor high pressure trip.

(continued)

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

9. Pressurizer Water Level—High (continued)

In MODE 1 when there is a potential for overfilling the pressurizer, the Pressurizer Water Level—High trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock. On decreasing power this trip Function is automatically blocked below P-7. Below the P-7 setpoint, transients which could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate unit conditions and take corrective actions.

10. Reactor Coolant Flow—Low

a. Reactor Coolant Flow—Low (Single Loop)

The Reactor Coolant Flow—Low (Single Loop) trip Function ensures that protection is provided against violating the DNBR limit due to low flow in one or more RCS loops, while avoiding reactor trips due to normal variations in loop flow. Above the P-8 setpoint, which is approximately 48% of RTP, a loss of flow in any RCS loop will actuate a reactor trip. Each RCS loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

The LCO requires three Reactor Coolant Flow—Low channels per loop to be OPERABLE in MODE 1 above P-8.

In MODE 1 above the P-8 setpoint, a loss of flow in one RCS loop could result in DNB conditions in the core. In MODE 1 below the P-8 setpoint, a loss of flow in two or more loops is required to actuate a reactor trip (Function 10.b) because of the lower power level and the greater margin to the design limit DNBR.

b. Reactor Coolant Flow—Low (Two Loops)

The Reactor Coolant Flow—Low (Two Loops) trip Function ensures that protection is provided against violating the DNBR limit due to low flow in two or more RCS loops while avoiding reactor trips due to normal variations in loop flow.

(continued)

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

b. Reactor Coolant Flow—Low (Two Loops) (continued)

Above the P-7 setpoint and below the P-8 setpoint, a loss of flow in two or more loops will initiate a reactor trip. Each loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

The LCO requires three Reactor Coolant Flow—Low channels per loop to be OPERABLE.

In MODE 1 above the P-7 setpoint and below the P-8 setpoint, the Reactor Coolant Flow—Low (Two Loops) trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on low flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on low flow in two or more RCS loops is automatically enabled. Above the P-8 setpoint, a loss of flow in any one loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR.

11. Reactor Coolant Pump (RCP) Breaker Position

Both RCP Breaker Position trip Functions operate together on two sets of auxiliary contacts, with one set on each RCP breaker. These Functions anticipate the Reactor Coolant Flow—Low trips to avoid RCS heat up that would occur before the low flow trip actuates.

1

a. Reactor Coolant Pump Breaker Position (Single Loop)

The RCP Breaker Position (Single Loop) reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in one RCS loop. The position of each RCP breaker is monitored. If one RCP breaker is open above the P-8 setpoint, a reactor trip is initiated. This trip Function will generate a

(continued)

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APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

a. Reactor Coolant Pump Breaker Position (Single Loop) (continued)

reactor trip before the Reactor Coolant Flow—Low (Single Loop) Trip Setpoint is reached.

The LCO requires one RCP Breaker Position channel per RCP to be OPERABLE. One OPERABLE channel is sufficient for this trip Function because the RCS Flow—Low trip alone provides sufficient protection of unit SLs for loss of flow events. The RCP Breaker Position trip serves only to anticipate the low flow trip, minimizing the thermal transient associated with loss of a pump.

This Function measures only the discrete position (open or closed) of the RCP breaker, using a position switch. Therefore, the Function has no adjustable trip setpoint with which to associate an LSSS.

In MODE 1 above the P-8 setpoint, when a loss of flow in any RCS loop could result in DNB conditions in the core, the RCP Breaker Position (Single Loop) trip must be OPERABLE. In MODE 1 below the P-8 setpoint, a loss of flow in two or more loops is required to actuate a reactor trip because of the lower power level and the greater margin to the design limit DNBR.

b. Reactor Coolant Pump Breaker Position (Two Loops)

The RCP Breaker Position (Two Loops) reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops. The position of each RCP breaker is monitored. Above the P-7 setpoint and below the P-8 setpoint, a loss of flow in two or more loops will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow—Low (Two Loops) Trip Setpoint is reached.

The LCO requires one RCP Breaker Position channel per RCP to be OPERABLE. One OPERABLE channel is sufficient for this Function because the RCS

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

b. Reactor Coolant Pump Breaker Position (Two Loops) (continued)

Flow-Low trip alone provides sufficient protection of unit SLs for loss of flow events. The RCP Breaker Position trip serves only to anticipate the low flow trip, minimizing the thermal transient associated with loss of an RCP.

This Function measures only the discrete position (open or closed) of the RCP breaker using a position switch. Therefore, the Function has no adjustable trip setpoint with which to associate an LSSS.

In MODE 1 above the P-7 setpoint and below the P-8 setpoint, the RCP Breaker Position (Two Loops) trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on loss of flow in two RCS loops is automatically enabled. Above the P-8 setpoint, a loss of flow in any one loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR.

~~12.~~ 11. Undervoltage Reactor Coolant Pumps

The Undervoltage RCPs reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops. The voltage to each RCP is monitored. Above the P-7 setpoint ~~and below the P-8 setpoint~~, a loss of voltage detected on two or more RCP buses will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow-Low (Two Loops) Trip Setpoint is reached. Time delays are incorporated into the Undervoltage RCPs channels to prevent reactor trips due to momentary electrical power transients.

The loss of voltage in two loops must be sustained for a length of time equal to or greater than that set in the time delay

The LCO requires ~~three~~ one Undervoltage RCP~~s~~ channel~~s~~ ~~(one per phase)~~ per bus to be OPERABLE.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

~~12.~~ 11. Undervoltage Reactor Coolant Pumps (continued)

Above the P-7 setpoint, a loss of voltage detected in two or more loops will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow—Low (Two Loops) Trip Setpoint is reached. The loss of voltage in two loops must be sustained for a length of time equal to or greater than that set in the time delays. Time delays are incorporated into the Undervoltage RCPs channels to prevent reactor trips due to momentary electrical power transients.

In MODE 1 above the P-7 setpoint, the Undervoltage RCP trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on loss of flow in two or more RCS loops is automatically enabled. This Function uses the same relays as the ESFAS Function 6.f, "Undervoltage Reactor Coolant Pump (RCP)" start of the AFW pumps.

A D

~~13.~~ 12. Underfrequency Reactor Coolant Pumps

The Underfrequency RCPs reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops from a major network frequency disturbance. An underfrequency condition will slow down the pumps, thereby reducing their coastdown time following a pump trip. The proper coastdown time is required so that reactor heat can be removed immediately after reactor trip. The frequency of each RCP bus is monitored. Above the P-7 setpoint ~~and below the P-8 setpoint~~, a loss of frequency detected on two or more RCP buses will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow—Low (Two Loops) Trip Setpoint is reached. Time delays are incorporated into the Underfrequency RCPs channels to prevent reactor trips due to momentary electrical power transients.

A The LCO requires ~~three~~ one Underfrequency RCP~~s~~ channel~~s~~ per bus to be OPERABLE.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

~~13.~~ 12. Underfrequency Reactor Coolant Pumps (continued)

Above the P-7 setpoint, a loss of frequency detected in two or more RCP buses will initiate a reactor trip.

This trip Function will generate a reactor trip before the Reactor Coolant Flow–Low Trip Setpoint is reached. Time delays are incorporated into the Underfrequency RCPs channels to prevent reactor trips due to momentary electrical power transients.

In MODE 1 above the P-7 setpoint, the Underfrequency RCPs trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on loss of flow in two or more RCS loops is automatically enabled.

~~14.~~ 13. Steam Generator Water Level–Low Low

The SG Water Level–Low Low trip Function ensures that protection is provided against a loss of heat sink and actuates the AFW System prior to uncovering the SG tubes. The SGs are the heat sink for the reactor. In order to act as a heat sink, the SGs must contain a minimum amount of water. A narrow range low low level in any SG is indicative of a loss of heat sink for the reactor. The level transmitters provide input to the SG Level Control System. ~~Therefore, the actuation logic must be able to withstand an input failure to the control system which may then require the protection Function actuation and a single failure in the other channels providing the protection Function actuation.~~ This Function also performs the ESFAS function of starting the AFW pumps on low low SG level.

INSERT "YY" (new paragraph) (C)

The LCO requires ~~four~~ three channels of SG Water Level–Low Low per SG to be OPERABLE. ~~for four loop units in which these channels are shared between protection and control. In two, three, and four loop units where three SG Water Levels are dedicated to the RTS, only three channels per SG are required to be OPERABLE.~~

(D)

(continued)

INSERT YY

Control/protection interaction is addressed by the use of a Median Signal Selector which prevents a single failure of a channel providing input to the control system requiring protection function action. That is, a single failure of a channel providing input to the control system does not result in the control system initiating a condition requiring protection function action. The Median Signal Selector performs this by <u>not</u> selecting the channels indicating the highest or lowest steam generator levels as input to the control system.

Because one failed protection instrument channel would not result in an adverse control system action, a second random protection system failure (as otherwise required by IEEE 279-1971) need not be considered.

The Steam Generator Water Level Trip Time Delay (TTD) creates additional operational margin when the plant needs it most, during escalation to power, by allowing the operator time to recover level when the primary side load is sufficiently small to allow such action. The TTD is based on continuous monitoring of primary side power through the use of vessel $\Delta T$. Two time delays are calculated, based on the number of steam generators indicating less than the Low-Low Trip Setpoint. The magnitude of the delays decreases with increasing primary side power level, up to 50% RTP. Above 50% RTP there are no time delays for the Low-Low Level channel trips.

In the event of failure of a Steam Generator Water Level channel, the channel is placed in the trip condition as input to the Solid State Protection System (SSPS) and does not affect the TTD setpoint calculations for the remaining OPERABLE channels. It is then necessary for the operator to force the use of the shorter TTD time delay by adjustment of the single steam generator time delay calculation ($T_s$) to match the multiple steam generator time delay calculation ($T_m$) for the affected protection set, through the Man Machine Interface. Failure of the vessel $\Delta T$ channel input (failure of more than one $T_H$ RTD or failure of $T_C$ RTDs) does not affect the TTD calculation for a protection set. This results in the requirement that the operator adjust the threshold power level for zero seconds time delay from 50% RTP to 0% RTP, through the Man Machine Interface.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

~~14~~. 13. Steam Generator Water Level – Low Low (continued)

In MODE 1 or 2, when the reactor requires a heat sink, the SG Water Level – Low Low trip must be OPERABLE. The normal source of water for the SGs is the Main Feedwater (MFW) System (not safety related). ~~The MFW System is only in operation in MODE 1 or 2.~~ The AFW System is the safety related backup source of water to ensure that the SGs remain the heat sink for the reactor. During normal startups and shutdowns, the MFW → ~~AFW~~ System provides feedwater to maintain SG level. In MODE 3, 4, 5, or 6, the SG Water Level – Low Low Function does not have to be OPERABLE because ~~the MFW System is not in operation and the~~ reactor is not operating or even critical. Decay heat removal is accomplished by the AFW (or MFW) System in MODE 3 and by the Residual Heat Removal (RHR) System in MODE 4, 5, or 6.

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~~15~~. 14. Steam Generator Water Level – Low, Coincident With Steam Flow/Feedwater Flow Mismatch

SG Water Level – Low, in conjunction with the Steam Flow/Feedwater Flow Mismatch, ensures that protection is provided against a loss of heat sink and actuates the AFW System prior to uncovering the SG tubes. In addition to a decreasing water level in the SG, the difference between feedwater flow and steam flow is evaluated to determine if feedwater flow is significantly less than steam flow. With less feedwater flow than steam flow, SG level will decrease at a rate dependent upon the magnitude of the difference in flow rates. There are two SG level channels and two Steam Flow/Feedwater Flow Mismatch channels per SG. One narrow range level channel sensing a low level coincident with one Steam Flow/Feedwater Flow Mismatch channel sensing flow mismatch (steam flow greater than feed flow) will actuate a reactor trip.

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The LCO requires two channels of SG Water Level – Low coincident with Steam Flow/Feedwater Flow Mismatch.

In MODE 1 or 2, when the reactor requires a heat sink, the SG Water Level – Low coincident with Steam Flow/Feedwater Flow Mismatch trip must be OPERABLE. The normal source of water for the SGs is the MFW

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

15. Steam Generator Water Level – Low, Coincident With Steam Flow/Feedwater Flow Mismatch (continued)

System (not safety related). The MFW System is only in operation in MODE 1 or 2. The AFW System is the safety related backup source of water to ensure that the SGs remain the heat sink for the reactor. During normal startups and shutdowns, the AFW System provides feedwater to maintain SG level. In MODE 3, 4, 5, or 6, the SG Water Level – Low coincident with Steam Flow/Feedwater Flow Mismatch Function does not have to be OPERABLE because the MFW System is not in operation and the reactor is not operating or even critical. Decay heat removal is accomplished by the AFW System in MODE 3 and by the RHR System in MODE 4, 5, or 6. The MFW System is in operation only in MODE 1 or 2 and, therefore, this trip Function need only be OPERABLE in these MODES.

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14 ~~16.~~ ~~Turbine Trip~~ ← stet

14. a. Turbine Trip – Low Fluid Oil Pressure

G reactor trip on

The Turbine Trip – Low Fluid Oil Pressure ~~trip Function~~ is anticipatory for the loss of heat removal capabilities of the secondary system following a turbine trip. This trip Function acts to minimize the pressure/temperature transient on the reactor. Any turbine trip from a power level below the P-9 setpoint, approximately 50% power, will not actuate a reactor trip. Three pressure switches monitor the control oil pressure in the Turbine Electrohydraulic Control System. A low pressure sensed by two out of three of the pressure switches will actuate a reactor trip. These pressure switches do not provide any input to the control system. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure – High trip Function and RCS integrity is ensured by the pressurizer safety valves.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

14. a. Turbine Trip – Low Fluid Oil Pressure (continued)

The LCO requires three channels of Turbine Trip – Low Fluid Oil Pressure to be OPERABLE in MODE 1 above P-9.

Below the P-9 setpoint, a turbine trip does not actuate a reactor trip. In MODE 2, 3, 4, 5, or 6, there is no potential for a turbine trip, and the Turbine Trip – Low Fluid Oil Pressure trip Function does not need to be OPERABLE.

14. b. Turbine Trip – Turbine Stop Valve Closure

The Turbine Trip – Turbine Stop Valve Closure ~~trip Function~~ is anticipatory for the loss of heat removal capabilities of the secondary system following a turbine trip from a power level below the P-9 setpoint, approximately 50% power. This ~~action~~ will not actuate a reactor trip. The trip Function anticipates the loss of secondary heat removal capability that occurs when the stop valves close. Tripping the reactor in anticipation of loss of secondary heat removal acts to minimize the pressure and temperature transient on the reactor. This trip Function will not and is not required to operate in the presence of a single channel failure. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure – High trip Function, and RCS integrity is ensured by the pressurizer safety valves. This trip Function is diverse to the Turbine Trip – Low Fluid Oil Pressure trip Function. Each turbine stop valve is equipped with one limit switch that inputs to the RTS. If all four limit switches indicate that the stop valves are all closed, a reactor trip is initiated.

The LSSS for this Function is set to assure channel trip occurs when the associated stop valve is completely closed.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

14.b. Turbine Trip – Turbine Stop Valve Closure (continued)

The LCO requires four Turbine Trip – Turbine Stop Valve Closure channels, one per valve, to be OPERABLE in MODE 1 above P-9. All four channels must trip to cause reactor trip.

Below the P-9 setpoint, a load rejection can be accommodated by the Steam Dump System. In MODE 2, 3, 4, 5, or 6, there is no potential for a load rejection, and the Turbine Trip – Stop Valve Closure trip Function does not need to be OPERABLE.

~~14.~~ 15. Safety Injection Input from Engineered Safety Feature Actuation System

The SI Input from ESFAS ensures that if a reactor trip has not already been generated by the RTS, the ESFAS automatic actuation logic will initiate a reactor trip upon any signal which initiates SI. This is a condition of acceptability for the LOCA. However, other transients and accidents take credit for varying levels of ESF performance and rely upon rod insertion, except for the most reactive rod which is assumed to be fully withdrawn, to ensure reactor shutdown. Therefore, a reactor trip is initiated every time an SI signal is present.

Trip Setpoint and Allowable Values are not applicable to this Function. The SI Input is provided by ~~relay~~ Solid State logic in the ESFAS. Therefore, there is no measurement signal with which to associate an LSSS. (A)

The LCO requires two trains of SI Input from ESFAS to be OPERABLE in MODE 1 or 2.

(G) A reactor trip is initiated every time an SI signal is present. Therefore, this trip Function must be OPERABLE in MODE 1 or 2, when the reactor is critical, ~~and must be shut down in the event of an accident~~. In MODE 3, 4, 5, or 6, the reactor is not critical, and this trip Function does not need to be OPERABLE.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

~~18.~~ 16. Reactor Trip System Interlocks

Reactor protection interlocks are provided to ensure reactor trips are in the correct configuration for the current unit status. They back up operator actions to ensure protection system Functions are not bypassed during unit conditions under which the safety analysis assumes the Functions are not bypassed. Therefore, the interlock Functions do not need to be OPERABLE when the unit is not in associated reactor trip functions ~~are outside~~ the applicable MODES. These are: which required the

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16. a. Intermediate Range Neutron Flux, P-6

The Intermediate Range Neutron Flux, P-6 interlock is actuated when any NIS intermediate range channel goes approximately one decade above the minimum channel reading. If both channels drop below the setpoint, the permissive will automatically be defeated. The LCO requirement for the P-6 interlock ensures that the following Functions are performed:

- on increasing power, the P-6 interlock allows the manual block of the NIS Source Range, Neutron Flux reactor trip. This prevents a premature block of the source range trip and allows the operator to ensure that the intermediate range is OPERABLE prior to leaving the source range; when the source range trip is blocked, the high voltage to the detectors is also removed;

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- on decreasing power, the P-6 interlock automatically energizes the NIS source range detectors and enables the NIS Source Range Neutron Flux reactor trip; and

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- ~~on increasing power, the P-6 interlock provides a backup block signal to the source range flux doubling circuit. Normally, this function is manually blocked by the control room operator during the reactor startup.~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

16.a. Intermediate Range Neutron Flux, P-6 (continued)

The LCO requires two channels of Intermediate Range Neutron Flux, P-6 interlock to be OPERABLE in MODE 2 when below the P-6 interlock setpoint.

Above the P-6 interlock setpoint, the NIS Source Range Neutron Flux reactor trip will be blocked, and this Function will no longer be necessary. In MODE 3, 4, 5, or 6, the P-6 interlock does not have to be OPERABLE because the NIS Source Range is providing core protection.

16.b. Low Power Reactor Trips Block, P-7

The Low Power Reactor Trips Block, P-7 interlock is actuated by input from either the Power Range Neutron Flux, P-10, or the Turbine Impulse Pressure, P-13 interlock. The LCO requirement for the P-7 interlock ensures that the following Functions are performed:

(1) on increasing power, the P-7 interlock automatically enables reactor trips on the following Functions:

- Pressurizer Pressure—Low;
- Pressurizer Water Level—High;
- Reactor Coolant Flow—Low (Two Loops);
- A RCPs Breaker Open (Two Loops);
- Undervoltage RCPs; and
- Underfrequency RCPs.

These reactor trips are only required when operating above the P-7 setpoint (approximately 10% power). The reactor trips provide protection against violating the DNBR limit. Below the P-7 setpoint, the RCS is capable of providing sufficient natural circulation without any RCP running.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

b. Low Power Reactor Trips Block, P-7 (continued)

(2) on decreasing power, the P-7 interlock automatically blocks reactor trips on the following Functions:

- Pressurizer Pressure – Low;
- Pressurizer Water Level – High;
- Reactor Coolant Flow – Low (Two Loops);
- A RCP Breaker Position (Two Loops);
- Undervoltage RCPs; and
- Underfrequency RCPs.

Trip Setpoint and Allowable Value are not applicable to the P-7 interlock because it is a logic Function and thus has no parameter with which to associate an LSSS.

The P-7 interlock is a logic Function with train and not channel identity. Therefore, the LCO requires one channel per train of Low Power Reactor Trips Block, P-7 interlock to be OPERABLE in MODE 1.

The low power trips are blocked below the P-7 setpoint and unblocked above the P-7 setpoint. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the interlock performs its Function when power level drops below 10% power, which is in MODE 1.

c. Power Range Neutron Flux, P-8

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Above approximately 48% power

The Power Range Neutron Flux, P-8 interlock is actuated at approximately 48% power as determined by two out of four NIS power range detectors. A The P-8 interlock automatically enables the Reactor Coolant Flow – Low (Single Loop) ~~and RCP Breaker Position (Single Loop)~~ reactor trips on low flow in one or more RCS loops on increasing power. The LCO requirement for this trip

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

c. Power Range Neutron Flux, P-8 (continued)

Function ensures that protection is provided against a loss of flow in any RCS loop that could result in DNB conditions in the core when greater than approximately 48% power. On decreasing power, the reactor trip on low flow in any loop is automatically blocked.

The LCO requires four channels of Power Range Neutron Flux, P-8 interlock to be OPERABLE in MODE 1.

F In MODE 1 a loss of flow in one RCS loop could result in DNB conditions, so the Power Range Neutron Flux, P-8 interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the core is not producing sufficient power to be concerned about DNB conditions.

d. Power Range Neutron Flux, P-9

The Power Range Neutron Flux, P-9 interlock is actuated at approximately 50% power as determined by two out of four NIS power range detectors. The LCO requirement for this Function ensures that the Turbine Trip-Low Fluid Oil Pressure and G Turbine Trip-Turbine Stop Valve Closure reactor trips are enabled above the P-9 setpoint. Above the P-9 setpoint, a turbine trip will cause a load rejection beyond the capacity of the Steam Dump System. A reactor trip is automatically initiated on a turbine trip when it is above the P-9 setpoint to minimize the transient on the reactor.

A,5 Combined and Rod Control System

The LCO requires four channels of Power Range Neutron Flux, P-9 interlock to be OPERABLE in MODE 1.

In MODE 1 a turbine trip could cause a load rejection beyond the capacity of the Steam Dump System, so the Power Range Neutron Flux interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

d. Power Range Neutron Flux, P-9 (continued)

reactor is not at a power level sufficient to have a load rejection beyond the capacity of the Steam Dump System.

16 e. Power Range Neutron Flux, P-10

The Power Range Neutron Flux, P-10 interlock is actuated at approximately 10% power as determined by two out of four NIS power range detectors. If power level falls below 10% ~~RTP~~ power on 3 of 4 channels, the nuclear instrument trips will be automatically unblocked. The LCO requirement for the P-10 interlock ensures that the following Functions are performed:

- on increasing power, the P-10 interlock allows the operator to manually block the Intermediate Range Neutron Flux reactor trip. Note that blocking the reactor trip also blocks the signal to prevent automatic and manual rod withdrawal;
- on increasing power, the P-10 interlock allows the operator to manually block the Power Range Neutron Flux—Low reactor trip;
- on increasing power, the P-10 interlock automatically provides a backup signal to block the Source Range Neutron Flux reactor trip and also to de-energize the NIS source range detectors;
- the P-10 interlock provides one of the two inputs to the P-7 interlock; and
- on decreasing power, the P-10 interlock automatically enables the Power Range Neutron Flux—Low reactor trip and the Intermediate Range Neutron Flux reactor trip (and rod stop).

The LCO requires four channels of Power Range Neutron Flux, P-10 interlock to be OPERABLE in MODE 1 or 2.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

~~19~~. 17 Reactor Trip Breakers (continued)

upon the system configuration. Two OPERABLE trains ensure no single random failure can disable the RTS trip capability.

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the RTBs or associated bypass breakers are closed, and the CRD System is capable of rod withdrawal.

~~20~~. 18 Reactor ~~Trip~~ F Trip Breaker Undervoltage and Shunt Trip Mechanisms

The LCO requires both the Undervoltage and Shunt Trip Mechanisms to be OPERABLE for each RTB that is in service. The trip mechanisms are not required to be OPERABLE for trip breakers that are open, racked out, incapable of supplying power to the CRD System, or declared inoperable under Function ~~19~~ 17 above. OPERABILITY of both trip mechanisms on each breaker ensures that no single trip mechanism failure will prevent opening any breaker on a valid signal.

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the RTBs and associated bypass breakers are closed, and the CRD System is capable of rod withdrawal.

~~21~~. 19, Automatic Trip Logic

The LCO requirement for the RTBs (Functions ~~19~~ 17 and ~~20~~ 18) and Automatic Trip Logic (Function ~~21~~ 19) ensures that means are provided to interrupt the power to allow the rods to fall into the reactor core. Each RTB is equipped with an undervoltage coil and a shunt trip coil to trip the breaker open when needed. Each RTB is equipped with a bypass breaker to allow testing of the trip breaker while the unit is at power. The reactor trip signals generated by the RTS Automatic Trip Logic cause the RTBs and associated bypass breakers to open and shut down the reactor.

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCOs, and APPLICABILITY

~~21~~. 19, Automatic Trip Logic (continued)

The LCO requires two trains of RTS Automatic Trip Logic to be OPERABLE. Having two OPERABLE channels ensures that random failure of a single logic channel will not prevent reactor trip.

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the RTBs and associated bypass breakers are closed, and the CRD System is capable of rod withdrawal.

The RTS instrumentation satisfies Criterion 3 of the NRC Policy Statement.

ACTIONS

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.1-1. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

D ~~Reviewer Note: Certain LCO Completion Times are based on approved topical reports. In order for a licensee to use the times the licensee must justify the Completion Times as required by the staff SER for the topical report.~~

Setpoint comparator ~~output~~ Trip output, C

Contact output 18.

In the event a channel's Trip Setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition entered for the particular protection Function affected.

When the number of inoperable channels in a trip Function exceed those specified in one or other related Conditions associated with a trip Function, then the unit is outside the safety analysis. Therefore, LCO 3.0.3 must be immediately entered if applicable in the current MODE of operation.

(continued)

BASES

ACTIONS (continued)

A.1

Condition A is applicable to all RTS protection Functions. Condition A addresses the situation where one or more required channels for one or more Functions are inoperable at the same time. The Required Action is to refer to Table 3.3.1-1 and to take the Required Actions for the protection Functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

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

Condition B applies to the Manual Reactor Trip in MODE 1 or 2. This action addresses the train orientation of the SSPS for this Function. With one channel inoperable, the inoperable channel must be restored to OPERABLE status within 48 hours. In this Condition, the remaining OPERABLE channel is adequate to perform the safety Function.

The Completion Time of 48 hours is reasonable considering that there are two automatic actuation trains and another manual initiation channel OPERABLE and the low probability of an event occurring during this interval.

If the Manual Reactor Trip Function cannot be restored to OPERABLE status within the allowed 48 hour Completion Time, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 additional hours (54 hours total time) followed by opening the RTBs within 1 additional hour (55 hours total time). The 6 additional hours to reach MODE 3 and the 1 hour to open the RTBs are reasonable, based on operating experience, to reach MODE 3 and open the RTBs from full power operation in an orderly manner and without challenging unit systems. With the RTBs open and the unit in MODE 3, this trip Function is no longer required to be OPERABLE.

(continued)

BASES

ACTIONS

D.1.1, D.1.2, D.2.1, D.2.2, and D.3 (continued)

In addition to placing the inoperable channel in the tripped condition, THERMAL POWER must be reduced to ≤ 75% of RTP within ~~4~~ 12 hours. Reducing the power level prevents operation of the core with radial power distributions beyond the design limits. With one of the NIS power range detectors inoperable, 1/4 of the radial power distribution monitoring capability is lost.

As an alternative to the above actions, the inoperable channel can be placed in the tripped condition within 6 hours and the QPTR monitored once every 12 hours as per SR 3.2.4.2, QPTR verification. Calculating QPTR every 12 hours compensates for the lost monitoring capability due to the inoperable NIS power range channel and allows continued unit operation at power levels ≥ 75% of RTP. The 6 hour Completion Time and the 12 hour Frequency are consistent with LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

If the Required Actions described above cannot be met within the specified Completion Times, the unit must be placed in a MODE where this Function is no longer required OPERABLE. An additional 6 hours beyond the Completion Time for Required Action D.1.~~2~~1 and Required Action D.2.1 are allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. If Required Actions D.2.2 cannot be completed within their allowed Completion Times, LCO 3.0.3 must be entered.

The Required Actions have been modified by a Note, which allows placing the inoperable channel or one additional channel in the bypass condition for up to 4 hours while performing routine surveillance testing of other channels. The 4 hour time limit is justified in Reference 7.

The Note also allows placing the inoperable channel into the bypass condition to allow setpoint adjustments of other channels when required to reduce the setpoint in accordance with other technical specifications

E.1 and E.2

Condition E applies to the following reactor trip Functions:

- Power Range Neutron Flux—Low;
- Power Range Neutron Flux—f(ΔI);

(continued)

BASES

ACTIONS

E.1 and E.2 (continued)

- Overtemperature ΔT;
- Overpower ΔT;
- Power Range Neutron Flux—High Positive Rate;
- Power Range Neutron Flux—High Negative Rate;
- Pressurizer Pressure—High;
- ~~SG Water Level—Low Low; and~~
- ~~SG Water Level—Low coincident with Steam Flow/Feedwater Flow Mismatch.~~

C

A known inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one out of two logic for actuation of the two out of three trips and one out of three logic for actuation of the two out of four trips. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 7.

If the operable channel cannot be placed in the trip condition within the specified Completion Time, the unit must be placed in a MODE where these Functions are not required OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to place the unit in MODE 3 from full power in an orderly manner and without challenging unit systems.

C

or one additional channel

The Required Actions have been modified by a Note which allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 7.

F.1 and F.2

Condition F applies to the Intermediate Range Neutron Flux trip when THERMAL POWER is above the P-6 setpoint and below

(continued)

BASES

ACTIONS

F.1 and F.2 (continued)

the P-10 setpoint and one channel is inoperable. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. If THERMAL POWER is greater than the P-6 setpoint but less than the P-10 setpoint, 2 hours is allowed to reduce THERMAL POWER below the P-6 setpoint or increase to THERMAL POWER above the P-10 setpoint. The NIS Intermediate Range Neutron Flux channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10. If THERMAL POWER is greater than the P-10 setpoint, the NIS power range detectors perform the monitoring and protective Functions and the intermediate range is not required. The Completion Times allow for a slow and controlled power adjustment above P-10 or below P-6 and take into account the redundant capability afforded by the redundant OPERABLE channel and the low probability of its failure during this period. This action does not require the inoperable channel to be tripped because the Function uses one out of two logic. Tripping one channel would trip the reactor. Thus, the Required Actions specified in this Condition are only applicable when channel failure does not result in reactor trip.

G.1 and G.2

Condition G applies to two inoperable Intermediate Range Neutron Flux trip channels in MODE 2 when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint. Required Actions specified in this Condition are only applicable when channel failures do not result in reactor trip. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. With no intermediate range channels OPERABLE, the Required Actions are to suspend operations involving positive reactivity additions immediately. This will preclude any power level increase since there are no OPERABLE Intermediate Range Neutron Flux channels. The operator must also reduce THERMAL POWER below the P-6 setpoint within two hours. Below P-6, the Source Range Neutron Flux channels will be able to monitor the core power level. The Completion Time of 2 hours will allow a slow and controlled power reduction to less than the P-6 setpoint and

(continued)

BASES

ACTIONS

G.1 and G.2 (continued)

takes into account the low probability of occurrence of an event during this period that may require the protection afforded by the NIS Intermediate Range Neutron Flux trip.

H.1

Condition H applies to the Intermediate Range Neutron Flux trip when THERMAL POWER is below the P-6 setpoint and one or two channels are inoperable. Below the P-6 setpoint, the NIS source range performs the monitoring and protective Functions. The inoperable NIS intermediate range channel(s) must be returned to OPERABLE status prior to increasing power above the P-6 setpoint. The NIS intermediate range channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10.

I.1

Condition I applies to one inoperable Source Range Neutron Flux trip channel when in MODE 2, below the P-6 setpoint, and performing a reactor startup. With the unit in this Condition, < P-6, the NIS source range performs the monitoring and protective Functions. With one of the two channels inoperable, operations involving positive reactivity additions shall be suspended immediately.

This will preclude any power escalation. With only one source range channel OPERABLE, core protection is severely reduced and any actions that add positive reactivity to the core must be suspended immediately.

J.1 and J.2

Condition J applies to two inoperable Source Range Neutron Flux trip channels when in MODE 2, below the P-6 setpoint, and performing a reactor startup; or in MODE 3, 4, or 5 with the RTBs closed and the CRD System capable of rod withdrawal. With the unit in this Condition, < P-6, the NIS source range performs the monitoring and protective Functions. With both source range channels inoperable,

(continued)

BASES

ACTIONS

J.1 and J.2 (continued)

operations involving positive reactivity additions shall be suspended immediately. With no source range channels OPERABLE, core protection is severely reduced and any actions that add positive reactivity to the core must be suspended immediately. In addition to suspending positive reactivity additions, the RTBs must be opened immediately. With the RTBs open ~~and positive reactivity additions suspended,~~ the core is in a ~~relatively safe and~~ stable condition, ~~With the RTBs open,~~ the unit enters Condition L.

and

and

more

K.1, K.2.1, K.2.2, and K.2.3

Condition K applies to one inoperable source range channel in MODE 3, 4, or 5 with the RTBs closed and the CRD System capable of rod withdrawal. With the unit in this condition, < P-6, the NIS source range performs the monitoring and protective Functions. With one of the source range channels inoperable, 48 hours is allowed to restore it to an OPERABLE status. If the channel cannot be returned to an OPERABLE status, 1 additional hour is allowed to open the RTBs. In addition to opening the RTBs, all operations involving positive reactivity additions must be suspended within the same hour. Suspension of positive reactivity additions and opening the RTBs will preclude any power excursion. Also, all valves that could add unborated water to the RCS must be closed within the same 1 hour as specified in LCO 3.9.2, "Unborated Water Source Isolation Valves." The isolation of unborated water sources will preclude a boron dilution accident. The allowance of 48 hours to restore the channel to OPERABLE status and the additional hour to open the RTBs are justified in Reference 7. The allowance of 1 hour to stop positive reactivity additions and close the unborated water source isolation valves is based on operating experience which provides sufficient time to accomplish the Required Actions in an orderly manner.

is not met

L.1, L.2, and L.3

~~is inoperable~~

when the required

Condition L applies ~~to no OPERABLE~~ Source Range Neutron Flux trip channels ~~when~~ in MODE 3, 4, or 5 with the RTBs open. With the unit in this condition, the NIS source range performs the monitoring and protective Functions. With ~~no~~ the required inoperable

(continued)

Once the RTBs are open, the core is in a more stable condition and the unit enters Condition L.

BASES

ACTIONS

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

inoperable source range channels ~~OPERABLE~~, operations involving positive reactivity additions that would reduce the SDM to less than the limits specified in LCO 3.1.1 and LCO 3.1.2 shall be suspended immediately. This will preclude any power escalation. In addition to suspension of positive reactivity additions, all valves that could add unborated water to the RCS must be closed within 1 hour as specified in LCO 3.9.2. The isolation of unborated water sources will preclude a boron dilution accident.

INSERT 1, Page B 3.3-46, See next page.

Also, the SDM must be verified within 1 hour and once every 12 hours thereafter as per SR 3.1.1.1, SDM verification. With no source range channels OPERABLE, core protection is severely reduced. Verifying the SDM within 1 hour allows sufficient time to perform the calculations and determine that the SDM requirements are met. The SDM must also be verified once per 12 hours thereafter to ensure that the core reactivity has not changed. Required Action L.1 precludes any positive reactivity additions; therefore, core reactivity should not be increasing, and a 12 hour Frequency is adequate. The Completion Times of within 1 hour and once per 12 hours are based on operating experience in performing the Required Actions and the knowledge that unit conditions will change slowly.

M.1 and M.2

Condition M applies to the following reactor trip Functions:

- Pressurizer Pressure–Low;
- Pressurizer Water Level–High;
- Reactor Coolant Flow–Low (Two Loops);
- ~~RCP Breaker Position (Two Loops);~~
- Undervoltage RCPs; and
- Underfrequency RCPs.

With one channel inoperable, the inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip

(continued)

INSERT 1
(PAGE B 3.3-46)

Required Action L.1 precludes operations involving systems that contain large volumes of water (i.e. CVCS, SIS, and RWST) at reduced boron concentrations with respect to the RCS that could dilute the boron concentrations of the RCS to less than that required to maintain the SDM requirements of LCO 3.1.1 and LCO 3.1.2. This Required Action does not preclude positive reactivity additions that cannot reduce the SDM to less than the limits specified in LCO 3.1.1 and LCO 3.1.2. The addition of water with a boron concentration greater than that required to maintain the reactor shutdown within the requirements of LCO 3.1.1 and LCO 3.1.2, but less than the RCS, is permitted. Positive reactivity additions such as small volume chemical additions and normal plant cooldowns are also permitted as long as the SDM limits specified in LCO 3.1.1 and LCO 3.1.2 are met.

BASES

ACTIONS

M.1 and M.2 (continued)

These Functions do

condition requiring only one additional channel to initiate a reactor trip above the P-7 setpoint and below the P-8 setpoint. ~~This Function does~~ not have to be OPERABLE below the P-7 setpoint. ~~because there are no loss of flow trips below the P-7 setpoint.~~ The 6 hours allowed to place the channel in the tripped condition is justified in Reference 7. An additional 6 hours is allowed to reduce THERMAL POWER to < P-7 if the inoperable channel cannot be restored to OPERABLE status or placed in trip within the specified Completion Time.

G

Allowance of this time interval takes into consideration the redundant capability provided by the remaining redundant OPERABLE channel and the low probability of occurrence of an event during this period that may require the protection afforded by the Functions associated with Condition M.

any single channel, including

The Required Actions have been modified by a Note, which allows ~~placing~~ the inoperable channel, ~~in the~~ bypassed ~~condition~~ for up to 4 hours while performing ~~routine~~ surveillance testing of the other channels. The 4 hour time limit is justified in Reference 7.

C

To be

N.1 and N.2

Condition N is applicable to the Reactor Coolant Flow—Low (Single Loop) reactor trip Function. With one channel inoperable, the inoperable channel must be placed in trip within 6 hours. If the channel cannot be restored to OPERABLE status or the channel placed in trip within the 6 hours, then THERMAL POWER must be reduced below the P-8 setpoint within the next 4 hours. This places the unit in a MODE where the LCO is no longer applicable. This trip Function does not have to be OPERABLE below the P-8 setpoint because other RTS trip Functions provide core protection below the P-8 setpoint. The 6 hours allowed to restore the channel to OPERABLE status or place in trip and the 4 additional hours allowed to reduce THERMAL POWER to below the P-8 setpoint are justified in Reference 7.

The Required Actions have been modified by a Note which allows ~~placing~~ the inoperable channel, ~~in the~~ bypassed ~~condition~~ for up to 4 hours while performing ~~routine~~

C

to be

(continued)

BASES

ACTIONS

N.1 and N.2 (continued)

surveillance testing of the other channels. The 4 hour time limit is justified in Reference 7.

~~O.1 and O.2~~

~~Condition O is applicable to the RCP Breaker Position (Single Loop) reactor trip Function. There is one breaker position device per RCP breaker. With one channel inoperable, the inoperable channel must be restored to OPERABLE status within 6 hours. If the channel cannot be restored to OPERABLE status within the 6 hours, then THERMAL POWER must be reduced below the P-8 setpoint within the next 4 hours. This places the unit in a MODE where the LCO is no longer applicable. This Function does not have to be OPERABLE below the P-8 setpoint because other RTS Functions provide core protection below the P-8 setpoint. The 6 hours allowed to restore the channel to OPERABLE status and the 4 additional hours allowed to reduce THERMAL POWER to below the P-8 setpoint are justified in Reference 7.~~

~~The Required Actions have been modified by a Note which allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 7.~~

O.1 and O.2

Condition O applies to Turbine Trip on Low Fluid Oil Pressure or on Turbine Stop Valve Closure. With one channel inoperable, the inoperable channel must be placed in the trip condition within 6 hours. If placed in the tripped condition, this results in a partial trip condition requiring only one additional channel to initiate a reactor trip. If the channel cannot be restored to OPERABLE status or placed in the trip condition, then power must be reduced below the P-9 setpoint within the next 4 hours. The 6 hours allowed to place the inoperable channel in the tripped condition and the 4 hours allowed for reducing power are justified in Reference 7.

(continued)

BASES

ACTIONS

O O
R.1 and R.2 (continued)

or one additional channel

The Required Actions have been modified by a Note which allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing. ~~of the other channels.~~ The 4 hour time limit is justified in Reference 7.

P P
Q.1 and Q.2

P
Condition Q applies to the SI Input from ESFAS reactor trip and the RTS Automatic Trip Logic in MODES 1 and 2. These actions address the train orientation of the RTS for these Functions. With one train inoperable, 6 hours are allowed to restore the train to OPERABLE status (Required Action Q.1) or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours (Required Action Q.1) is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function and given the low probability of an event during this interval. The Completion Time of 6 hours (Required Action Q.2) is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems.

The Required Actions have been modified by a Note that allows bypassing one train up to 4 hours for surveillance testing, provided the other train is OPERABLE.

Q Q
R.1 and R.2

Q
Condition R applies to the RTBs in MODES 1 and 2. These actions address the train orientation of the RTS for the RTBs. With one train inoperable, 1 hour is allowed to restore the train to OPERABLE status or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1 hour and 6 hour Completion Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS Function. Placing the unit in MODE 3 removes the requirement for this particular Function.

(continued)

BASES

ACTIONS

~~R~~Q.1 and ~~R~~Q.2 (continued)

The Required Actions have been modified by two Notes. Note 1 allows one channel to be bypassed for up to 2 hours for surveillance testing, provided the other channel is OPERABLE. Note 2 allows one RTB to be bypassed for up to 2 hours for maintenance on undervoltage or shunt trip mechanisms if the other RTB train is OPERABLE. The 2 hour time limit is justified in Reference 7.

~~S~~R.1 and ~~S~~R.2

Condition ~~S~~R applies to the P-6 and P-10 interlocks. With one channel inoperable for one out of two or two out of four coincidence logic, the associated interlock must be verified to be in its required state for the existing unit condition within 1 hour or the unit must be placed in MODE 3 within the next 6 hours. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1 hour and 6 hour Completion Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS Function.

~~T~~S.1 and ~~T~~S.2

Condition ~~T~~S applies to the P-7, P-8, P-9, and P-13 interlocks. With one channel inoperable for one out of two or two out of four coincidence logic, the associated interlock must be verified to be in its required state for the existing unit condition within 1 hour or the unit must be placed in MODE 2 within the next 6 hours. These actions are conservative for the case where power level is being raised. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power in an orderly manner and without challenging unit systems.

(continued)

BASES

ACTIONS (continued)

~~U~~T.1, ~~U~~T.2.1, and ~~U~~T.2.2

Condition ~~U~~T applies to the RTB Undervoltage and Shunt Trip Mechanisms, or diverse trip features, in MODES 1 and 2. With one of the diverse trip features inoperable, it must be restored to an OPERABLE status within 48 hours or the unit must be placed in a MODE where the requirement does not apply. This is accomplished by placing the unit in MODE 3 within the next 6 hours (54 hours total time) followed by opening the RTBs in 1 additional hour (55 hours total time). The Completion Time of 6 hours is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. With the RTBs open and the unit in MODE 3, this trip Function is no longer required to be OPERABLE. The affected RTB shall not be bypassed while one of the diverse features is inoperable except for the time required to perform maintenance to one of the diverse features. The allowable time for performing maintenance of the diverse features is 2 hours for the reasons stated under Condition ~~R~~C.

The Completion Time of 48 hours for Required Action ~~U~~T.1 is reasonable considering that in this Condition there is one remaining diverse feature for the affected RTB, and one OPERABLE RTB capable of performing the safety Function and given the low probability of an event occurring during this interval.

Insert D (C)

SURVEILLANCE REQUIREMENTS

The SRs for each RTS Function are identified by the SRs column of Table 3.3.1-1 for that Function.

A Note has been added to the SR Table stating that Table 3.3.1-1 determines which SRs apply to which RTS Functions.

Note that each channel of process protection supplies both trains of the RTS. When testing Channel I, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel II, Channel III, and Channel IV ~~(if applicable)~~. The CHANNEL CALIBRATION and ~~X~~COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies. (D)

(continued)

INSERT D

U.1.1, U.1.2, and U.2

Condition U applies to the Steam Generator Water Level--Low - Low reactor trip Function.

A known inoperable channel must be restored to OPERABLE status or placed in the tripped condition within 6 hours. Placing the channel in the tripped condition requiring only one out of two logic for actuation of the two out of three trips. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 7.

If a channel fails, it is placed in the tripped condition and does not affect the TTD setpoint calculations for the remaining OPERABLE channels. It is then necessary for the operator to force the use of the shorter TTD time delay by adjustment of the single steam generator time delay calculation ($T_s$) to match the multiple steam generator time delay calculation ($T_m$) for the affected protection set, through the Man Machine Interface.

If the inoperable channel cannot be restored or placed in the trip condition within the specified Completion Time, the unit must be placed in a MODE where these Functions are not required OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to place the unit in MODE 3 from MODE 1 from full power in an orderly manner and without challenging plant systems.

V.1 and V.2

Condition V applies to the Vessel $\Delta T$ Equivalent to Power reactor trip Function.

Failure of the vessel $\Delta T$ channel input (failure of more than one $T_H$ RTD or failure of both $T_C$ RTDs) does not affect the TTD calculation for a protection set. This results in the requirement that the operator adjust the threshold power level for zero seconds time delay from 50% RTP to 0% RTP, through the Man Machine Interface.

If the inoperable channel cannot be restored or placed in the trip condition within the specified Completion Time, the unit must be placed in a MODE where these Functions are not required to be OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to place the unit in MODE 3 from MODE 1 from full power in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~Reviewer Note: Certain Surveillance Frequencies are based on approval topical reports. In order for a licensee to use these times the licensee must justify the Surveillance Frequencies as required by the staff SER for the topical report.~~ (D)

(C)

The protection Functions associated with the EAGLE-21 Process Protection System have an installed bypass capability, and may be tested in either the trip or bypass mode, as approved in Reference 7.

TM (24)

SR 3.3.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or even something more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

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

The Frequency is based upon operating experience that demonstrates channel failure is rare. Thus, performance of the CHANNEL CHECK ensures that undetected overt channel failure is limited to 12 hours. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.1.2

SR 3.3.1.2 compares the calorimetric heat balance calculation to the NIS channel output every 24 hours. If the calorimetric exceeds the NIS channel output by > 2% RTP the NIS is declared inoperable, but must be adjusted. If the NIS channel output cannot be properly adjusted, the channel is declared inoperable.

not (5)

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2 (continued)

Three Notes modify SR 3.3.1.2. The first Note indicates that the NIS channel output shall be adjusted consistent with the calorimetric results if the absolute difference between the NIS channel output and the calorimetric is > 2%. The second Note clarifies that this Surveillance is required only if reactor power is ≥ 15% RTP and that 12 hours is allowed for performing the first Surveillance after reaching 15% RTP. At lower power levels, calorimetric data are inaccurate.

The Frequency of every 24 hours is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate the change in the absolute difference between NIS and heat balance calculated powers rarely exceeds 2% in any 24 hour period.

In addition, control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

SR 3.3.1.3 compares the incore system to the NIS channel output every 31 EFPD. If the absolute difference is ≥ 3% the NIS channel is still OPERABLE, but must be readjusted. If the NIS channel cannot be properly readjusted, the channel is declared inoperable.

Two ~~Three~~ Notes modify SR 3.3.1.3. The first Note indicates that the excore NIS channel shall be adjusted ~~recalibrated~~ if the absolute difference between the incore and excore AFD is ≥ 3%. The second Note clarifies that the Surveillance is required only if reactor power is ≥ [15%] RTP and that 96 ~~24~~ hours is allowed for performing the first Surveillance after reaching [15%] RTP.

28

This surveillance is typically performed at 50% RTP to ensure the results of the evaluation are more accurate and the adjustments more reliable. Seventy two hours are allowed [illegible] xenon stability and [illegible] for instrumentation alignments.

The Frequency of every 31 EFPD is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Also, the slow changes in neutron flux during the fuel cycle can be detected during this interval.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.4 (continued)

SR 3.3.1.4 is the performance of a TADOT every 31 days on a STAGGERED TEST BASIS. This test shall verify OPERABILITY by actuation of the end devices.

The RTB test shall include independent verification of the undervoltage and shunt trip mechanisms. The bypass breaker test shall include a local shunt trip. A Note has been added to indicate that this test must be performed on the bypass breaker prior to placing it in service.

The Frequency of every 31 days on a STAGGERED TEST BASIS is justified in Reference 7.

SR 3.3.1.5

SR 3.3.1.5 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested every 31 days on a STAGGERED TEST BASIS, using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection Function. The time allowed for the testing, 4 hours, and the Frequency of once every 31 days on a STAGGERED TEST BASIS are justified in Reference 7.

SR 3.3.1.6 is a calibration of (H) calibrated to agree with the incore detectors

SR 3.3.1.6 ~~compares~~ the excore channels to the incore channels. If the measurements do not agree the excore channels are not declared inoperable, but must be ~~adjusted.~~ e If the excore channels cannot be adjusted, the channels are declared inoperable.

Two ~~Three~~ Notes modify SR 3.3.1.6. The first Note indicates that the NIS channel outputs shall be ~~adjusted~~ calibrated (H) to agree with the incore channel measurements. The second Note clarifies that this Surveillance is required only if reactor power > 50% RTP and that ~~[24 hours]~~ up to 31 days (2) is allowed for performing the first surveillance after reaching 50% RTP.

The Frequency of 92 EFPD is justified in Reference 7. (H) Thirty-one days are allowed to ensure xenon stability and allow for instrumentation alignments.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.7 (continued)

SR 3.3.1.7 is the performance of a~~n~~ ~~A~~COT every [92] days.

A~~n~~ ~~A~~COT is performed on each required channel to ensure the entire channel will perform the intended Function. Setpoints must be within the Allowable Values specified in Table 3.3.1-1.

The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology. The setpoint shall be left set consistent with the assumptions of the current unit specific setpoint methodology.

The "as found" and "as left" values must also be recorded and reviewed for consistency with the assumptions of Reference 7.

The Frequency of [92] days is justified in Reference 7.

SR 3.3.1.8

SR 3.3.1.8 is the performance of a~~n~~ ~~A~~COT as described in SR 3.3.1.7, except it is modified by a Note that this test shall include verification that the P-6 and P-10 interlocks are in their required state for the existing unit condition. This test ensures that the NIS source range and intermediate range channels are OPERABLE prior to taking the reactor critical.

SR 3.3.1.9

Replace with Insert SR 3.3.1.9

3

SR 3.3.1.9 ~~is~~ the performance of a TADOT ~~as described in SR 3.3.1.4, except that the test~~ and is performed every [92] days, ~~instead of every 31 days and is~~ as justified in Reference 7.

SR 3.3.1.10

A CHANNEL CALIBRATION is performed every [18] months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the of the instrument loop, including the

(continued)

INSERT SR 3.3.1.9

SR 3.3.1.9 is the performance of a TADOT every 92 days. This test is a check of RTS Function 11, Undervoltage RCPs and 12, Underfrequency RCPs. These Functions are tested up to, and including, the master transfer relay coils.

This test does not require verification of relay setpoints. Setpoint verification requires removal of the relays from service for extended periods of time, thereby jeopardizing electrical equipment protection for that period of time.

These devices are reliable components and their setpoints are verified during a CHANNEL CALIBRATION performed every 18 months. The Frequency is justified in Reference 7.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. Since this SR applies to RCP undervoltage and underfrequency relays setpoint verification is accomplished during the CHANNEL CALIBRATION.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.10 (continued)

sensor. The test verifies that the channel responds to measured parameter within the necessary range and accuracy.

D

Watts Bar

CHANNEL CALIBRATION measurement and setpoint error determination and readjustment must be performed consistent with the assumptions of the ~~unit specific~~ setpoint methodology. The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology.

The Surveillance Frequency of 18 months is based on the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.

SR 3.3.1.10 is modified by a Note stating that this test shall include verification that the time constants are adjusted to the prescribed values where applicable.

6 SR 3.3.1.11

as described in SR 3.3.1.10,

SR 3.3.1.11 is the performance of a CHANNEL CALIBRATION, every ~~[18]~~ 18 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the power range ~~neutron~~ channels detectors consists of a normalization of the ~~detectors~~ based on a power calorimetric and flux map performed above 15% RTP. The CHANNEL CALIBRATION for the source range and intermediate range ~~neutron~~ detectors consists of verifying equipment outputs to known electrical inputs ~~obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data~~. This Surveillance is not required for the NIS power range detectors for entry into MODE 2 or 1, and is not required for the NIS intermediate range detectors for entry into MODE 2, because the unit must be in at least MODE 2 to perform the test for the intermediate range detectors and MODE 1 for the power range detectors. The ~~[18]~~ 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 these

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.11 (continued)

components usually pass the Surveillance when performed on the [18] month Frequency.

~~SR 3.3.1.12~~

~~SR 3.3.1.12 is the performance of a CHANNEL CALIBRATION, as described in SR 3.3.1.10, every [18] months. This SR is modified by a Note stating that this test shall include verification of the RCS resistance temperature detector (RTD) bypass loop flow rate.~~

7

~~This test will verify the rate lag compensation for flow from the core to the RTDs.~~

~~The Frequency is justified by the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.~~

SR 3.3.1.~~13~~ 12

SR 3.3.1.13 is the performance of an ACOT of RTS interlocks every [18] months.

The Frequency is based upon the known reliability of the interlocks and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.1.~~14~~ 13 , RCP breaker position, H

SR 3.3.1.~~14~~ 13 is the performance of a TADOT of the Manual Reactor Trip and the SI input from ESFAS. This TADOT is as described in SR 3.3.1.4, except that the test is performed every [18] months.

H

The Frequency is based upon the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludeds verification of setpoints from the TADOT. The functions affected have no setpoints associated with them.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.~~15~~ 14

SR 3.3.1.~~15~~ 14 is the performance of a TADOT of Turbine Trip Functions. This TADOT is as described in SR 3.3.1.4, except that this test is performed prior to reactor startup. A Note states that this Surveillance is not required if it has been performed within the previous 31 days. Verification of the Trip Setpoint does not have to be performed for this Surveillance. Performance of this test will ensure that the turbine trip Function is OPERABLE prior to taking the reactor critical. This test cannot be performed with the reactor at power and must therefore be performed prior to reactor startup.

SR 3.3.1.~~16~~ 15

SR 3.3.1.~~16~~ 15 verifies that the individual channel/train actuation response times are less than or equal to the maximum values assumed in the accident analysis. Response Time testing acceptance criteria are included in Reference 8. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the equipment reaches the required functional state (i.e., control and shutdown rods fully inserted in the reactor core).

For channels that include dynamic transfer Functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer Function set to one, with the resulting measured response time compared to the appropriate FSAR response time. Alternately, the response time test can be performed with the time constants set to their nominal value, provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

As appropriate, each channel's response must be verified every [18] months on a STAGGERED TEST BASIS. Testing of the final actuation devices is included in the testing. Response times cannot be determined during unit operation because equipment operation is required to measure response

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.16 (continued)

times. 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.

SR 3.3.1.~~16~~ 15 is modified by a Note stating that neutron detectors may be excluded from RTS RESPONSE TIME testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

REFERENCES

1. FSAR, Chapter [7].
2. FSAR, Chapter [6].
3. FSAR, Chapter [15].
4. IEEE-279-1971, April 5, 1972.
5. 10 CFR 50.49.
6. RTS/ESFAS Setpoint Methodology Study.
7. WCAP-10271-P-A, May 1986.
8. Technical Requirements Manual, Section 15, "Response Times."

25 (see Reference Insert, next page)

## REFERENCES INSERT


1. Watts Bar FSAR, Section 6.0, "Engineered Safety Features."

2. Watts Bar FSAR, Section 7.0, "Instrumentation and Controls."

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

4. Institute of Electrical and Electronic Engineers, IEEE-279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," April 5, 1972.

5. 10 CFR Part 50.49, "Environmental Qualifications of Electric Equipment Important to Safety for Nuclear Power Plants."

6. WCAP-12096, Rev. 4, "Westinghouse Setpoint Methodology for Protection System, Watts Bar 1 and 2," November 1990.

7. WCAP-10271-P-A, Supplement 1, Rev. 1 and Supplement 2, Rev. 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," May 1986 and June 1990.

8. Watts Bar Technical Requirement Manual, Section 3.3.1, Reactor Trip System Response Times.

# JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.3.1

## Generic Justifications

A. Modified to reflect WBN plant specific design.

B. Modified to reflect WBN plant specific terminology.

C. Modified to reflect changes due to the EAGLE-21 Microprocessor Based Process Protection System used at WBN.

D. Deletion or modification of information which is generic in nature and has been modified to be plant specific to WBN.

E. Bases revised to reflect on changes made to its LCO.

F. Editorial (i.e., spelling, punctuation, capitalization, etc.) needing no further explanation.

G. Consistency of presentation. i.e.:

- Only Required and Not Required Note corrections
- "(s)" issues
- Article deletion
- Identical requirement wording
- Vendor preferred wording
- Plant specific terminology

H. This change reflects comments made by the industry to the NRC at the RSTS P & R review meeting held in Irvine, CA July 13 - 20, 1992.

## Specific Justifications

1. The Reactor Coolant Pump trip due to breaker position does not exist at WBN.

2. For SR 3.3.1.6, the time period shown in Note 2 has been changed from [24 hours] to 31 days. This requirement is based upon the assumption that SR 3.3.1.6 is being performed based on the previous cycle. During a reload startup, initial alignments are performed based on the previous cycle. During the initial startup, a preliminary calibration is performed before exceeding 50% power, and the formal calibration at 75%.

JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.3.1 (continued)

3. WBN Technical Specifications exclude the performance of monthly and quarterly setpoint verifications during trip actuating device operational tests for reactor trip. Setpoint verification requires that the relays be removed from service for extended periods of time and, thereby, jeopardize electrical equipment protection for that period of time.

4. The sentence was deleted because it unnecessarily repeated information in the paragraph above.

5. This item was a correction to the text.

6. This paragraph was modified to more accurately reflect what is performed.

7. Modified to reflect the fact that WBN no longer has the resistance temperature detector bypass loops.

8. Added for clarification.

9. Modified to reflect the fact that WBN does not have a Boron Dilution Protection System.

10. Modified to reflect the WBN preferred wording.

11. Required Action J.1 was deleted. Once the RTBs are opened, Condition J no longer applies. Required Action J.2 opens the RTBs which places the plant immediately in Condition L. The Required Action J.1 to suspend positive reactivity additions is redundant to the same action in Condition L. All that is required in Condition J is to open the RTBs immediately, then Condition L is applicable.

    The same situation exists for Condition K and Required Actions K.2.2 and K.2.3. Once the RTBs are opened, Condition L applies and Required Actions K.2.2 and K.2.3 are repeated in Condition L.

12. A NOTE has been added to Required Actions to allow surveillance testing in bypass for four hours for automatic trip logic. SSPS can only be testing when bypassed.

13. The NOTE above the Required Actions for Condition D has been deleted because Watts Bar does not have bypass capability for NIS Channels.

14. Specific titles have been added to Conditions N, U, and V because each Condition only applies to one function. This addition makes Conditions N, U, and V consistent with other "one function" Conditions.

15. "Turbine Trip (Low Fluid Oil Pressure)" has been added to the NOTE to Required Actions 0.1 & 0.2 because this Turbine Trip Function can be tested in bypass for Watts Bar.

16. SR 3.3.1.12 has been deleted because the CHANNEL CALIBRATION (SR 3.3.1.11) performs all requirements of the COT (SR 3.3.1.12).

    Since SR 3.3.1.12 only applied to Function 16 it has been deleted from the SR table. The SR table and the function table have been renumbered accordingly.

    This is an error in the standard.

17. The Condition for Automatic Trip Logic, MODES 1 & 2, has been changed from R to P because Condition R applies to interlocks. The 1985 Watts Bar draft and the WOG comments on NUREG-1431 (WCAP-13029) indicate that the Required Actions for Condition P are correct.

    Condition P had previously been changed to only apply to "SI input from ESFAS." This specific function wording has been removed from Condition P.

18. Changes have been made to B 3.3-2, B 3.3-3, B.3.3-4, B 3.3-5, and B 3.3-38 to clarify the technical definition of the Reactor Protection System.

19. The number of field transmitters or sensors on page B 3.3-2 has been changed from "four" to "five." Watts Bar uses five RTDs to get RCS temperature for $\Delta T$.

20. The title of Reference 6 on Page B 3.3-4 has been changed to agree with Reference 6.

21. The CHANNEL CALIBRATION accuracy (Page B 3.3-4) has been deleted because, for Eagle-21 equipment, the rack calibration accuracy defines the comparator setting accuracy.

22. The words "in this range" have been added to make this wording consistent with Function 2.b. (last paragraph, Page B 3.3-10).

23. Editorial changes have been made to the Bases description of Condition V. RAD and RAP should be replaced by RTD and RTP, respectively.

    The statement has also been changed to read "failure of both to RTDs." The $\Delta T$ channels are designed to operate with one $T_c$ RTD. Since there are two $T_c$ RTD per loop, both $T_c$ RTDs must fail to affect the $\Delta T$ channel.

24. "TM" has been added to "EAGLE-21" because "EAGLE-21" is a Westinghouse trademark.

25. Change to specify WBN specific reference information and format.

26. The Completion Time for reducing power to ≤ 75% was changed from 4 hours to 12 hours. The 12-hour Completion Time is based on the Frequency for the QPTR Surveillance with the alarm or one power range channel inoperable. Since the reason for reducing power is based on QPTR concerns, and with one power range channel or the QPTR alarm inoperable, 12 hours between QPTR verifications is considered adequate to ensure the stability of the core. It is also reasonable to allow 12 hours to reduce power to ≤ 75% which will prevent radial power distributions beyond the design limits. Required Actions D.1.1 and D.1.2 are reversed to order the Completion Time from the shortest to the longest. Bases changes also made to reflect new completion time.

27. This change corrects an error to the standard. Offsite dose criteria are in 10 CFR 20 & 100.

28. The bases for the 96 hours is as follows:

| | |
|---|---|
| 4 hours | Allowance for power escalation maneuvers from 15 to 30% RTP. |
| 40 hours | Allowance for xenon to reach equilibrium conditions, |
| 4 hours | Allowance for obtaining incore flux mapping data. |
| 3 hours | Allowance for transferring data from the plant process computer to the Prime computer, running the INCORE code, printing the results and analyzing the results to ensure the flux map is a valid map. |
| 2 hours | Allowance for completing the Surveillance Instruction (SI). |
| 1 hour | Allowance for review of the SI performance package. |
| 1 hour | Allowance for issuance of the recalibration data calculation procedure. |
| 1 hour | Allowance for completing the recalibration data calculation procedure. |
| 1 hour | Allowance for review of the performance of the recalibration data calculation procedure. |
| 2 hours | Allowance for notification of Instrument Maintenance and issuance of the work order/procedure for recalibration. |
| 32 hours | Allowance to performing the recalibration on all four (4) NIS channels. |
| 5 hours | Combined float for all of the above activities. |

Experience gained from Sequoyah Nuclear Plant's operation has been factored in the above estimate.

29. Since Watts Bar has only one source range channel required OPERABLE in MODES 3, 4, or 5 with the RTBs open, Condition L has been changed to reflect this.

30. Statements were deleted since they are not always true. Appropriate wording was added to replace.

## 3.3 INSTRUMENTATION

### 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

LCO 3.3.2 The ESFAS instrumentation for each Function in Table 3.3.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.2-1.

ACTIONS

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

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more Functions with one or more required channels inoperable. | A.1 Enter the Condition referenced in Table 3.3.2-1 for the channel(s). | Immediately |
| B. One channel or train inoperable. | B.1 Restore (1) Channel or train to OPERABLE status. | 48 hours |
| | OR | |
| | B.2.1 Be in MODE 3. | 54 hours |
| | AND | |
| | B.2.2 Be in MODE 5. | 84 hours |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| C. One train inoperable. | ----------NOTE---------- One train may be bypassed for up to [4] hours for surveillance testing provided the other train is OPERABLE. ---------- | |
| | C.1 Restore train to OPERABLE status. | 6 hours |
| | OR | |
| | C.2.1 Be in MODE 3. | 12 hours |
| | AND | |
| | C.2.2 Be in MODE 5. | 42 hours |
| D. One channel inoperable. C any single | ----------NOTE---------- ~~The inoperable~~ channel may be bypassed for up to 40 hours for surveillance testing ~~of other channels~~. ---------- | |
| | D.1 Place channel in trip. | 6 hours |
| | OR | |
| | D.2.1 Be in MODE 3. | 12 hours |
| | AND | |
| | D.2.2 Be in MODE 4. | 18 hours |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| E. One Containment Pressure channel inoperable. | ----------NOTE---------- One additional channel may be bypassed for up to 4 hours for surveillance testing. ---------- | |
| | E.1 Place channel in bypass. | 6 hours |
| | OR | |
| | E.2.1 Be in MODE 3. | 12 hours |
| | AND | |
| | E.2.2 Be in MODE 4. | 18 hours |
| F. One channel or train inoperable. | F.1 Restore channel or train to OPERABLE status. | 48 hours |
| | OR | |
| | F.2.1 Be in MODE 3. | 54 hours |
| | AND | |
| | F.2.2 Be in MODE 4. | 60 hours |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| G. One train inoperable. | ----------NOTE---------- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ---------- | |
| | G.1 Restore train to OPERABLE status. | 6 hours |
| | OR | |
| | G.2.1 Be in MODE 3. | 12 hours |
| | AND | |
| | G.2.2 Be in MODE 4. | 18 hours |
| H. One train inoperable. | ----------NOTE---------- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ---------- | |
| | H.1 Restore train to OPERABLE status. | 6 hours |
| | OR | |
| | H.2 Be in MODE 3. | 12 hours |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| I. One channel inoperable. any single (c) Steam Generator Water Level -- High High (6) | ----------NOTE---------- ~~The inoperable~~ channel may be bypassed for up to 24 hours for surveillance testing. ~~of other channels~~ ---------- | |
| | I.1 Place channel in trip. | 6 hours |
| | OR | |
| | I.2 Be in MODE 3. | 12 hours |
| J. One Main Feedwater Pump Trip channel inoperable. | J.1 Restore channel to OPERABLE status. | 48 hours |
| | OR | |
| | J.2 Be in MODE 3. | 54 hours |
| K. One channel inoperable. | ----------NOTE---------- One additional channel may be bypassed for up to 24 hours for surveillance testing. ---------- | |
| | K.1 Place channel in bypass. | 6 hours |
| | OR | |
| | K.2.1 Be in MODE 3. | 12 hours |
| | AND | |
| | K.2.2 Be in MODE 5. | 42 hours |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| P-11 INTERLOCK<br>L. One channel inoperable. (1) | L.1 Verify interlock is in required state for existing unit condition. | 1 hour |
| | OR | |
| | L.2.1 Be in MODE 3. | 7 hours |
| | AND | |
| | L.2.2 Be in MODE 4. | 13 hours |
| M. One P-14 Interlock channel inoperable. | M.1 Verify interlock is in required state for existing unit condition. | 1 hour |
| | OR | |
| | M.2 Be in MODE 3. | 7 hours |

Insert Conditions N and O.
(NEXT PG.)

(C)

INSERT
Conditions N and O

| | | | | |
|---|---|---|---|---|
| | | ----------NOTE---------- Any single channel may be bypassed for up to 4 hours for surveillance testing. ---------- | | |
| N. | One Steam Generator Water Level--Low - Low channel inoperable. | N.1.1 | Place channel in trip. | 6 hours |
| | | AND | | |
| | | N.1.2 | For the affected protection set, set the Trip Time Delay ($T_s$) to match the Trip Time Delay ($T_m$). | 4 hours |
| | | OR | | |
| | | N.2.1 | Be in MODE 3. | 12 hours |
| | | AND | | |
| | | N.2.2 | Be in MODE 4. | 18 hours |
| O. | One Vessel ΔT channel inoperable. | O.1 | Set the Trip Time Delay threshold power level for ($T_s$) and ($T_m$) to 0% power. | 6 hours |
| | | OR | | |
| | | O.2 | Be in MODE 3. | 12 hours |

SURVEILLANCE REQUIREMENTS

AS 1

----------NOTE----------
Refer to Table 3.3.2-1 to determine which SRs apply for each ESF Function.
---

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.3.2.1 | Perform CHANNEL CHECK. | 12 hours |
| SR 3.3.2.2 | Perform ACTUATION LOGIC TEST. | 31 days on a STAGGERED TEST BASIS |
| SR 3.3.2.3 | ----------NOTE---------- The continuity check may be excluded. ---------- Perform ACTUATION LOGIC TEST. | 31 days on a STAGGERED TEST BASIS |
| SR 3.3.2.~~4~~ 3 | Perform MASTER RELAY TEST. | 31 days on a STAGGERED TEST BASIS |
| SR 3.3.2.~~5~~ 4 | Perform ~~A~~COT. 5 | 92 days |
| SR 3.3.2.~~6~~ 5 | INSERT 3.3-7a 14 Perform SLAVE RELAY TEST. | 92 days |
| SR 3.3.2.~~7~~ 6 | 10 ----------NOTE---------- Verification of relay setpoints not required. ---------- Perform TADOT. | 92 days |

INSERT 3.3-7b 14

(continued)

INSERT 3.3-7a

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

Slave relays tested by SR 3.3.2.7 are excluded from this surveillance.

---

INSERT 3.3-7b

| | | |
|---|---|---|
| SR 3.3.2.7 | Perform SLAVE RELAY TEST on slave relays K603A, K603B, K604A, K604B, K609A, K609B, K625A, and K625B. | 18 months |

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.3.2.~~7~~ 8 (14) | Perform TADOT. | 9 (18) 9 months |
| SR 3.3.2.~~8~~ 9 (14) | ----------NOTE----------<br>This Surveillance shall include verification that the time constants are adjusted to the prescribed values.<br>----------<br><br>Perform CHANNEL CALIBRATION. | [18] months |
| SR 3.3.2.~~9~~ 10 (14) | ----------NOTE----------<br>Not required to be performed for the turbine driven AFW pump until (24) hours after SG pressure is ≥ [~~1000~~] 1092 psig.<br>----------<br><br>Verify ESFAS RESPONSE TIMES are within limit. | [18] months on a STAGGERED TEST BASIS |
| SR 3.3.2.~~10~~ 11 (14) | Perform TADOT. | Once per Reactor Trip Breaker cycle |

A

Table 3.3.2-1 (page 1 of 8)
Engineered Safety Feature Actuation System Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS | CONDITIONS | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE | TRIP SETPOINT |
|---|---|---|---|---|---|---|
| 1. Safety Injection | | | | | | |
| a. Manual Initiation | 1,2,3,4 | 2 | B | SR 3.3.2.~~8~~ 7,8 (14) | N/A | N/A |
| b. Automatic Actuation Logic and Actuation Relays | 1,2,3,4 | 2 trains | C | SR 3.3.2.2<br>SR 3.3.2.~~4~~ 3<br>SR 3.3.2.~~8~~ 5 ← SR 3.3.2.7<br>SR 3.3.2.~~10~~ 9,10 (14) | N/A | N/A |
| c. Containment Pressure – High | 1,2,3 | 3 | D | SR 3.3.2.1<br>SR 3.3.2.~~5~~ 4<br>SR 3.3.2.~~8~~ ~~9~~ 9<br>SR 3.3.2.~~10~~ 9,10 | ≤ ~~[3.86]~~ 1.6 ~~psig~~ PSID | ≤ ~~[3.6]~~ 1.54 ~~psig~~ PSID |
| d. Pressurizer Pressure – Low | 1,2,3(a) | ~~3~~ | D | SR 3.3.2.1 (14)<br>SR 3.3.2.~~5~~ 4<br>SR 3.3.2.~~9~~ ~~8~~ 9<br>SR 3.3.2.~~10~~ 9,10 | ≥ ~~[1839]~~ 1864.8 psig | ≥ ~~[1850]~~ 1870 psig |
| e. Steam Line Pressure ~~(b)~~ Low | 1,2,3 ~~[(b)]~~ a | 3 per steam line | D | SR 3.3.2.1<br>SR 3.3.2.~~5~~ 4<br>SR 3.3.2.~~9~~ ~~8~~ 9<br>SR 3.3.2.~~10~~ 9,10 (14) | ≤ ~~[635]~~ (c) 666.6 psig b | ≥ ~~[675]~~ (c) 675 psig b |
| (2) High Differential Pressure Between Steam Lines | 1,2,3 | 3 per steam line | D | SR 3.3.2.1<br>SR 3.3.2.~~5~~ 4<br>SR 3.3.2.~~9~~ 8<br>SR 3.3.2.~~10~~ 9 | ≤ [106] psig | ≤ [97] psig |
| f. High Steam Flow in Two Steam Lines | 1,2,3(d) | 2 per steam line | D | SR 3.3.2.1<br>SR 3.3.2.~~5~~ 4<br>SR 3.3.2.~~9~~ 8<br>SR 3.3.2.~~10~~ 9 | (e) | (f) |
| Coincident with $T_{avg}$ – Low Low | 1,2,3(d) | 1 per loop | D | SR 3.3.2.1<br>SR 3.3.2.~~5~~ 4<br>SR 3.3.2.~~9~~ 8<br>SR 3.3.2.~~10~~ 9 | ≥ [550.6] °F | ≥ [553] °F |

(continued)

~~(a) Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~
(a) ~~(b)~~ Above the P-11 (Pressurizer Pressure) interlock.
(b) ~~(c)~~ Time constants used in the lead/lag controller are $t_1 \geq$ (50) seconds and $t_2 \leq$ (5) seconds.
~~(c) (d) Above the P-12 ($T_{avg}$ – Low Low) interlock.~~



~~INSERT NOTE (e) and (f) or move function f to next page~~

Table 3.3.2-1 (page 2 of 8)
Engineered Safety Feature Actuation System Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS | CONDITIONS | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE | TRIP SETPOINT ~~(a)~~ |
|---|---|---|---|---|---|---|
| 1. Safety Injection (continued) | | | | | | |
| g. High Steam Flow in Two Steam Lines | 1,2,3(d) | 2 per steam line | D | SR 3.3.2.1<br>SR 3.3.2.5<br>SR 3.3.2.9<br>SR 3.3.2.10 | (e) | (f) |
| Coincident with Steam Line Pressure – Low | 1,2,3(d) | 1 per steam line | D | SR 3.3.2.1<br>SR 3.3.2.5<br>SR 3.3.2.9<br>SR 3.3.2.10 | ≥ [635](c) psig | ≥ [675] psig |
| 2. Containment Spray | | | | | | |
| a. Manual Initiation | 1,2,3,4 | 2 per train, 2 trains | B | SR 3.3.2.~~8~~ 7 8 | N/A | N/A |
| b. Automatic Actuation Logic and Actuation Relays | 1,2,3,4 | 2 trains | C | SR 3.3.2.2<br>SR 3.3.2.~~4~~ 3<br>SR 3.3.2.~~6~~ 5<br>SR 3.3.2.~~10~~ ~~9~~ 10 | N/A | N/A |
| c. Containment Pressure | | | | | | |
| ~~High-3~~ High High | 1,2,3 | 4 | E | SR 3.3.2.1<br>SR 3.3.2.~~5~~ 4<br>SR 3.3.2.~~9~~ ~~8~~ 9<br>SR 3.3.2.~~10~~ ~~9~~ 10 | ≤ ~~[12.31]~~ 2.9 psig | ≤ ~~[12.05]~~ 2.81 psig |
| High – 3 (Two Loop Plants) | 1,2,3 | [3] sets of [2] | E | SR 3.3.2.1<br>SR 3.3.2.5<br>SR 3.3.2.9<br>SR 3.3.2.10 | ≤ [12.31] psig | ≤ [12.05] psig |

(continued)

~~(a) Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~
~~(c) Time constants used in the lead/lag controller are $t_1 \geq [50]$ seconds and $t_2 \leq [5]$ seconds.~~
~~(d) Above the P-12 ($T_{avg}$ – Low Low) interlock.~~
~~(e) Less than or equal to a function defined as ΔP corresponding to [44]% full steam flow below [20]% load, ~~an~~ and ΔP increasing linearly from [44]% full steam flow at [20]% load to [114]% full steam flow at [100]% load, and ΔP corresponding to [114]% full steam flow above 100% load.~~
~~(f) Less than or equal to a function defined as ΔP corresponding to [40]% full steam flow between [0]% and [20]% load and then a ΔP increasing linearly from [40]% steam flow at [20]% load to [110]% full steam flow at [100]% load.~~

Table 3.3.2-1 (page 3 of 8)
Engineered Safety Feature Actuation System Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS | CONDITIONS | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE | TRIP SETPOINT |
|---|---|---|---|---|---|---|
| 3. Containment Isolation | | | | 2 | | |
| a. Phase A Isolation | | | | | | |
| (1) Manual Initiation | 1,2,3,4 | 2 | B | SR 3.3.2.~~8~~ ~~7~~ 8 | N/A | N/A |
| (2) Automatic Actuation Logic and Actuation Relays | 1,2,3,4 | 2 trains | C | SR 3.3.2.2<br>SR 3.3.2.~~4~~ 3<br>SR 3.3.2.~~6~~ ~~5~~<br>SR 3.3.2.~~10~~ ~~11~~ 12 | N/A | N/A |
| (3) Safety Injection | Refer to Function 1 (Safety Injection) for all initiation functions and requirements. | | | | | |
| b. Phase B Isolation | | | | | | |
| (1) Manual Initiation | 1,2,3,4 | 2 per train, 2 trains | B | SR 3.3.2.~~8~~ ~~7~~ 8 | N/A | N/A |
| (2) Automatic Actuation Logic and Actuation Relays | 1,2,3,4 | 2 trains | C | SR 3.3.2.2<br>SR 3.3.2.~~4~~ 3<br>SR 3.3.2.~~6~~ ~~5~~ SR 3.3.2.7<br>SR 3.3.2.~~10~~ ~~9~~ 10 14 | N/A | N/A |
| (3) Containment Pressure<br>High ~~3~~ 2<br>~~High High~~ | 1,2,3 | ~~2~~ 4 | E | SR 3.3.2.1<br>SR 3.3.2.~~5~~ 4<br>SR 3.3.2.~~9~~ ~~8~~ 9<br>SR 3.3.2.~~10~~ ~~9~~ 10 14 | ≤ [~~12.31~~] 2.9 psig | ≤ ~~[12.05]~~ 2.81 psig |
| 4. Steam Line Isolation | | | | | | |
| a. Manual Initiation | 1,2,3 | A 4 ~~2~~ | F | SR 3.3.2.~~8~~ ~~7~~ 8 | N/A | N/A |
| b. Automatic Actuation Logic and Actuation Relays | 1,2,3 | 2 trains | G | SR 3.3.2.2<br>SR 3.3.2.~~4~~ 3<br>SR 3.3.2.~~6~~ ~~5~~<br>SR 3.3.2.~~10~~ ~~9~~ 10 14 | N/A | N/A |

(continued)

D ~~(a) Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~

Table 3.3.2-1 (page 4 of 8)
Engineered Safety Feature Actuation System Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS | CONDITIONS | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE | TRIP SETPOINT |
|---|---|---|---|---|---|---|
| 4. Steam Line Isolation (continued) | | | | | | |
| B c. Containment Pressure – High 2 HIGH | 1,2,3 | 4 | E ~~D~~ | SR 3.3.2.1<br>SR 3.3.2.~~5~~ 4<br>SR 3.3.2.~~9~~ 9<br>SR 3.3.2.~~10~~ 10 (14) | ≤ ~~[6.61]~~ 3.1 psig | ≤ ~~[6.35]~~ 2.81 psig |
| d. Steam Line Pressure<br>(1) Low | 1,2,3 (a) | 1 ~~3~~ per steam line | D | SR 3.3.2.1<br>SR 3.3.2.~~5~~ 4<br>SR 3.3.2.~~9~~ 9<br>SR 3.3.2.~~10~~ 10 (14) | ≥ ~~[635]~~ 666.6 (b) (c) psig | ≥ ~~[675]~~ (b) (c) psig |
| C e. STEAMLINE PRESSURE<br>~~(2)~~ Negative Rate – High | 3 (a) | 3 per steam line | D | SR 3.3.2.1<br>SR 3.3.2.~~5~~ 4<br>SR 3.3.2.~~9~~ 9<br>SR 3.3.2.~~10~~ 10 (14) | ≤ ~~[121.6]~~ 107.8 (d) psi/~~sec~~ | ≤ ~~[110]~~ 110 (d) psi/~~sec~~ |
| e. High Steam Flow in Two Steam Lines | 1,2,3 | 2 per steam line | D | SR 3.3.2.1<br>SR 3.3.2.5<br>SR 3.3.2.9<br>SR 3.3.2.10 | (e) | (f) |
| Coincident with $T_{avg}$ – Low Low | 1,2,3(d) | 1 per loop | D | SR 3.3.2.1<br>SR 3.3.2.5<br>SR 3.3.2.9<br>SR 3.3.2.10 | ≥ [550.6] °F | ≥ [553] °F |
| C f. High Steam Flow in Two Steam Lines | 1,2,3 | 2 per steam | D | SR 3.3.2.1<br>SR 3.3.2.5<br>SR 3.3.2.9<br>SR 3.3.2.10 | (e) | (f) |
| Coincident with Steam Line Pressure – Low | 1,2,3 | 1 per steam line | D | SR 3.3.2.1<br>SR 3.3.2.5<br>SR 3.3.2.9<br>SR 3.3.2.10 | ≥ [635] (c) psig | ≥ [675] (c) psig |

(continued)

(c) FUNCTION AUTOMATICALLY BLOCKED ABOVE P-11 (PRESSURIZER PRESSURE INTERLOCK) SETPOINT AND MAY BE MANUALLY BLOCKED BELOW P-11 WHEN SAFETY INJECTION ON STEAM LINE PRESSURE LOW IS NOT BLOCKED.

A

Manually F

~~(a) Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~

(a) ~~(b)~~ Above the P-11 (Pressurizer Pressure) interlock.

(b) (c) Time constants used in the lead/lag controller are $t_1 \geq [50]$ seconds and $t_2 \leq [5]$ seconds.

D

~~(d) Above the P-12 ($T_{avg}$ – Low Low) interlock.~~

~~(e) Less than or equal to a function defined as ΔP corresponding to [44]% full steam flow below [20]% load, ΔP increasing linearly from [44]% full steam flow at [20]% load to [114]% full steam flow at [100]% load, and ΔP corresponding to [114]% full steam flow above 100% load.~~

~~(f) Less than or equal to a function defined as ΔP corresponding to [40]% full steam flow between [0]% and [20]%~~

(d) TIME CONSTANT UTILIZED IN THE RATE/LAG CONTROLLER IS ≤ 50 SECONDS.

~~INSERT (g) and (h) MOVE function F to next page.~~

move to previous page

load and then a ΔP increasing linearly from [40]% steam flow at [20]% load to [110]% full steam flow at [100]% load.

(g) Below the P-11 (Pressurizer Pressure) interlock.

(h) Time constant utilized in the rate/lag controller is ≤ [50] seconds.

Table 3.3.2-1 (page 5 of 8)
Engineered Safety Feature Actuation System Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS | CONDITIONS | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE | TRIP SETPOINT[a] |
|---|---|---|---|---|---|---|
| 4. Steam Line Isolation (continued) | | | | | | |
| g. High Steam Flow | 1,2,3 | 2 per steam line | D | SR 3.3.2.1<br>SR 3.3.2.5<br>SR 3.3.2.9<br>SR 3.3.2.10 | ≤ [25]% of full steam flow at no load steam pressure | ≤ [ ] full steam flow at no load steam pressure |
| Coincident with Safety Injection | Refer to Function 1 (Safety Injection) for all initiation functions and requirements. | | | | | |
| and | | | | | | |
| Coincident with $T_{avg}$ – Low Low | 1,2,3[d] | [2] per loop | D | SR 3.3.2.1<br>SR 3.3.2.5<br>SR 3.3.2.9<br>SR 3.3.2.10 | ≥ 550.6]°F | ≥ [553]°F |
| h. High High Steam Flow | 1,2,3 | 2 per steam line | D | SR 3.3.2.1<br>SR 3.3.2.5<br>SR 3.3.2.9<br>SR 3.3.2.10 | ≤ [130]% of full steam flow at full load steam pressure | ≤ [ ] of full steam flow at full load steam pressure |
| Coincident with Safety Injection | Refer to Function 1 (Safety Injection) for all initiation functions and requirements. | | | | | |

(continued)

(a) Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.
(d) Above the P-12 ($T_{avg}$ – Low Low) interlock.

C

Table 3.3.2-1 (page 6 of 8)
Engineered Safety Feature Actuation System Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS | CONDITIONS | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE | TRIP SETPOINT |
|---|---|---|---|---|---|---|
| 5. Turbine Trip and Feedwater Isolation | | | | | | |
| a. Automatic Actuation Logic and Actuation Relays | 1,2 | 2 trains | H | SR 3.3.2.2<br>SR 3.3.2.~~4~~ 3<br>SR 3.3.2.~~6~~ 5<br>SR 3.3.2.~~10~~ ~~9~~ 10 | N/A | N/A |
| A b. SG Water Level – High High (P-14) | 1,2 | ~~3~~ 3 per SG | I | SR 3.3.2.1<br>SR 3.3.2.~~5~~ 4<br>SR 3.3.2.~~9~~ ~~8~~ 9<br>SR 3.3.2.~~10~~ ~~9~~ 10 | ≤ ~~[82.2]~~ 83.1 % | ≤ [82.4]% |
| c. Safety Injection | Refer to Function 1 (Safety Injection) for all initiation functions and requirements. | | | | | |
| 6. Auxiliary Feedwater | | | | | | |
| a. Automatic Actuation Logic and Actuation Relays ~~(Solid State Protection System)~~ | 1,2,3 | 2 trains | G | SR 3.3.2.2<br>SR 3.3.2.~~4~~ 3<br>SR 3.3.2.~~6~~ 5<br>SR 3.3.2.~~10~~ ~~9~~ 10 | N/A | N/A |
| A ~~b. Automatic Actuation Logic and Actuation Relays (Balance of Plant ESFAS)~~ | ~~1,2,3~~ | ~~2 trains~~ | ~~G~~ | ~~SR 3.3.2.3~~<br>~~SR 3.3.2.10~~ | ~~N/A~~ | ~~N/A~~ |
| b. SG Water Level – Low Low | 1,2,3 | ~~3~~ 3 per SG | ~~G~~ N | SR 3.3.2.1<br>SR 3.3.2.~~5~~ 4<br>SR 3.3.2.~~9~~ ~~8~~ 9<br>SR 3.3.2.~~10~~ ~~9~~ 10 | ≥ ~~[30.4]~~ 15.4 % | ≥ ~~[32.2]~~ 17.0 % |

INSERT → B C A

(continued)

D ~~(a) Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~

# INSERT B

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS | CONDITIONS | SURVEILLANCE REQUIREMENTS | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|---|
| Coincident with: | | | | | | |
| 1. Vessel $\Delta T$ equivalent to power $\leq$ 50% RTP | 1,2 | 4 (1/loop) | 0 | SR 3.3.2.1<br>SR 3.3.2.4<br>SR 3.3.2.9<br>SR 3.3.2.10 | Vessel $\Delta T$ variables input $\leq$ 50% RTP | Vessel $\Delta T$ variable input $\leq$ 51.8% RTP |
| With a time delay ($T_s$) if one SG is affected | | | | | $\leq T_s$ (Note 3, page 3.3-21) | $\leq$ [ ] $T_s$ (Note 3, page 3.3-21) |
| or | | | | | | |
| With a time delay ($T_m$) if two or more SGs are affected | | | | | $\leq T_m$ (Note 3, page 3.3-21) | $\leq$ [ ] $T_m$ (Note 3, page 3.3-21) |
| or | | | | | | |
| 2. Vessel $\Delta T$ equivalent to power > 50% RTP | 1,2 | 4 (1/loop) | 0 | SR 3.3.2.1<br>SR 3.3.2.4<br>SR 3.3.2.9 | Vessel $\Delta T$ variable input $\leq$ 50% RTP | Vessel $\Delta T$ variable input $\leq$ 51.8% RTP |

Table 3.3.2-1 (page 7 of 8)
Engineered Safety Feature Actuation System Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS | CONDITIONS | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE | TRIP SETPOINT |
|---|---|---|---|---|---|---|
| 6. Auxiliary Feedwater (continued) | | | | | | |
| c. ~~d.~~ Safety Injection | Refer to Function 1 (Safety Injection) for all initiation functions and requirements. | | | | | |
| d. ~~e.~~ Loss of Offsite Power | 1,2,3 | 2 ~~[3]~~ per bus | F | SR 3.3.2.~~7~~ 6 (a)<br>SR 3.3.2.~~8~~ 9<br>SR 3.3.2.~~10~~ 10 | ≥ [2912] V ~~with ≤ 0.8 sec. time delay~~ | ~~≥ [2975] V with ≤ 0.8 sec. time delay~~ |
| ~~f. Undervoltage Reactor Coolant Pump~~ | ~~1,2~~ | ~~[3] per bus~~ | ~~I~~ | ~~SR 3.3.2.7~~<br>~~SR 3.3.2.9~~<br>~~SR 3.3.2.10~~ | ~~≥ [69]% bus voltage~~ | ~~≥ [70]% bus voltage~~ |
| e. ~~g.~~ Trip of all Main Feedwater Pumps | 1,2 | ~~[2]~~ per pump | J | SR 3.3.2.~~8~~ 7<br>SR 3.3.2.9<br>~~SR 3.3.2.10~~ | ≥ [ ] psig | ≥ [ ] psig |
| f. ~~h.~~ MOTOR-DRIVEN ~~Auxiliary~~ AFW ~~Feedwater~~ Pump Suction Transfer on Suction Pressure – Low | 1,2,3 | ~~[2]~~ 3 | F | ~~SR 3.3.2.1~~ 6<br>SR 3.3.2.~~7~~ 7<br>SR 3.3.2.~~9~~ 8 9<br>~~SR 3.3.2.10~~<br>~~SR 3.3.2.9e~~ | ≥ ~~[20.53]~~ [psia] | ≥ ~~[ ]~~ 2.15 psia |
| INSERT C → g. | | | | | | |
| 7. Automatic Switchover to Containment Sump | | | | | | |
| a. Automatic Actuation Logic and Actuation Relays | 1,2,3,4 | 2 trains | C | SR 3.3.2.2<br>SR 3.3.2.~~4~~ 3<br>SR 3.3.2.~~6~~ 5<br>SR 3.3.2.~~10~~ 10 | N/A | |
| b. Refueling Water Storage Tank (RWST) Level – Low Low | 1,2,3,4 | 4 | K | SR 3.3.2.1<br>SR 3.3.2.~~5~~ 4<br>SR 3.3.2.~~9~~ 8 9<br>SR 3.3.2.~~10~~ 9 10 | ≥ ~~[15]%~~ 126 INCHES and ≤ [ ]% FROM TANK BASE | ≥ ~~[ ]~~ 130 INCHES and ~~≤ [ ]%~~ FROM TANK BASE |
| Coincident with Safety Injection | Refer to Function 1 (Safety Injection) for all initiation functions and requirements. | | | | | |

(continued)

~~(a) Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~

(a) SETPOINT VERIFICATION NOT REQUIRED

INSERT C

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS | CONDITIONS | SURVEILLANCE REQUIREMENTS | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|---|
| g. Turbine-driven AFW Pump Transfer on Suction Pressure Low | 1,2,3 | 2/train<br>2 trains | F | SR 3.3.2.6<br>SR 3.3.2.9 | 13.1 psig | 12.1 psig |

Table 3.3.2-1 (page 8 of 8)
Engineered Safety Feature Actuation System Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS | CONDITIONS | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE | TRIP SETPOINT(a) |
|---|---|---|---|---|---|---|
| 7. Automatic Switchover to Containment Sump (continued) | | | | | | |
| ~~c. RWST Level – Low Low~~ | 1,2,3,4 | 4 | K | SR 3.3.2.1<br>SR 3.3.2.5<br>SR 3.3.2.9<br>SR 3.3.2.10 | ≥ [15]% | ≥ [18]% |
| Coincident with Safety Injection | Refer to Function 1 (Safety Injection) for all initiation functions and requirements. | | | | | |
| and | | | | | | |
| Coincident with Containment Sump Level – High | 1,2,3,4 | 4 | K | SR 3.3.2.1<br>SR 3.3.2.~~5~~4<br>SR 3.3.2.~~9~~8 9<br>SR 3.3.2.~~10~~9 10 | ≥ ~~[30]~~ 32.5 in. above el. ~~[ ] ft.~~ (1703) ft. | ≥ ~~[ ] in.~~ 30.0 in. above el. ~~[ ] ft.~~ 703 ft. |
| 8. ESFAS Interlocks | | | | | | |
| a. Reactor Trip, P-4 | 1,2,3 | 1 per train, 2 trains | F | SR 3.3.2.~~11~~ 10 11 | N/A | N/A |
| b. Pressurizer Pressure, P-11 | 1,2,3 | 3 | L | SR 3.3.2.1<br>SR 3.3.2.~~5~~4<br>SR 3.3.2.~~9~~8 9 | ≤ ~~[1996]~~ 1964.2 psig | ≤ ~~[ ] psig~~ 1970 psig |
| ~~c. $T_{avg}$ – Low Low, P-12~~ Hold space | ~~1,2,3~~ | ~~[1] per loop~~ | ~~L~~ | ~~SR 3.3.2.1<br>SR 3.3.2.5<br>SR 3.3.2.9~~ | ~~≥ [550.6]°F~~ | ~~≥ [553]°F~~ |
| ~~c.~~ SG Water Level – High High, P-14 | 1,2 | ~~[3]~~ 3 per SG | M | SR 3.3.2.1<br>SR 3.3.2.~~5~~4<br>SR 3.3.3.~~9~~8 9 14 | ≤ ~~[86.2]%~~ 33.1 | ≤ ~~82.4%~~ |

~~(a) Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~

Global ACOT to COT



## B 3.3 INSTRUMENTATION

### B 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

BASES

BACKGROUND

The ESFAS initiates necessary safety systems, based upon the values of selected unit parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary, and to mitigate accidents.

The ESFAS instrumentation is segmented into three distinct but interconnected modules as identified below:

7 and C

- Field transmitters or process sensors ~~and instrumentation~~: provide a measurable electronic signal based upon the physical characteristics of the parameter being measured; or contact actuation

- Signal processing equipment including ~~analog~~ process protection system, Nuclear Instrumentation System, field contacts, ~~and protection channel sets~~ provide signal conditioning, ~~bistable~~ setpoint comparison, process algorithm actuation, compatible electrical signal output to protection system devices, and control board/control room/miscellaneous indications; and

analog to digital conversion

(Digital Protection System)

- Solid State Protection System (SSPS) including input, logic, and output bays: initiates the proper unit shutdown or engineered safety feature (ESF) actuation in accordance with the defined logic and based upon the ~~bistable outputs~~ from the signal process control and protection system.

bistable, setpoint comparators, or contact outputs

CPU and

Field Transmitters and Sensors

A five

In order to meet the design demands for redundancy and reliability, more than one, and often as many as ~~four~~, field sensors or transmitters are used to measure unit parameters. In many cases field transmitters and sensors that input to the ESFAS are shared with the Reactor Trip System (RTS). In some cases, the same channels also provide control system inputs. To account for calibration tolerances and instrument drift, which are assumed to occur between calibrations, statistical allowances are provided in the

(continued)

BASES

BACKGROUND.

Field Transmitters and Sensors (continued)

Trip Setpoint and Allowable Values. The OPERABILITY of each transmitter or sensor can be evaluated when its "as found" calibration data are compared against its documented acceptance criteria.

(Digital Protection System)

Signal Processing Equipment analog to digital conversion

7

C

Generally, three or four channels of process control equipment are used for the signal processing of unit parameters measured by the field instruments. The process control equipment provides signal conditioning, comparable output signals for instruments located on the main control board, and comparison of measured input signals with setpoints established by safety analyses. These setpoints are defined in References 1, 2, and 3. If the measured value of a unit parameter exceeds the predetermined setpoint, an output from a bistable is forwarded to the SSPS for decision evaluation. Channel separation is maintained up to and through the input bays. However, not all unit parameters require four channels of sensor measurement and signal processing. Some unit parameters provide input only to the SSPS, while others provide input to the SSPS, main control board, unit computer, and one or more control systems.

, setpoint comparator or contact

~~CPU or~~ a

C

Generally, if a parameter is used only for input to the protection circuits, three channels with a two out of three logic are sufficient to provide the required reliability redundancy. If one channel fails in a direction that would not result in a partial Function trip, the Function is still OPERABLE with a two out of two logic. If one channel fails such that a partial Function trip occurs, a trip will not occur and the Function is still OPERABLE with a one out of two logic.

Generally, if a parameter is used for input to the SSPS and a control function, four channels with a two out of four logic are sufficient to provide the required reliability and redundancy. The circuit must be able to withstand both an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function

(continued)

BASES

BACKGROUND

Signal Processing Equipment (continued)

actuation. Again, a single failure will neither cause nor prevent the protection function actuation.

These requirements are described in IEEE-279-1971 (Ref. 4). The actual number of channels required for each unit parameter is specified in Reference 2.

Trip Setpoints and Allowable Values

The Trip Setpoints are the nominal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION accuracy (i.e., ± rack calibration + comparator setting accuracy).

The Trip Setpoints used in the bistables are based on the analytical limits stated in Reference 2. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors, for those ESFAS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 5), the Trip Setpoints and Allowable Values specified in Table 3.3.2-1 in the accompanying LCO are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the Trip Setpoints, including their explicit uncertainties, is provided in the "RTS/ESFAS Setpoint Methodology Study" (Ref. 6). The actual nominal Trip Setpoint entered into the bistable is normally still more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by an CHANNEL OPERATIONAL TEST (COT). One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.

Setpoints in accordance with the Allowable Value ensure that the consequences of Design Basis Accidents (DBAs) will be acceptable, providing the unit is operated from within the LCOs at the onset of the DBA and the equipment functions as designed.

(continued)

BASES

BACKGROUND

Trip Setpoints and Allowable Values (continued)

Each channel can be tested on line to verify that the signal processing equipment and setpoint accuracy is within the specified allowance requirements of Reference 2. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated. SRs for the channels are specified in the SR section.

The Trip Setpoints and Allowable Values listed in Table 3.3.2-1 are based upon the methodology described in Reference 6, which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each Trip Setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

Solid State Protection System

The SSPS equipment is used for the decision logic processing of signal outputs from the signal processing equipment, ~~bistables~~. To meet the redundancy requirements, two trains of SSPS, each performing the same functions, are provided. In the event that one train is taken out of service for maintenance or test purposes, the second train will provide ESF actuation for the unit. In the event that both trains are taken out of service or placed in test, a reactor trip will result. Each train is packaged in its own cabinet for physical and electrical separation to satisfy separation and independence requirements. The system has been designed to trip in the event of a loss of power, directing the unit to a safe shutdown condition.

The SSPS performs the decision logic for actuating ESF equipment actuation; generates the electrical output signals that initiate the required actuation; and provides the status, permissive, and annunciator output signals to the main control room of the unit.

C

The ~~bistable~~ outputs from the signal processing equipment are sensed by the SSPS equipment and combined into logic

(continued)

BASES

BACKGROUND

Solid State Protection System (continued)

matrices that represent combinations indicative of various transients. If a required logic matrix combination is completed, the system will send actuation signals via master and slave relays to those components whose aggregate Function best serves to alleviate the condition and restore the unit to a safe condition. Examples are given in the section on Applicable Safety Analyses.

Each train has a builtin testing device that can automatically test the decision logic matrix functions and the actuation devices while the unit is at power. When any one train is taken out of service for testing, the other train is capable of providing unit monitoring and protection until the testing has been completed. The testing device is semiautomatic to minimize testing time.

The actuation of ESF components is accomplished through master and slave relays. The SSPS energizes the master relays appropriate for the condition of the unit. Each master relay then energizes one or more slave relays, which then cause actuation of the end devices. The master and slave relays are routinely tested to ensure operation. The test of the master relays energizes the relay, which then operates the contacts and applies a low voltage to the associated slave relays. The low voltage is not sufficient to actuate the slave relays but only demonstrates signal path continuity. The SLAVE RELAY TEST actuates the devices if their operation will not interfere with continued unit operation. For the latter case, actual component operation is prevented by the SLAVE RELAY TEST circuit, and slave relay contact operation is verified by a continuity check of the circuit containing the slave relay.

D

~~Reviewers Note: No one unit ESFAS incorporates all of the Functions listed in Table 3.3.2-1. In some cases (e.g., Containment Pressure - High 3, Function 2.c), the table reflects several different implementations of the same Function. Typically, only one of these implementations are used at any specific unit.~~

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident. An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be a secondary, or backup, actuation signal for one or more other accidents. For example, Pressurizer Pressure–Low is a primary actuation signal for small loss of coolant accidents (LOCAs) and a backup actuation signal for steam line breaks (SLBs) outside containment. Functions such as manual initiation, not specifically credited in the accident safety analysis, are qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the unit. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. These Functions may also serve as backups to Functions that were credited in the accident analysis (Ref. 3).

The LCO requires all instrumentation performing an ESFAS Function to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

The LCO generally requires OPERABILITY of four or three channels in each instrumentation function and two channels in each logic and manual initiation function. The two out of three and the two out of four configurations allow one channel to be tripped during maintenance or testing without causing an ESFAS initiation. Two logic or manual initiation channels are required to ensure no single random failure disables the ESFAS.

The required channels of ESFAS instrumentation provide unit protection in the event of any of the analyzed accidents. ESFAS protective functions are as follows:

1. <u>Safety Injection</u>

   Safety Injection (SI) provides two primary functions:

   1. Primary side water addition to ensure maintenance or recovery of reactor vessel water level (coverage of the active fuel for heat removal and clad integrity, peak clad temperature < 2200°F); and

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

1. Safety Injection (continued)

2. Boration to ensure recovery and maintenance of SDM ($k_{eff} < 1.0$).

These functions are necessary to mitigate the effects of high energy line breaks (HELBs) both inside and outside of containment. The SI signal is also used to initiate other Functions such as:

- Phase A Isolation; Vent B
- Containment ~~Purge~~ Isolation;
- Reactor Trip;
- Turbine Trip;
- Feedwater Isolation;
- Start of motor driven auxiliary feedwater (AFW) pumps;
- Control room ventilation isolation; and
- Enable automatic switchover of Emergency Core Cooling System (ECCS) suction to containment sump.

These other functions ensure:

- Isolation of nonessential systems through containment penetrations;
- Trip of the turbine and reactor to limit power generation;
- Isolation of main feedwater (MFW) to limit secondary side mass losses;
- Start of AFW to ensure secondary side cooling capability;
- Isolation of the control room to ensure habitability; and
- Enabling ECCS suction from the refueling water storage tank (RWST) switchover on low low RWST level to ensure continued cooling via use of the containment sump.

1. a. Safety Injection – Manual Initiation

The LCO requires one channel per train to be OPERABLE. The operator can initiate SI at any time by using either of two switches in the control room. This action will cause actuation

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

1. a. Safety Injection – Manual Initiation (continued)

of all components in the same manner as any of the automatic actuation signals.

1 The ~~LCO on~~ STEP E Manual Initiation Function for ensures the proper amount of redundancy is maintained in the manual ESFAS actuation circuitry to ensure the operator has manual ESFAS initiation capability.

Each channel consists of one push button and the interconnecting wiring to the actuation logic cabinet. Each ~~push button~~ hand switch B actuates both trains. This configuration does not allow testing at power.

1. b. Safety Injection – Automatic Actuation Logic and Actuation Relays

This LCO requires two trains to be OPERABLE. Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment, CREVS, and ABGTS.

Manual and automatic initiation of SI must be OPERABLE in MODES 1, 2, and 3. In these MODES there is sufficient energy in the primary and secondary systems to warrant automatic initiation of ESF systems. Manual Initiation is also required in MODE 4 even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of a DBA, but because of the large number of components actuated on a SI, actuation is simplified by the use of the manual actuation push buttons. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation.

These Functions are not required to be OPERABLE in MODES 5 and 6 because there is adequate time for the operator to evaluate unit conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

1.b. Safety Injection – Automatic Actuation Logic and Actuation Relays (continued)

consequences of an abnormal condition or accident. Unit pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

1.c. Safety Injection – Containment Pressure – High 1

This signal provides protection against the following accidents:

- SLB inside containment;
- LOCA; and
- Feed line break inside containment.

Containment Pressure – High 1 provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with a two out of three logic. The transmitters (d/p cells) and electronics are located outside of containment, with the sensing line (high pressure side of the transmitter) located inside containment.

inside the containment annulus

A

The transmitters and electronics are located inside the containment annulus, but outside containment, and experience more adverse environmental conditions than if they were located outside containment altogether. However, the environmental effects are less severe than if the transmitters were located inside containment. The Trip Setpoint reflects the inclusion of both steady state instrument uncertainties and slightly more adverse environmental instrument uncertainties.

~~Thus the high pressure Function will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties~~.

B

Containment Pressure – High 1 must be OPERABLE in MODES 1, 2 and 3 when there is sufficient energy in the primary and secondary systems to pressurize the containment following a pipe break. In MODES 4, 5, and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

1.d. Safety Injection – Pressurizer Pressure – Low

This signal provides protection against the following accidents:

- Inadvertent opening of a steam generator (SG) relief or safety valve;
- SLB;
- A spectrum of rod cluster control assembly ejection accidents (rod ejection);
- Inadvertent opening of a pressurizer relief or safety valve;
- LOCAs; and
- Steam Generator Tube Rupture.

D

~~At some units pressurizer pressure provides both control and protection functions: input to the Pressurizer Pressure Control System, reactor trip, and SI. Therefore, the actuation logic must be able to withstand both an input failure to control system (which may then require the protection function actuation) and a single failure in the other channels providing the protection function actuation. Thus, four OPERABLE channels are required to satisfy the requirements with a two out of four logic. For units that have dedicated protection and control channels, only three protection channels are necessary to satisfy the protective requirements.~~

The transmitters are located inside containment, with the taps in the vapor space region of the pressurizer, and thus possibly experiencing adverse environmental conditions (LOCA, SLB inside containment, rod ejection). Therefore, the Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties.

1 High Energy Line Break

This Function must be OPERABLE in MODES 1, 2 and 3 (above P-11) to mitigate the consequences of an (HELB) inside containment. This signal may be manually blocked by the operator below the P-11 setpoint. Automatic SI actuation below this pressure setpoint is then performed by the Containment Pressure – High 1 signal.

B

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

1.d. Safety Injection – Pressurizer Pressure – Low (continued)

This Function is not required to be OPERABLE in MODE 3 below the P-11 setpoint. Other ESF functions are used to detect accident conditions and actuate the ESF systems in this MODE. In MODES 4, 5 and 6, this Function is not needed for accident detection and mitigation.

1.e. Safety Injection – Steam Line Pressure

(1) Steam Line Pressure – Low

Steam Line Pressure – Low provides protection against the following accidents:

- SLB;
- Feed line break; and
- Inadvertent opening of an SG relief or an SG safety valve.

Steam Line Pressure – Low provides no input to any control functions. Thus, three OPERABLE channels on each steam line are sufficient to satisfy the protective requirements with a two out of three logic on each steam line.

INSERT 1, see next pg. →

A,1

With the transmitters typically located inside the steam tunnels, it is possible for them to experience adverse environmental conditions during a secondary side break. Therefore, the Trip Setpoint reflects ~~both~~ only steady state ~~and adverse~~ environmental instrument uncertainties.

This Function ~~is anticipatory in nature and has~~ a ~~typical~~ lead/lag ratio of 50/5.

B

has lead/lag compensation with

Steam Line Pressure – Low must be OPERABLE in MODES 1, 2, and 3 (above P-11) when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the

(continued)

INSERT 1

With the transmitters located in areas away from the steam lines it is not possible for them to experience adverse environmental conditions during a secondary side break.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

(1) Steam Line Pressure – Low (continued)

P-11 setpoint. Below P-11, feed line break is not a concern. Inside containment, SLB will be terminated by automatic SI actuation via Containment Pressure – High (1) and outside containment SLB will be terminated by the Steam Line Pressure – Negative Rate – High signal for steam line isolation. This Function is not required to be OPERABLE in MODE 4, 5, or 6, because there is insufficient energy in the secondary side of the unit to cause an accident.

B

e. Safety Injection – Steam Line Pressure

(2) Steam Line Pressure – High Differential Pressure Between Steam Lines

Steam Line Pressure – High Differential Pressure Between Steam Lines provides protection against the following accidents:

- SLB;
- Feed line break; and
- Inadvertent opening of an SG relief or an SG safety valve.

C

Steam Line Pressure – High Differential Pressure Between Steam Lines provides no input to any control functions. Thus three OPERABLE channels on each steam line are sufficient to satisfy the requirements with a two out of three logic on each steam line.

With the transmitters typically located inside the steam tunnels, it is possible for them to experience adverse environmental conditions during an SLB event. Therefore, the Trip Setpoint reflects both steady state and adverse environmental instrument uncertainties. Steam line high differential pressure must be OPERABLE in MODES 1, 2, and 3 when a

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

(2) Steam Line Pressure – High Differential Pressure Between Steam Lines (continued)

secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). This Function is not required to be OPERABLE in MODE 4, 5, or 6, because there is not sufficient energy in the secondary side of the unit to cause an accident.

f, g. Safety Injection – High Steam Flow in Two Steam Lines Coincident With $T_{avg}$ – Low Low or Coincident With Steam Line Pressure – Low

These Functions (1.f and 1.g) provide protection against the following accidents:

- SLB; and
- the inadvertent opening of an SG relief or an SG safety valve.

C

Two steam line flow channels per steam line are required OPERABLE for these Functions. The steam line flow channels are combined in a one out of two logic to indicate high steam flow in one steam line. The steam flow transmitters provide control inputs, but the control function cannot cause the events that the function must protect against. Therefore, two channels are sufficient to satisfy redundancy requirements. The one out of two configuration allows online testing because trip of one high steam flow channel is not sufficient to cause initiation. High steam flow in two steam lines is acceptable in the case of a single steam line fault due to the fact that the remaining intact steam lines will pick up the full turbine load. The increased steam flow in the remaining intact lines will actuate the required second high steam flow trip. Additional protection is provided by Function 1.e.(2), High Differential Pressure Between Steam Lines.

One channel of $T_{avg}$ per loop and one channel of low steam line pressure per steam line are required OPERABLE. For each parameter the

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

f, g. Safety Injection – High Steam Flow in Two Steam Lines Coincident With $T_{avg}$ – Low Low or Coincident With Steam Line Pressure – Low (continued)

channels for all loops or steam lines are combined in a logic such that two channels tripped will cause a trip for the parameter. For example, for three loop units, the low steam line pressure channels are combined in two out of three logic. Thus, the Function trips on one out of two high flow in any two out of three steam lines if there is one out of one low low $T_{avg}$ trip in any two out of three RCS loops, or if there is a one out of one low pressure trip in any two out of three steam lines. Since the accidents that this event protects against cause both low steam line pressure and low low $T_{avg}$, provision of one channel per loop or steam line ensures no single random failure can disable both of these Functions. The steam line pressure channels provide no control inputs. The $T_{avg}$ channels provide control inputs, but the control function cannot initiate events that the Function acts to mitigate.

C

The Allowable Value for high steam flow is a linear function that varies with power level. The function is a ΔP corresponding to 44% of full steam flow between 0% and 20% load to 114% of full steam flow at 100% load. The nominal trip setpoint is similarly calculated.

With the transmitters typically located inside the containment ($T_{avg}$) or inside the steam tunnels (High Steam Flow), it is possible for them to experience adverse steady state environmental conditions during an SLB event. Therefore, the Trip Setpoint reflects both steady state and adverse environmental instrument uncertainties. The Steam Line Pressure – Low signal was discussed previously under Function 1.e.(1).

This Function must be OPERABLE in MODES 1, 2, and 3 (above P-12) when a secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). This

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

f, g. Safety Injection – High Steam Flow in Two Steam Lines Coincident With $T_{avg}$ – Low Low or Coincident With Steam Line Pressure – Low (continued)

signal may be manually blocked by the operator when below the P-12 setpoint. Above P-12, this Function is automatically unblocked. This Function is not required OPERABLE below P-12 because the reactor is not critical so feed line break is not a concern. SLB may be addressed by Containment Pressure High 1 (inside containment) or by High Steam Flow in Two Steam Lines coincident with Steam Line Pressure – Low for Steam Line Isolation followed by High Differential Pressure Between Two Steam Lines for SI. This Function is not required to be OPERABLE in MODE 4, 5, or 6, because there is insufficient energy in the secondary side of the unit to cause an accident.

C

A

2. Containment Spray

The primary function of Containment Spray is to lower containment pressure and temperature after an HELB in containment, and subsequently reduce the amount of fission product activity released from the containment to the environment.

~~Containment Spray provides three primary functions:~~

1. ~~Lowers containment pressure and temperature after an HELB in containment~~;

2. ~~Reduces the amount of radioactive iodine in the containment atmosphere; and~~

3. ~~Adjusts the pH of the water in the containment recirculation sump after a large break LOCA~~.

~~These~~ This functions are necessary to:

1

- Ensure the pressure boundary integrity of the containment structure;
- Limit the release of radioactive iodine to the environment in the event of a failure of the containment structure; and
- ~~Minimize corrosion of the components and systems inside containment following a LOCA~~.

A

The containment spray actuation signal starts the containment spray pumps and aligns the discharge of the pumps to the containment spray nozzle headers in

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

2. Containment Spray (continued)

the upper levels of containment. Water is initially drawn from the RWST by the containment spray pumps ~~and mixed with a sodium hydroxide solution from the spray additive tank.~~ When the RWST reaches the low low level setpoint, the spray pump suctions are shifted to the containment sump if continued containment spray is required. Containment spray is actuated manually ~~by~~ or by ~~Containment Pressure - High 3 or~~ Containment Pressure - High High.

2. a. Containment Spray - Manual Initiation

The operator can initiate containment spray at any time from the control room by simultaneously turning two containment spray actuation switches in the same train. Because an inadvertent actuation of containment spray could have such serious consequences, two switches must be turned simultaneously to initiate containment spray. There are two sets of two switches each in the control room. Simultaneously turning the two switches in either set will actuate containment spray in both trains in the same manner as the automatic actuation signal. Two Manual Initiation switches in each train are required to be OPERABLE to ensure no single failure disables the manual initiator function. Note that Manual Initiation of containment spray also actuates Phase B containment isolation.

2. b. Containment Spray - Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

Manual and automatic initiation of containment spray must be OPERABLE in MODES 1, 2, and 3 when there is a potential for an accident to occur, and sufficient energy in the primary or secondary systems to pose a threat to containment integrity due to overpressure conditions. Manual initiation is also required in MODE 4, even

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

2.b. Containment Spray – Automatic Actuation Logic and Actuation Relays (continued)

though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of a DBA. Because of the large number of components actuated on a containment spray, however, actuation is simplified by the use of the manual actuation push buttons. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary and secondary systems to result in containment overpressure. In MODES 5 and 6, there is also adequate time for the operators to evaluate unit conditions and respond to mitigate the consequences of abnormal conditions by manually starting individual components.

2 c. Containment Spray – Containment Pressure - HIGH HIGH (B)

This signal provides protection against a LOCA or an SLB inside containment. The transmitters (d/p cells) are located outside of containment, with the sensing line (high pressure side of the transmitter) located inside containment. ~~The transmitters and electronics are located outside of containment. Thus, they will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.~~

(A) inside the containment annulus

The transmitter and electronics are located inside the containment annulus, but outside containment, and experience more adverse environmental conditions than if they were located outside containment altogether. However, the environmental effects are less severe than if the transmitter were located inside containment. The Trip Setpoint reflects the inclusion of both steady state instrument uncertainties, and slightly more adverse environmental instrument uncertainties,

This is one of the only Functions that requires the ~~bistable~~ output to energize to perform its required action. It is not desirable to have a loss of power actuate containment spray, since the consequences of an inadvertent actuation of containment spray could be serious. Note that this function also has the inoperable channel placed in bypass rather than trip to decrease the probability of an inadvertent actuation.

~~Two different logic configurations are typically used. Three and four loop units use~~ four channels ~~in a two out of four logic~~ (D)

This Function uses

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

c. Containment Spray – Containment Pressure (continued)

configuration. This configuration may be called the Containment Pressure – High 3 Setpoint for three and four loop units, and Containment Pressure – High High Setpoint for other units. Some two loop units use three sets of two channels, each set combined in a one out of two configuration, with these outputs combined so that two out of three sets tripped initiates containment spray. This configuration is called Containment Pressure – High 3 Setpoint. Since containment pressure is not used for control, both of these arrangements exceed the minimum redundancy requirements. Additional redundancy is warranted because this Function is energize to trip. Containment Pressure – ~~[High 3]~~ [High High] must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the primary and secondary sides to pressurize the containment following a pipe break. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to pressurize the containment and reach the Containment Pressure – ~~[High 3] and~~ [High High] setpoints.

This arrangement exceeds

D

B

3. Containment Isolation

B

Containment Isolation provides isolation of the containment atmosphere, and all process systems that penetrate containment, from the environment. This Function is necessary to prevent or limit the release of radioactivity to the environment in the event of a large break LOCA.

There are two separate Containment Isolation signals, Phase A and Phase B. Phase A isolation isolates all automatically isolable process lines, except containment cooling ~~water~~, at a relatively low containment pressure indicative of primary or secondary system leaks. For these types of events, forced circulation cooling using the reactor coolant pumps (RCPs) and SGs is the preferred (but not required) method of decay heat removal. Since containment ~~cooling water~~ is required to support RCP

E System (CCWS) E and Essential Raw Cooling Water System (ERCW)

E Component

~~Component~~

the CCWS

1

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

3. Containment Isolation (continued)

operation, not isolating the ~~containment cooling water~~ on the low pressure Phase A signal enhances unit safety by allowing operators to use forced RCS circulation to cool the unit. Isolating ~~containment cooling water~~ on the low pressure signal may force the use of feed and bleed cooling, which could prove more difficult to control.

Phase A containment isolation is actuated automatically by SI, or manually via the automatic actuation logic. All process lines penetrating containment, with the exception of ~~component cooling water (CCW)~~, are isolated. ~~CCW~~ is not isolated at this time to permit continued operation of the RCPs with cooling water flow to the thermal barrier heat exchangers and air or oil coolers. All process lines not equipped with remote operated isolation valves are manually closed, or otherwise isolated, prior to reaching MODE 4.

Manual Phase A Containment Isolation is accomplished by either of two switches in the control room. Either switch actuates both trains. Note that manual actuation of Phase A containment isolation also actuates Containment ~~Purge~~ Vent Isolation.

INSERT MOVE From page 20-21.

3. a. Containment Isolation – Phase A Isolation

(1) Phase A Isolation – Manual Initiation

Manual Phase A Containment Isolation is actuated by either of two switches in the control room. Either switch actuates both trains. Note that manual initiation of Phase A Containment Isolation also actuates Containment ~~Purge~~ Vent Isolation.

(2) Phase A Isolation – Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

a. Containment Isolation – Phase A Isolation (continued)

Manual and automatic initiation of Phase A Containment Isolation must be OPERABLE in MODES 1, 2, and 3, when there is a potential for an accident to occur. Manual initiation is also required in MODE 4 even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of a DBA, but because of the large number of components actuated on a Phase A containment isolation, actuation is simplified by the use of the manual actuation ~~push buttons~~ hand switches. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment to require Phase A containment isolation. There also is adequate time for the operator to evaluate unit conditions and manually actuate individual isolation valves in response to abnormal or accident conditions.

(3) Phase A Isolation – Safety Injection

Containment Phase A Isolation is also initiated by all Functions that initiate SI. The Containment Phase A Isolation requirements for these Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, SI (Function 1) is referenced for all initiating Functions and requirements.

The Phase B signal isolates ~~Containment cooling water, is isolated by the Phase B signal, which~~ This occurs at a relatively high containment pressure that is indicative of a large break LOCA or an SLB. For these events, forced circulation using the RCPs is no longer desirable. Isolating the CCWS at the higher pressure does not pose a challenge to the containment boundary, because the CCWS ~~System~~ is a closed loop inside containment. Although some

MOVE TO Page 19

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

3 a. Containment Isolation – Phase A Isolation (continued)

system components do not meet all of the American Society of Mechanical Engineers (ASME) Code requirements applied to the containment itself, the system is continuously pressurized to a pressure greater than the Phase B setpoint. Thus, routine operation demonstrates the integrity of the system pressure boundary for pressures exceeding the Phase B setpoint. Furthermore, because system pressure exceeds the Phase B setpoint, any system leakage prior to initiation of Phase B isolation would be into the containment. Therefore, the combination of CCWS System design and Phase B isolation ensures the CCWS ~~System~~ is not a potential path for radioactive release from containment.

MOVE TO Pg 19

Phase B containment isolation is actuated by ~~Containment Pressure – High 3 or~~ Containment Pressure – High High, or manually, via the automatic actuation logic, as previously discussed. For containment pressure to reach a value high enough to actuate ~~Containment Pressure – High 3 or~~ Containment Pressure – High High, a large break LOCA or SLB must have occurred and containment spray must have been actuated. RCP operation will no longer be required and CCWS to the RCPs is, therefore, no longer necessary. The RCPs ~~can~~ could be operated with seal injection flow alone and without CCWS flow to the thermal barrier heat exchanger.

Manual Phase B Containment Isolation is accomplished by the same switches that actuate Containment Spray. When the two switches in either set are turned simultaneously, Phase B Containment Isolation and Containment Spray will be actuated in both trains.

3. b. Containment Isolation – Phase B Isolation

Phase B Isolation is accomplished by Manual Initiation, Automatic Actuation Logic and Actuation Relays, and by Containment Pressure

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

3. b. Containment Isolation – Phase B Isolation (continued)

A

channels (~~the same channels that actuate Containment Spray, Function 2~~). The Containment Pressure ~~trip~~ initiation of Phase B Isolation is energized to ~~trip~~ initiations in order to minimize the potential of spurious ~~trips~~ initiations that may damage the RCPs.

B

(1) Phase B Isolation – Manual Initiation

(2) Phase B Isolation – Automatic Actuation Logic and Actuation Relays

1

Automatic Actuation Logic and Actuation Relays consist of the same features, and operate in the same manner as described in ESFAS FUNCTION 1.b

Manual and automatic initiation of Phase B containment isolation must be OPERABLE in MODES 1, 2, and 3, when there is a potential for an accident to occur. Manual initiation is also required in MODE 4 even though automatic actuation is not required. In this MODE adequate time is available to manually actuate required components in the event of a DBA. However, because of the large number of components actuated on a Phase B containment isolation, actuation is simplified by the use of the manual actuation push buttons. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment to require Phase B containment isolation. There also is adequate time for the operator to evaluate unit conditions and manually actuate individual isolation valves in response to abnormal or accident conditions.

(3) Phase B Isolation – Containment Pressure HIGH HIGH

B

The basis for containment pressure MODE applicability is as discussed for Function 2.c above.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

4. Steam Line Isolation

Isolation of the main steam lines provides protection in the event of an SLB inside or outside containment. Rapid isolation of the steam lines will limit the steam break accident to the blowdown from one SG, at most. For an SLB upstream of the main steam isolation valves (MSIVs), inside or outside of containment, closure of the MSIVs limits the accident to the blowdown from only the affected SG. For an SLB downstream of the MSIVs, closure of the MSIVs terminates the accident as soon as the steam lines depressurize. ~~For units that do not have steam line check valves,~~ Steam Line Isolation also mitigates the effects of a feed line break and ensures a source of steam for the turbine driven AFW pump during a feed line break.

D

4.a. Steam Line Isolation – Manual Initiation

Manual initiation of Steam Line Isolation can be accomplished from the control room. There are four ~~two~~ switches in the control room ~~and either switch can initiate action to~~ which can immediately close each individual ~~all~~ MSIV~~s~~. The LCO requires ~~two channel~~s to be OPERABLE.

A

4.b. Steam Line Isolation – Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

Manual and automatic initiation of steam line isolation must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the RCS and SGs to have an SLB or other accident. This could result in the release of significant quantities of energy and cause a cooldown of the primary system. Although it is possible to close all MSIVs in MODES 2 and 3, the Steam Line Isolation Function remains OPERABLE in these MODES to ensure that the MSIVs can be immediately closed if they are ~~inadvertently~~ opened. In MODES 4, 5, and 6, there is insufficient energy in

E

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

4. Steam Line Isolation (continued)

the RCS and SGs to experience an SLB or other accident releasing significant quantities of energy.

c. Steam Line Isolation – Containment Pressure – High ~~2~~ HIGH

This function actuates closure of the MSIVs in the event of a LOCA or an SLB inside containment to maintain at least one unfaulted SG as a heat sink for the reactor and to limit the mass and energy release to containment. The transmitters (d/p cells) are located outside containment, with the sensing line (high pressure side of the transmitter) located inside containment. Containment Pressure – High ~~2~~ HIGH provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with two out of three logic. However, for enhanced reliability, this function was designed with four channels and two out of four logic. ~~The transmitters and electronics are located outside of containment. Thus, they will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.~~

A inside the containment annulus

A The transmitters and electronics are located inside the containment annulus, but outside containment, and experience more adverse environmental conditions than if they were located outside containment altogether. However, the environmental effects are less severe than if the transmitters were located inside containment. The Trip Setpoint reflects the inclusion of both steady state instrument uncertainties, and slightly more adverse environmental instrument uncertainties.

Containment Pressure – High ~~2~~ HIGH must be OPERABLE in MODES 1, 2, and 3, when there is sufficient energy in the primary and secondary side to pressurize the containment following a pipe break. This would cause a significant increase in the containment pressure, thus allowing detection and closure of the MSIVs. Although it is possible to close all MSIVs in MODES 2 and 3, the Steam Line Isolation function remains OPERABLE in these MODES to ensure that the MSIVs can be immediately closed if ~~inadvertently~~ open. E In MODES 4, 5, and 6, there is not enough energy in the primary and secondary sides to pressurize the containment to the Containment Pressure – High ~~2~~ HIGH setpoint.

B

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

4.d. Steam Line Isolation – Steam Line Pressure

(1) Steam Line Pressure – Low

Steam Line Pressure – Low provides closure of the MSIVs in the event of an SLB to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. This function provides closure of the MSIVs in the event of a feed line break to ensure a supply of steam for the turbine driven AFW pump. Steam Line Pressure – Low was discussed previously under SI function 1.e.

Steam Line Pressure – Low Function must be OPERABLE in MODES 1, 2, and 3 (above P-11), ~~with any main steam valve open~~ when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the P-11 setpoint. Below P-11, an inside containment SLB will be terminated by automatic actuation via Containment Pressure – High, and stuck valve transients and outside containment SLBs will be terminated by the Steam Line Pressure – Negative Rate – High signal for Steam Line Isolation. Although it is possible to close all MSIVs in MODES 2 and 3, the Steam Line Isolation function remains OPERABLE in these MODES to ensure that the MSIVs can be immediately closed if they are ~~inadvertently~~ opened. This Function is not required to be OPERABLE in MODES 4, 5, and 6, because there is insufficient energy in the secondary side of the unit to have an accident.

4.e Steam Line Pressure – Negative Rate – High

Steam Line Pressure – Negative Rate – High provides closure of the MSIVs for an SLB, when less than the P-11 setpoint, to

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

4.e. ~~17~~ Steam Line Pressure – Negative Rate – High (continued)

maintain at least one unfaulted SG as a heat sink for the reactor and to limit the mass and energy release to containment. When the operator manually blocks the Steam Line Pressure – Low main steam isolation signal when less than the P-11 setpoint, the Steam Line Pressure – Negative Rate – High signal is automatically enabled. Steam Line Pressure – Negative Rate – High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy requirements with a two out of three logic on each steam line.

Steam Line Pressure – Negative Rate – High must be OPERABLE in MODE 3, when less than the P-11 setpoint, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). In MODES 1 and 2, and in MODE 3 when above the P-11 setpoint, this signal is automatically disabled and the Steam Line Pressure – Low signal is automatically enabled. Although it is possible to close all MSIVs in MODES 2 and 3, the Steam Line Isolation Function must remain OPERABLE in these MODES to ensure that the MSIVs can be immediately closed if they are E ~~inadvertently~~ opened. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to have an SLB or other accident that would result in a release of significant enough quantities of energy to cause a cooldown of the RCS.

While the transmitters may experience elevated ambient temperatures due to an SLB, the trip function is based on rate of change, not the absolute accuracy of the indicated steam pressure. Therefore, the Trip Setpoint reflects only steady state instrument uncertainties.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

e, f. Steam Line Isolation—High Steam Flow in Two Steam Lines Coincident with $T_{avg}$—Low Low or Coincident With Steam Line Pressure—Low (Three and Four Loop Units)

These Functions (4.e and 4.f) provide closure of the MSIVs during an SLB or inadvertent opening of an SG relief or a safety valve, to maintain at least one unfaulted SG as a heat sink for the reactor and to limit the mass and energy release to containment.

These Functions were discussed previously as Functions 1.f. and 1.g.

These Functions must be OPERABLE in MODES 1 and 2, and in MODE 3, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. These Functions are not required to be OPERABLE in MODES 4, 5, and 6, because there is insufficient energy in the secondary side of the unit to have an accident.

C

g. Steam Line Isolation—High Steam Flow Coincident With Safety Injection and Coincident With $T_{avg}$—Low Low (Two Loop Units)

This Function provides closure of the MSIVs during an SLB or inadvertent opening of an SG relief or safety valve to maintain at least one unfaulted SG as a heat sink for the reactor and to limit the mass and energy release to containment.

Two steam line flow channels per steam line are required OPERABLE for this Function. These are combined in a one out of two logic to indicate high steam flow in one steam line. The steam flow transmitters provide control inputs, but the control function cannot cause the events that the function must protect against. Therefore, two channels are sufficient to satisfy redundancy requirements. The one out of two configuration allows online testing because trip of one high

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

g. Steam Line Isolation – High Steam Flow Coincident With Safety Injection and Coincident With $T_{avg}$ – Low Low (Two Loop Units) (continued)

steam flow channel is not sufficient to cause initiation.

The High Steam Flow Allowable Value is a $\Delta P$ corresponding to 25% of full steam flow at no load steam pressure. The Trip Setpoint is similarly calculated.

C

With the transmitters (d/p cells) typically located inside the steam tunnels, it is possible for them to experience adverse environmental conditions during an SLB event. Therefore, the Trip Setpoints reflect both steady state and adverse environmental instrument uncertainties.

The main steam line isolates only if the high steam flow signal occurs coincident with an SI and low low RCS average temperature. The Main Steam Line Isolation Function requirements for the SI Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1, SI, is referenced for all initiating functions and requirements.

Two channels of $T_{avg}$ per loop are required to be OPERABLE. The $T_{avg}$ channels are combined in a logic such that two channels tripped cause a trip for the parameter. The accidents that this Function protects against cause reduction of $T_{avg}$ in the entire primary system. Therefore, the provision of two OPERABLE channels per loop in a two out of four configuration ensures no single random failure disables the $T_{avg}$ – Low Low Function. The $T_{avg}$ channels provide control inputs, but the control function cannot initiate events that the Function acts to mitigate. Therefore, additional channels are not required to address control protection interaction issues.

With the $T_{avg}$ resistance temperature detectors (RTDs) located inside the containment, it is

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

g. Steam Line Isolation—High Steam Flow Coincident With Safety Injection and Coincident With $T_{avg}$—Low Low (Two Loop Units) (continued)

possible for them to experience adverse environmental conditions during an SLB event. Therefore, the Trip Setpoint reflects both steady state and adverse environmental instrumental uncertainties.

This Function must be OPERABLE in MODES 1 and 2, and in MODE 3, when above P-12 setpoints, when a secondary side break or stuck open valve could result in rapid depressurization of the steam lines. Below P-12 this Function is not required to be OPERABLE because the High High Steam Flow coincident with SI Function provides the required protection. Although it is possible to close all MSIVs in MODES 2 and 3, the Steam Line Isolation Function remains OPERABLE in these MODES to ensure that the MSIVs can be immediately closed if inadvertently open. This Function is not required to be OPERABLE in MODES 4, 5, and 6, because there is insufficient energy in the secondary side of the unit to have an accident.

C

h. Steam Line Isolation—High High Steam Flow Coincident With Safety Injection (Two Loop Units)

This Function provides closure of the MSIVs during an SLB (or inadvertent opening of an SG relief or safety valve) to maintain at least one unfaulted SG as a heat sink for the reactor and to limit the mass and energy release to containment.

Two steam line flow channels per steam line are required to be OPERABLE for this Function. These are combined in a one out of two logic to indicate high steam flow in one steam line. The steam flow transmitters provide control inputs, but the control function cannot cause the events that the Function must protect against. Therefore, two channels are sufficient to satisfy redundancy requirements.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

h. Steam Line Isolation – High High Steam Flow Coincident With Safety Injection (Two Loop Units) (continued)

The Allowable Value for high steam flow is a ΔP, corresponding to 130% of full steam flow at full steam pressure. The Trip Setpoint is similarly calculated.

With the transmitters typically located inside the steam tunnels, it is possible for them to experience adverse environment conditions during an SLB event. Therefore, the Trip Setpoint reflects both steady state and adverse environmental instrument uncertainties.

C

The main steam lines isolate only if the high steam flow signal occurs coincident with an SI signal. The Main Steam Line Isolation Function requirements for the SI Functions are the same as the requirements for their SI Function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1, SI, is referenced for all initiating functions and requirements.

This Function must be OPERABLE in MODES 1, 2, and 3 when a secondary side break or stuck open valve could result in rapid depressurization of the steam lines. This Function is not required to be OPERABLE in MODES 4, 5, and 6, because there is insufficient energy in the secondary side of the unit to have an accident.

5. Turbine Trip and Feedwater Isolation

The primary functions of the Turbine Trip and Feedwater Isolation signals are to prevent damage to the turbine due to water in the steam lines, and to stop the excessive flow of feedwater into the SGs. These Functions are necessary to mitigate the effects of a high water level in the SGs, which could result in carryover of water into the steam lines and excessive cooldown of the primary system. The SG high water level is due to excessive feedwater flows.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

5. Turbine Trip and Feedwater Isolation (continued)

This Function is actuated by SG Water Level – High High, or by an SI signal. The RTS also initiates a turbine trip signal whenever a reactor trip (P-4) is generated. In the event of SI, the unit is taken off line and the turbine generator must be tripped. The MFW System is also taken out of operation and the AFW System is automatically started. The SI signal was discussed previously.

a. Turbine Trip and Feedwater Isolation – Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

b. Turbine Trip and Feedwater Isolation – Steam Generator Water Level – High High (P-14)

This signal provides protection against excessive feedwater flow. The ESFAS SG water level instruments provide input to the SG Water Level Control System. Therefore, the actuation logic must be able to withstand both an input failure to the control system (which may then require the protection function actuation) and a single failure in the other channels providing the protection function actuation. ~~Thus four OPERABLE channels are required to satisfy the requirements with a two out of four logic. For units that have dedicated protection and control channels, only three protection channels are necessary to satisfy the protective requirements. For other units that have only three channels, justification is provided in NUREG-1218 (Ref. 7).~~

The transmitters (d/p cells) are located inside containment. However, the events that this Function protects against cannot cause a severe environment in containment. Therefore, the Trip Setpoint reflects only steady state instrument uncertainties.

A level per SG C

Since Watts Bar has only 3 channels, control/protection interaction is addressed by the use of a Median Signal Selector which prevents a single failure of a channel providing input to the control system requiring protection function action. That is, a single failure of a channel providing input to the control system does not result in the control system initiating a condition requiring protection function action. The Median Signal Selector performs this by not selecting the channels indicating the highest or lowest steam generator levels as input to the control system.

~~a median signal selector is provided~~

(continued)



Since no adverse control system action may now result from a single failed protection instrument channel, a second random protection system failure (as would otherwise be required by IEEE 279-1971) need not be considered.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

c. Turbine Trip and Feedwater Isolation – Safety Injection

Turbine Trip and Feedwater Isolation ~~is~~ are also initiated by all Functions that initiate SI. The Feedwater Isolation Function requirements for these functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1, SI, is referenced for all initiating functions and requirements.

Turbine Trip and Feedwater Isolation Functions must be OPERABLE in MODES 1 and 2 when the MFW System is in operation and the turbine generator may be in operation. In MODES 3, 4, 5, and 6, the MFW System and the turbine generator are not in service and this Function is not required to be OPERABLE.

6. Auxiliary Feedwater

The AFW System is designed to provide a secondary side heat sink for the reactor in the event that the MFW System is not available. The system has two motor driven pumps and a turbine driven pump, making it available during normal unit operation, during a loss of AC power, a loss of MFW, and during a Feedwater System pipe break. The normal source of water for the AFW System is the condensate storage tank (CST) (~~normally not~~ non safety related). A low ~~level in the CST~~ suction pressure to the AFW pump (A) will automatically realign the pump suctions to the Essential ~~Service Water (ESW)~~ Raw Cooling Water (ERCW) (B) System (safety related). The AFW System is aligned so that upon a pump start, flow is initiated to the respective SGs immediately.

a. Auxiliary Feedwater – Automatic Actuation Logic and Actuation Relays (Solid State Protection System)

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

3

~~b. Auxiliary Feedwater – Automatic Actuation Logic and Actuation Relays (Balance of Plant ESFAS)~~

~~Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.~~

6b. Auxiliary Feedwater – Steam Generator Water Level – Low Low

SG Water Level – Low Low provides protection against a loss of heat sink. A feed line break, inside or outside of containment, or a loss of MFW would result in a loss of SG water level. SG Water Level – Low Low provides input to the SG Level Control System. ~~Therefore, the actuation logic must be able to withstand both an input failure to the control system which may then require a protective function actuation and a single failure in the other channels providing the protection function actuation. Thus, four OPERABLE channels are required to satisfy the requirements with two out of four logic. For units that have dedicated protection and control channels, only three protection channels are necessary to satisfy the protective requirements. For other units that have only three channels, justification is provided in Reference 7.~~

as well as Automatic Actuation of AFW,

C

Since WATTS BAR has only 3 channels, control protection interaction is addressed by the use of a Median Signal Selector, as discussed in the ... for Function 5b, "Steam Generator Water Level High High."

~~a median signal selector is provided~~ or

INSERT YY

With the transmitters (d/p cells) located inside containment and thus possibly experiencing adverse environmental conditions (feed line break), the Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties.

6c. Auxiliary Feedwater – Safety Injection

An SI signal starts the motor driven and turbine driven AFW pumps. The AFW initiation functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating Functions and requirements.

(continued)

INSERT YY

The Steam Generator Water Level Trip Time Delay (TTD) creates additional operational margin when the plant needs it most, during escalation to power, by allowing the operator time to recover level when the primary side load is sufficiently small to allow such action. The TTD is based on continuous monitoring of primary side power through the use of vessel $\Delta T$. Two time delays are calculated, based on the number of steam generators indicating less than the Low-Low Trip Setpoint. The magnitude of the delays decreases with increasing primary side power level, up to 50% RTP. Above 50% RTP there are no time delays for the Low-Low Level channel trips.

In the event of failure of a Steam Generator Water Level channel, the channel is placed in the trip condition as input to the Solid State Protection System (SSPS) and does not affect the TTD setpoint calculations for the remaining OPERABLE channels. It is then necessary for the operator to force the use of the shorter TTD time delay by adjustment of the single steam generator time delay calculation ($T_s$) to match the multiple steam generator time delay calculation ($T_m$) for the affected protection set, through the Man Machine Interface. Failure of the vessel $\Delta T$ channel input (failure of more than one $T_H$ RTD or failure of a $T_C$ RTD) does not affect the TTD calculation for a protection set. This results in the requirement that the operator adjust the threshold power level for zero seconds time delay from 50% RTP to 0% RTP, through the Man Machine Interface.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

6.d. Auxiliary Feedwater – Loss of Offsite Power

A loss of offsite power to the ~~service~~ RCP buses will be accompanied by a loss of reactor coolant pumping power and the subsequent need for some method of decay heat removal. The loss of offsite power is detected by a voltage drop on each ~~service bus~~ 6.9 kv shutdown board. Loss of power to ~~either~~ a ~~service bus~~ will start the turbine driven AFW pump to ensure that at least one SG contains enough water to serve as the heat sink for reactor decay heat and sensible heat removal following the reactor trip.

A

Functions 6.a through 6.d must be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink for the reactor. SG Water Level – Low Low in any operating SG will cause the motor driven AFW pumps to start. The system is aligned so that upon a start of the pump, water immediately begins to flow to the SGs. SG Water Level – Low Low in any two operating SGs will cause the turbine driven pumps to start. These Functions do not have to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink. In MODE 4, AFW actuation does not need to be OPERABLE because either AFW or RHR will already be in operation to remove decay heat.

A

~~f. Auxiliary Feedwater – Undervoltage Reactor Coolant Pump~~

~~A loss of power on the buses that provide power to the RCPs provides indication of a pending loss of RCP forced flow in the RCS. The Undervoltage RCP Function senses the voltage downstream of each RCP breaker. A loss of power or open RCP breaker on two or more RCPs will start the turbine driven AFW pump to ensure that at least one SG contains enough water to serve as the heat sink for reactor decay heat and sensible heat removal following the reactor trip.~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

6.e ~~g.~~ Auxiliary Feedwater – Trip Of All Main Feedwater Pumps

A

A Trip of ~~all~~ (both turbine driven) MFW pumps is an indication of a loss of MFW and the subsequent need for some method of decay heat and sensible heat removal to bring the reactor back to no load temperature and pressure. A turbine driven MFW pump is equipped with ~~two~~ (one) pressure switch~~es~~ on the control ~~air~~ (OIL) line for the speed control system. A low pressure signal from ~~either of these~~ (this) pressure switch~~es~~ indicates a trip of that pump. ~~Motor driven MFW pumps are equipped with a breaker position sensing device. An open supply breaker indicates that the pump is not running. Two OPERABLE channels per pump satisfy redundancy requirements with one out of two taken twice logic.~~ A trip of ~~all~~ (both turbine driven) MFW pumps starts the motor driven and turbine driven AFW pumps to ensure that at least one SG is available with water to act as the heat sink for the reactor.

A

A

Function~~s~~ (This) ~~6.f and 6.g~~ must be OPERABLE in MODES 1 and 2. This ensures that at least one SG is provided with water to serve as the heat sink to remove reactor decay heat and sensible heat in the event of an accident. In MODES 3, 4, and 5, the RCPs and MFW pumps may be normally shutdown and thus neither pump trip is indicative of a condition requiring automatic AFW initiation.

6.f and ~~6.g.~~ Auxiliary Feedwater – Pump Suction Transfer on Suction Pressure – Low

A low pressure on the AFW pump suction (Three each motor driven) protects the AFW pumps against a loss of the normal supply of water for the pumps, the CST. ~~Two~~ pressure switches are located on ~~the~~ AFW pump suction line (and two sets of three pressure switches on the turbine driven AFW pump suction line from the CST.) from the CST. A low pressure sensed by ~~any one~~ (two switches of a set) ~~of the switches~~ will cause the emergency supply of water for ~~both~~ (the respective) pump~~s~~ to be aligned ~~or cause the AFW pumps to stop until the emergency source of water is aligned~~. ~~ESW~~ (safety grade) is then lined up to supply the AFW pumps to ensure an adequate supply of water for the AFW System to

A

A

B ESW

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

6 f and 6g.

Auxiliary Feedwater – Pump Suction Transfer on Suction Pressure – Low (continued)

maintain at least one of the SGs as the heat sink for reactor decay heat and sensible heat removal.

Since the detectors are located in area not affected by HELBs or high radiation, they will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.

These Functions ~~This Function~~ must be OPERABLE in MODES 1, 2, and 3 to ensure a safety grade supply of water for the AFW System to maintain the SGs as the heat sink for the reactor. These ~~This~~ Functions do~~es~~ not have to be OPERABLE in MODES 5 and 6, because there is not enough heat being generated in the reactor to require the SGs as a heat sink. In MODE 4, AFW ~~actuation~~ does not need to be OPERABLE because ~~either AFW or~~ RHR will already be in operation, or sufficient time is available to place RHR in operation, to remove decay heat.

A

E automatic Suction transfer

E

7. Automatic Switchover to Containment Sump

At the end of the injection phase of a LOCA, the RWST will be nearly empty. Continued cooling must be provided by the ECCS to remove decay heat. The source of water for the ECCS pumps is automatically switched to the containment recirculation sump. The low head residual heat removal (RHR) pumps a~~nd containment spray pumps~~ draw the water from the containment recirculation sump, the RHR pumps pump the water through the RHR heat exchanger, inject the water back into the RCS, and supply the cooled water to the other ECCS pumps. Switchover from the RWST to the containment sump must occur before the RWST empties, to prevent damage to the RHR pumps and a loss of core cooling capability. For similar reasons, switchover must not occur before there is sufficient water in the containment sump to support ESF pump suction. Furthermore, early switchover must not occur to ensure that sufficient borated water is injected from the RWST. This ensures the reactor remains shut down in the recirculation mode.

A

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

7.a. Automatic Switchover to Containment Sump – Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

D ~~b. Automatic Switchover to Containment Sump – Refueling Water Storage Tank (RWST) Level – Low Low Coincident With Safety Injection and~~

7.b ~~c.~~ Automatic Switchover to Containment Sump – RWST Level – Low Low Coincident With Safety Injection and Coincident With Containment Sump Level – High

As the RWST empties, the water from the RWST and the RCS, and accumulators will collect inside containment.

1.

During the injection phase of a LOCA, the RWST is the source of water for all ECCS pumps. A low low level in the RWST coincident with an SI signal provides protection against a loss of water for the ECCS pumps and indicates the end of the injection phase of the LOCA. The RWST is equipped with four level transmitters. These transmitters provide no control Functions. Therefore, ~~a~~ two out of four logic is adequate to initiate the protective Function actuation. Although only three channels would be sufficient, a fourth channel has been added for increased reliability.

F

The RWST – Low Low Allowable Value/Trip Setpoint has both upper and lower limits. The lower limit is selected to ensure switchover occurs before the RWST empties to prevent ECCS pump damage. The upper limit is selected to ensure enough borated water is injected to ensure the reactor remains shut down. The ~~high~~ upper limit also ensures adequate water inventory in the containment sump to provide ECCS pump suction.

F

The transmitters are located in an area not affected by HELBs or post accident high radiation. Thus, they will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

7.6 ~~8~~. Automatic Switchover to Containment Sump – RWST Level – Low Low Coincident With Safety Injection and Coincident With Containment Sump Level – High (continued)

Automatic switchover occurs only if the RWST low low level signal is coincident with SI. This prevents accidental switchover during normal operation. Accidental switchover could damage ECCS pumps if they are attempting to take suction from an empty sump. The automatic switchover Function requirements for the SI Functions are the same as the requirements for their SI Function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating Functions and requirements.

D

11

~~Reviewer Note: In some units~~, ~~A~~dditional protection from spurious switchover is provided by requiring a Containment Sump Level – High signal as well as RWST Level – Low Low and SI.

INSERT SEE next Page

~~Units only have one of the Functions, 7.b or 7.c~~.

These Functions must be OPERABLE in MODES 1, 2, 3, and 4 when there is a potential for a LOCA to occur, to ensure a continued supply of water for the ECCS pumps. These Functions are not required to be OPERABLE in MODES 5 and 6, because there is adequate time for the operator to evaluate unit conditions and respond by manually starting systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. System pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

8. Engineered Safety Feature Actuation System Interlocks

To allow some flexibility in unit operations, several interlocks are included as part of the ESFAS. These interlocks permit the operator to block some signals, automatically enable other signals, prevent some

(continued)

INSERT
Function 7.b.

The containment sump is equipped with four level transmitters. These transmitters provide no control functions. ~~therefore, a~~ Two-out-of-four logic is adequate to initiate the protective function actuation. Although three channels would be sufficient, a fourth channel has been added for increased reliability. The containment sump level Trip Setpoint/Allowable Value is selected to ensure the reactor remains shutdown. The high limit also ensures adequate water inventory in the containment sump to provide ECCS pump suction. The transmitters are located inside containment, and thus possibly experience adverse environmental conditions. Therefore, the trip setpoint reflects the inclusion of both steady-state and environmental instrument uncertainties.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

8. Engineered Safety Feature Actuation System Interlocks (continued)

actions from occurring, and cause other actions to occur. The interlock Functions back up manual actions to ensure bypassable functions are in operation under the conditions assumed in the safety analyses.

a. Engineered Safety Feature Actuation System Interlocks – Reactor Trip, P-4

The P-4 interlock is enabled when ~~either~~ a ~~reactor trip breaker~~ (RTB) ~~or their~~ and its associated bypass breaker are open. Once the P-4 interlock is enabled, automatic SI initiation ~~is~~ may be blocked after a ~~30~~ 90 second time delay. This Function allows operators to take manual control of SI systems after the initial phase of injection is complete. Once SI is blocked, automatic actuation of SI cannot occur until the RTBs have been manually closed. The functions of the P-4 interlock are:

- Trip the main turbine;
- Isolate MFW with coincident low $T_{avg}$;
- Prevent reactuation of SI after a manual reset of SI;
- Transfer the steam dump from the load rejection controller to the unit trip controller; and
- Prevent opening of the ~~main feedwater~~ MFW isolation valves if they were closed on SI or SG Water Level – High High.

Each of the above Functions is interlocked with P-4 to avert or reduce the continued cooldown of the RCS following a reactor trip. An excessive cooldown of the RCS following a reactor trip could cause an insertion of positive reactivity with a subsequent increase in generated power. To avoid such a situation, the noted Functions have been interlocked with P-4 as part of the design of the unit control and protection system.

None of the noted Functions serves a mitigation Function in the unit licensing basis safety analyses. Only the turbine trip Function is

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

8. a. Engineered Safety Feature Actuation System Interlocks – Reactor Trip, P-4 (continued)

explicitly assumed since it is an immediate consequence of the reactor trip Function. Neither turbine trip nor any of the other four Functions associated with the reactor trip signal is required to show that the unit licensing basis safety analysis acceptance criteria are not exceeded.

The RTB position switches that provide input to the P-4 interlock only function to energize or de-energize or open or close contacts. Therefore, this Function has no adjustable trip setpoint with which to associate a Trip Setpoint and Allowable Value.

This Function must be OPERABLE in MODES 1, 2, and 3 when the reactor may be critical or approaching criticality. This Function does not have to be OPERABLE in MODES 4, 5 or 6, because the main turbine, the MFW System, and the Steam Dump System are not in operation.

8. b. Engineered Safety Feature Actuation System Interlocks – Pressurizer Pressure, P-11

When the Steam Line Pressure-Low SI and steamline isolation signals are manually blocked, the Steam Line Pressure-

C

Negative Rate-High is automatically enabled. With two out of three pressurizer pressure channels ≥ P-11 setpoint, the Pressurizer Pressure-Low and Steam Line Pressure Low SI signals are automatically enabled and Steamline Pressure Negative Rate-High is automatically blocked.

The P-11 interlock permits a normal unit cooldown and depressurization without actuation of SI or main steam line isolation. With two out of three pressurizer pressure channels (discussed previously) less than the P-11 setpoint, the operator can manually block the Pressurizer Pressure – Low and Steam Line Pressure – Low SI signals, and the Steam Line Pressure – Low steam line isolation signal (previously discussed). When the Steam Line Pressure – Low steam line isolation signal is manually blocked, a main steam isolation signal on Steam Line Pressure – Negative Rate – High is enabled. This provides protection for an SLB by closure of the MSIVs. With two out of three pressurizer pressure channels ≥ P-11 setpoint, the Pressurizer Pressure – Low and Steam Line Pressure – Low SI signals and the Steam Line

(continued)



and Steamline isolation

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

b. Engineered Safety Feature Actuation System Interlocks – Pressurizer Pressure, P-11 (continued)

Pressure – Low steam line isolation signal are automatically enabled. The operator can also enable these trips by use of the respective manual reset buttons. When the Steam Line Pressure – Low steam line isolation signal is enabled, the main steam isolation on Steam Line Pressure – Negative Rate – High is disabled. The Trip Setpoint reflects only steady state instrument uncertainties.

C

This Function must be OPERABLE in MODES 1, 2, and 3 to allow an orderly cooldown and depressurization of the unit without the actuation of SI, or main steam isolation. This Function does not have to be OPERABLE in MODE 4, 5, or 6, because system pressure must already be below the P-11 setpoint for the requirements of the heatup and cooldown curves to be met.

c. Engineered Safety Feature Actuation System Interlocks – $T_{avg}$ – Low Low, P-12

On increasing reactor coolant temperature, the P-12 interlock reinstates SI on High Steam Flow Coincident With Steam Line Pressure – Low or Coincident With $T_{avg}$ – Low Low and provides an arming signal to the Steam Dump System. On decreasing reactor coolant temperature, the P-12 interlock allows the operator to manually block SI on High Steam Flow Coincident With Steam Line Pressure – Low or Coincident with $T_{avg}$ – Low Low. On a decreasing temperature, the P-12 interlock also removes the arming signal to the Steam Dump System to prevent an excessive cooldown of the RCS due to a malfunctioning Steam Dump System. Since $T_{avg}$ is used as an indication of bulk RCS temperature, this Function meets redundancy requirements with one OPERABLE channel in each loop. In three loop units these channels are used in two out of three logic. In four loop units they are used in two out of four logic.

C

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

C

~~c. Engineered Safety Feature Actuation System Interlocks – $T_{avg}$ – Low Low, P-12 (continued)~~

~~This Function must be OPERABLE in MODES 1, 2, and 3 when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. This Function does not have to be OPERABLE in MODES 4, 5, or 6, because there is insufficient energy in the secondary side of the unit to have an accident.~~

B, ~~A~~.C Engineered Safety Feature Actuation System Interlocks – Steam Generator Water Level – High High, P-14

The P-14 interlock is actuated when the level in any SG exceeds the high high setpoint, and performs the following functions as part of Function 5:

- Trips the main turbine;
- Trips the MFW pumps;
- Initiates feedwater isolation; and
- Shuts the MFW regulating valves and the bypass feedwater regulating valves.

The MFW pumps are tripped, feedwater isolation is actuated, and the main and bypass feedwater regulating valves are closed to prevent any further addition of water to the SGs. The main turbine is tripped to prevent carryover of excessive moisture to the turbine, which would damage the turbine. This Function has previously been discussed as Function 5.b.

A

The reactor is tripped by the turbine trip to prevent excessive cooldown.

This Function must be OPERABLE in MODES 1 and 2 when the turbine generator and the MFW System may be in operation. This Function does not have to be OPERABLE in MODE 3, 4, 5, or 6, because the turbine generator and the MFW System are not in service.

is

A

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

d. Engineered Safety Feature Actuation System Interlocks – Steam Generator Water Level – High High, P-14 (continued)

The ESFAS instrumentation satisfies Criterion 3 of the NRC Policy Statement.

ACTIONS

In the event a channel's Trip Setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, a rack module, or an SSPS module is found inoperable, then the Function which that channel provides must be declared inoperable and the LCO Condition entered for the particular protection Function affected. When the Required Channels in Table 3.3.2-1 are specified (e.g., on a per steam line, per loop, per SG, etc., basis), then the Condition may be entered separately for each steam line, loop, SG, etc., as appropriate.

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.2-1. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

D

~~Reviewer Note: Certain LCO Completion Times are based on approved topical reports. In order for a licensee to use the times the licensee must justify the Completion Times as required by the staff Safety Evaluation Report (SER) for the topical report.~~

When the number of inoperable channels in a trip function exceed those specified in one or other related Conditions associated with a trip function, then the unit is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if applicable in the current MODE of operation.

(continued)

BASES

ACTIONS (continued)

C.1, C.2.1 and C.2.2

Condition C is applicable to the automatic actuation logic and actuation relays for the following functions:

- SI;
- Containment Spray;
- Phase A Isolation;
- Phase B Isolation; and
- Automatic Switchover to Containment Sump.

This action addresses the train orientation of the SSPS and the master and slave relays. If one train is inoperable, 6 hours are allowed to restore the train to OPERABLE status. The specified Completion Time is reasonable considering that there is another train OPERABLE and the low probability of an event occurring during this interval. If the train cannot be restored to OPERABLE status, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 within an additional 6 hours (12 hours total time) and in MODE 5 within an additional 30 hours (42 hours total time). The Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

The Required Actions are modified by a Note that allows one train to be bypassed for up to ~~8~~ 4 hours for surveillance testing, provided the other train is OPERABLE. This allowance is based upon the reliability analysis assumption of Reference ~~8~~ 7 that 4 hours is the average time required to perform channel surveillance.

D.1, D.2.1, and D.2.2

Condition D is applicable to:

- Containment Pressure – High ~~D~~ B;
- Pressurizer Pressure – Low ~~(two, three, and four loop units)~~ D;
- Steam Line Pressure – Low;
- ~~Steam Line Differential Pressure – High;~~ A

(continued)

BASES

ACTIONS

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

challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

The Required Actions are modified by a Note that allows ~~the~~ any single ~~inoperable~~ channel, including the inoperable channel, to be bypassed for up to ~~4~~ 4 hours for surveillance testing, ~~of other channels~~. The 6 hours allowed to restore channel to OPERABLE status or to place the inoperable channel in the tripped condition and the 4 hours allowed for testing is justified in Reference ~~8~~ 7. (C) (4)

E.1, E.2.1, and E.2.2

Condition E is applicable to:

- Containment Spray Containment Pressure – High ~~3 (High, High) (two, three, and four loop units), and~~
- Containment Phase B Isolation Containment Pressure – ~~High 3~~ High, High.

None of these signals has input to a control function. Thus two out of three logic is necessary to meet acceptable protective requirements. However, a two out of three design would require tripping a failed channel. This is undesirable because a single failure would then cause spurious containment spray initiation. Spurious spray actuation is undesirable because of the cleanup problems presented. Therefore, these channels are designed with two out of four logic so that a failed channel may be bypassed rather than tripped. Note that one channel may be bypassed and still satisfy the single failure criterion. Furthermore, with one channel bypassed, a single instrumentation channel failure will not spuriously initiate containment spray.

In order to avoid the inadvertent actuation of containment spray and Phase B containment isolation, the inoperable channel should not be placed in the tripped condition. Instead it is bypassed. Restoring the channel to OPERABLE status or placing the inoperable channel in the bypass condition within 6 hours is sufficient to assure that the Function remains OPERABLE and minimizes the time that the Function may be in a partial trip condition (assuming the inoperable channel has failed high), and is further

(continued)

BASES

ACTIONS

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

justified based on the low probability of an event occurring during this interval. Failure to restore the inoperable channel to OPERABLE status or place it in the bypassed condition within 6 hours requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

The Required Actions are modified by a Note that allows one additional channel to be bypassed for up to [4] hours for surveillance testing. Placing a second channel in the bypass condition for up to 4 hours for testing purposes is acceptable based on the results of Reference ~~8~~ 7. 4

F.1, F.2.1, and F.2.2

Condition F is applicable to:

- Manual Initiation of Steam Line Isolation;
- Loss of Offsite Power;
- Auxiliary Feedwater Pump Suction Transfer on Suction Pressure – Low; and
- P-4 Interlock.

For the Manual Initiation and the P-4 Interlock Functions, this action addresses the train orientation of the SSPS. For the Loss of Offsite Power function, this action recognizes the lack of manual trip provision for a failed channel. For the AFW System pump suction transfer channels, this action recognizes that placing a failed channel in trip during operation is not necessarily a conservative action. Spurious trip of this function could align the AFW System to a source that is not immediately capable of supporting pump suction. If a train or channel is inoperable, 48 hours is allowed to return it to OPERABLE status. The specified Completion Time is reasonable considering the nature of these Functions, the available redundancy, and the low probability of an event occurring during this interval. If the Function cannot be returned to OPERABLE status, the unit must be placed in MODE 3 within the next 6 hours and MODE 4

(continued)

BASES

ACTIONS

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

within the following 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power in an orderly manner and without challenging unit systems. In MODE 4, the unit does not have any analyzed transients or conditions that require the explicit use of the protection Functions noted above.

G.1, G.2.1 and G.2.2

Condition G is applicable to the automatic actuation logic and actuation relays for the Steam Line Isolation and AFW actuation Functions.

The action addresses the train orientation of the SSPS and the master and slave relays for these functions. If one train is inoperable, 6 hours are allowed to restore the train to OPERABLE status. The Completion Time for restoring a train to OPERABLE status is reasonable considering that there is another train OPERABLE and the low probability of an event occurring during this interval. If the train cannot be returned to an OPERABLE status, the unit must be brought to MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. Placing the unit in MODE 4 removes all requirements for OPERABILITY of the protection channels and actuation functions. In this MODE, the unit does not have analyzed transients or conditions that require the explicit use of the protection functions noted above.

The Required Actions are modified by a Note that allows one train to be bypassed for up to ~~14~~ 7 hours for surveillance testing provided the other train is OPERABLE. This allowance is based upon the reliability analysis (Ref. ~~8~~ 7) assumption that 4 hours is the average time required to perform channel surveillance.

(continued)

BASES

ACTIONS

I.1 and I.2 (continued)

restore the inoperable channel to OPERABLE status or place it in the tripped condition within 6 hours requires the unit to be placed in MODE 3 within the following 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. In MODE 3, these Functions are no longer required OPERABLE.

C any single, including the inoperable channel,

The Required Actions are modified by a Note that allows the ~~inoperable~~ channel to be bypassed for up to ~~2~~ 4 hours for surveillance testing of ~~other channels~~. The 6 hours allowed to place the inoperable channel in the tripped condition and the 4 hours allowed for a second channel to be in the bypassed condition for testing are justified in Reference ~~8~~ 7.

J.1 and J.2

Condition J is applicable to the AFW pump start on trip of all MFW pumps.

This action addresses the train orientation of the SSPS for the auto start function of the AFW System on loss of all MFW pumps. The OPERABILITY of the AFW System must be assured by allowing automatic start of the AFW System pumps. If a channel is inoperable, 48 hours are allowed to return it to an OPERABLE status. If the function cannot be returned to an OPERABLE status, 6 hours are allowed to place the unit in MODE 3. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. In MODE 3, the unit does not have any analyzed transients or conditions that require the explicit use of the protection Function noted above. The allowance of 48 hours to return the train to an OPERABLE status ~~and 6 hours to reach MODE 3~~ is justified in Reference ~~8~~ 7.

E

4

(continued)

BASES

ACTIONS (continued)

K.1, K.2.1 and K.2.2

Condition K is applicable to:

- ~~RWST Level – Low Low Coincident with Safety Injection and~~,
- RWST Level – Low Low Coincident with Safety Injection and Coincident with Containment Sump Level – High.

RWST Level – Low Low Coincident With SI and Coincident With Containment Sump Level – High provides actuation of switchover to the containment sump. Note that this Function requires the bistables to energize to perform their required action. The failure of up to two channels will not prevent the operation of this Function. However, placing a failed channel in the tripped condition could result in the premature switchover to the sump, prior to the injection of the minimum volume from the RWST. Placing the inoperable channel in bypass results in a two out of three logic configuration, which satisfies the requirement to allow another failure without disabling actuation of the switchover when required. Restoring the channel to OPERABLE status or placing the inoperable channel in the bypass condition within 6 hours is sufficient to ensure that the Function remains OPERABLE and minimizes the time that the Function may be in a partial trip condition (assuming the inoperable channel has failed high). The 6 hour Completion Time is justified in Reference 8. If the channel cannot be returned to OPERABLE status or placed in the bypass condition within 6 hours, the unit must be brought to MODE 3 within the following 6 hours and MODE 5 within the next 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 5, the unit does not have any analyzed transients or conditions that require the explicit use of the protection Functions noted above.

The Required Actions are modified by a Note that allows placing a second channel in the bypass condition for up to ~~24~~ 4 hours for surveillance testing. The total of 12 hours to reach MODE 3 and 4 hours for a second channel to be bypassed is acceptable based on the results of Reference ~~8~~ 7.

(continued)

BASES

ACTIONS (continued)

L.1, L.2.1 and L.2.2

A

Condition L is applicable to the P-11 ~~and P-12~~ interlock~~s~~.

With one channel inoperable, the operator must verify that the interlock is in the required state for the existing unit condition. This action manually accomplishes the function of the interlock. Determination must be made within 1 hour. The 1 hour Completion Time is equal to the time allowed by LCO 3.0.3 to initiate shutdown actions in the event of a complete loss of ESFAS function. If the interlock is not in the required state (or placed in the required state) for the existing conditions, the unit must be placed in MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. Placing the unit in MODE 4 removes all requirements for OPERABILITY of these interlocks.

M.1 and M.2

Condition M is applicable to the P-14 interlock.

The actions for Condition M are identical to those for Condition L except that the P-14 interlock is not required to be OPERABLE in MODE 3. Therefore, shutdown to MODE 3 within 7 hours is required if interlock status cannot be verified within 1 hour. The Completion Times are reasonable, based on operating experience, to reach the required MODE from full power in an orderly manner and without challenging unit systems.

C

INSERT N + O →

SURVEILLANCE REQUIREMENTS

The SRs for each ESFAS Function are identified by the SRs column of Table 3.3.2-1.

A Note has been added to the SR Table to clarify that Table 3.3.2-1 determines which SRs apply to which ESFAS Functions.

Note that each channel of process protection supplies both trains of the ESFAS. When testing channel I, train A and

(continued)

INSERT
Bases discussions for Conditions N and O

N.1.1, N.1.2, and N.2

Condition N is applicable to the SG Water Level Low-Low Function.

A known channel inoperable, must be restored to OPERABLE status, or placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one out of two logic for actuation of the two out of three trip. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 7.

If a channel fails, it is placed in the tripped condition and does not affect the TTD setpoint calculations for the remaining OPERABLE channels. It is then necessary for the operator to force the use of the shorter TTD Time Delay by adjustment of the single SG time delay calculation ($T_s$) to match the multiple SG time delay calculation ($T_m$) for the affected protection set, through the Man-Machine Interface.

If the inoperable channel cannot be restored or placed in the tripped condition within the specified Completion Time, the unit must be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours. The allowed Completion Times are reasonable, based on operating experience, to place the unit in MODE 3 from MODE 1 full power conditions in an orderly manner and without challenging unit systems.

The Required Actions have been modified by a Note which allows placing any single channel, including the inoperable channel, in the bypassed condition for up to 4 hours while performing routine Surveillance testing. The 4-hour time limit is justified in Reference 7.

O.1 and O.2

Condition O applies to the Vessel ΔT Equivalent to Power Function.

Failure of the vessel ΔT channel input (failure of more than one $T_H$ RTD or failure of both $T_C$ RTDs) ~~does not~~ will affect the TTD calculation for a protection set. This results in the requirement that the operator adjust the threshold power level for zero seconds time delay from 50% RTP to 0% RTP, through the Man-Machine Interface.

If the inoperable channel cannot be restored or placed in the tripped condition within the specified Completion Time, the unit must be placed in a MODE where this Function is not required to be OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is a reasonable time based on operating experience, to place the unit in MODE 3 from MODE 1 full power conditions in an orderly manner and without challenging unit systems.

BASES

SURVEILLANCE REQUIREMENTS (continued)

train B must be examined. Similarly, train A and train B must be examined when testing channel II, channel III, and channel IV (~~if applicable~~). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

~~Reviewer Note: Certain Surveillance Frequencies are based on approved topical reports. In order for a licensee to use these times the licensee must justify the Surveillance Frequencies as required by the staff SER for the topical report.~~

The protection Functions associated with the EAGLE-21 Process Protection System have an installed bypass capability, and may be tested in either the trip or bypass mode, as approved in Reference 7.

SR 3.3.2.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or even something more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and reliability. If a channel is outside the match criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. Thus, performance of the CHANNEL CHECK ensures that undetected overt channel failure is limited to 12 hours. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.2

SR 3.3.2.2 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested every 31 days on a STAGGERED TEST BASIS, using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissive, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and there is an intact voltage signal path to the master relay coils. The time allowed for the testing (4 hours) and the Frequency are justified in Reference ~~8~~ 7. 4

2

~~SR 3.3.2.3~~

~~SR 3.3.2.3 is the performance of an ACTUATION LOGIC TEST as described in SR 3.3.2.2, except that the semiautomatic tester is not used and the continuity check does not have to be performed as explained in the Note. This SR is applied to balance of plant actuation logic and relays that do not have the SSPS test circuits installed to utilize the semiautomatic tester or perform the continuity check. This test is also performed every 31 days on a STAGGERED TEST BASIS. The surveillance interval and the time allowed for the testing (4 hours) are justified in Reference 8.~~

SR 3.3.2.~~4~~ 3 2

SR 3.3.2.~~4~~ 3 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity. This test is performed every 31 days on a STAGGERED TEST BASIS. The time allowed for the testing (4 hours) and the surveillance interval are justified in Reference ~~8~~ 7.

4

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.~~5~~ 4 (2)

SR 3.3.2.~~5~~ 4 is the performance of a ~~A~~ ~~X~~COT. (5)

A~~X~~ ~~X~~COT is performed on each required channel to ensure the entire channel will perform the intended Function. Setpoints must be found within the Allowable Values specified in Table 3.3.1-1.

The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology. The setpoint shall be left set consistent with the assumptions of the current unit specific setpoint methodology.

The "as found" and "as left" values must also be recorded and reviewed for consistency with the assumptions of the surveillance interval extension analysis ~~(WCAP-10271)~~ (REFERENCE 7) when applicable. (A)

The Frequency of 92 days is justified in Reference ~~8~~ 7.

SR 3.3.2.~~6~~ 5 (2)

SR 3.3.2.~~6~~ 5 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation MODE is either allowed to function or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment that may not be operated in the design mitigation MODE is prevented from operation by the SLAVE RELAY TEST circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay. This test is performed every ~~18~~ 92 days. The time allowed for the testing (4 hours) and the Frequency are justified in Reference ~~8~~ 7. (4)

~~28~~ (14)

INSERT B3.3-56a

SR 3.3.2.~~7~~ 6 (2)

INSERT SR 3.3.2.6 from the next page (A)

~~SR 3.3.2.7 is the performance of a TADOT every 18 92 days. This test is a check of the Loss of Offsite Power, and~~

INSERT B3.3-56b ~~28~~ (14)

(continued)

INSERT B 3.3-56a

This SR is modified by a Note, which states that performance of this test is not required for those relays tested by SR 3.3.2.7.

INSERT SR 3.3.2.6

SR 3.3.2.6 is the performance of a TADOT every 92 days. This test is a check of Function 6.d., the Loss of Offsite Power, and Functions 6.f. and 6.g., AFW Pump Suction Transfer on Sunction Pressure--Low, for Motor-driven and Turbine-driven pumps respectively. These Functions are tested up to, and including, the master transfer relay coils.

The SR is modified by a Note that excludes verification of setpoints for relays. Relay setpoints are verified during a CHANNEL CALIBRATION.

INSERT B 3.3-56b

SR 3.3.2.7

SR 3.3.2.7 is the performance of a SLAVE RELAY TEST for slave relays K603A, K603B, K604A, K604B, K609A, K609B, K625A, and K625B. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment which may be operated in the design mitigation MODE is either allowed to function or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment which may not be operated in the design mitigation MODE is prevented from operation by the slave relay test circuit. For this letter case, contact operation is verified by a continuity check of the circuit containing the slave relay. This test is performed every 18 months. The time allowed for the testing (4 hours) is justified in Reference 7. The Frequency is justified by TVA correspondence to the NRC, dated November 9, 1984.

BASES

SURVEILLANCE REQUIREMENTS

~~SR 3.3.2.7 (continued)~~ and AFW Pump Suction Transfer on Suction Pressure - Low

~~Undervoltage RCP Functions. Each Function is tested up to, and including, the master transfer relay coils.~~

A

~~The test also includes trip devices that provide actuation signals directly to the SSPS. For these tests, the relay Trip Setpoints are verified and adjusted as necessary. The Frequency is justified in Reference 8.~~

14 ~~28~~ 8 11

SR 3.3.2.~~8~~ 7 2

SR 3.3.2.~~8~~ 7 is the performance of a TADOT. This test is a check of the Manual Actuation Functions and AFW pump start on trip of all MFW pumps. It is performed every [18] months. Each Manual Actuation Function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.). The Frequency is justified in Reference ~~8~~ 7. 4

14 ~~28~~ 9 8

SR 3.3.2.~~9~~ 8 2

SR 3.3.2.~~9~~ 8 is the performance of a CHANNEL CALIBRATION.

A CHANNEL CALIBRATION is performed every [18] months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATION measurement and setpoint error determination and readjustment must be performed consistent with the assumptions of the ~~unit specific~~ WATTS BAR setpoint methodology. The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology. 1

The Frequency of ~~[18]~~ months is based on the assumption of an ~~[18]~~ month calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.~~9~~ 8 9 (28) (14) (2) (continued)

This SR is modified by a Note, which states that this test should include verification that the time constants are adjusted to the prescribed values where applicable.

SR 3.3.2.~~10~~ 10 9 (14) (28) (2)

This SR ensures the individual channel (4) ESF RESPONSE TIMES are less than or equal to the maximum values assumed in the accident analysis. Response Time testing acceptance criteria are included in Reference ~~9~~.8 Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the Trip Setpoint value at the sensor, to the point at which the equipment in both trains reaches the required functional state (e.g., pumps at rated discharge pressure, valves in full open or closed position).

For channels that include dynamic transfer functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer functions set to one with the resulting measured response time compared to the appropriate FSAR response time. Alternately, the response time test can be performed with the time constants set to their nominal value provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

ESF RESPONSE TIME tests are conducted on an ~~5~~18~~7~~ month STAGGERED TEST BASIS. Testing of the final actuation devices, which make up the bulk of the response time, is included in the testing of each channel. The final actuation device in one train is tested with each channel. Therefore, staggered testing results in response time verification of these devices every ~~5~~18~~7~~ months. The ~~5~~18~~7~~ month Frequency is consistent with the typical refueling cycle and is based upon unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.~~10~~ 9 (continued)

This SR is modified with a Note that clarifies that the turbine driven AFW pump is tested within 24 hours after reaching ~~[1000]~~ 1092 psig in the SGs.

SR 3.3.2.~~15~~ 10

SR 3.3.2.~~15~~ 10 is the performance of a TADOT as described in SR 3.3.2.~~8~~ 7, except that it is performed for the P-4 Reactor Trip Interlock, and the Frequency is once per RTB cycle. This Frequency is based upon operating experience that undetected failure of the P-4 interlock sometimes occurs when the RTB is cycled.

REFERENCES

INSERT (see next page)

1. ~~FSAR, Chapter [6].~~ WATTS BAR UPDATED FSAR SECTION 6 "ENGINEERING SAFETY FEATURES."
2. ~~FSAR, Chapter [7].~~ WATTS BAR UPDATED FSAR, SECTION 7 "INSTRUMENTATION AND CONTROLS."
3. ~~FSAR, Chapter [15].~~ WATTS BAR UPDATED FSAR, SECTION 15, "ACCIDENT ANALYSES"
4. IEEE-279-1971, April 5, 1972.
5. 10 CFR 50.49, ~~[dated].~~
6. WATTS BAR RTS/ESFAS Setpoint Methodology Study.
7. ~~NUREG-1218, April 1988.~~
7 ~~8~~. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.
8 ~~9~~. Technical Requirements Manual, Section ~~15~~ 3.3.2, "Response Times."

REFERENCES INSERT


1. Watts Bar Updated FSAR, Section 6.0, "Engineered Safety Features."

2. Watts Bar Updated FSAR, Section 7.0, "Instrumentation and Controls."

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

4. Institute of Electrical and Electronic Engineers, IEEE-279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," April 5, 1972.

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

6. WCAP-12096, Rev. 4, "Westinghouse Setpoint Methodology for Protection System, Watts Bar 1 and 2," November 1990.

7. WCAP-10271-P-A, Supplement 1, Rev. 1 and Supplement 2, Rev. 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," dated May 1986 and June 1990.

8. Watts Bar Technical Requirements Manual, Section 3.3.2, Engineered Safety Feature Response Times.

# JUSTIFICATION FOR DEVIATIONS FROM RSTS 3.3.2

## Generic Justifications

A. Modified to reflect WBN plant specific design.

B. Modified to reflect WBN plant specific terminology.

C. Modified to reflect changes due to the EAGLE-21 Microprocessor Based Process Protection System used at WBN.

D. Deletion or modification of information which is generic in nature and has been modified to be plant specific to WBN.

E. This change reflects comments made by the industry to the NRC at the RSTS P&R review meeting held in Irvine, CA July 13-20, 1992.

F. Editorial change for clarification.

## Specific Justifications

1. This item was a correction to the text.

2. Modified to reflect WBN plant specific design. This change creates a renumbering of all surveillance after SR 3.3.2.2.

3. This paragraph was deleted because it is a repeat of the paragraph above.

4. Reference 7 has been deleted since the information referred to Reference 7 is not applicable to WBN and has been deleted. This causes a renumbering of all references from 7 on.

5. ANALOG CHANNEL OPERATIONAL TEST (ACOT) has been redefined to include both analog and digital testing, and is now known as CHANNEL OPERATIONAL TEST (COT).

6. Specific titles have been added to Conditions I, N, and O because each condition only applies to one function. This addition makes Condition I, N, and O consistent with other "one function" conditions.

7. Changes have been made to Pages B 3.3-1, B 3.3.-2, and B 3.3-2 to clarify the technical definition of ESFAS instrumentation.

8. An additional justification is provided for the Completion Time for Condition B. Reference 7 applies to manual initiation.

9. Change to specify WBN specific reference information and format.

10. These SRs are TADOTs that apply to undervoltage relays, underfrequency relays, manual functions, RCP breaker position, or SI input to RTS. The setpoint is not verified when performing a TADOT. These devices are simply triggered and actuation verified. Setpoint verification requires elaborate bench calibration and is done every 18 months.

    SR 3.3.2.7 applies to functions other than relays, so relays are specified in the Note.

    These changes are consistent with SR 3.3.1.15.

11. Redundant material has been deleted.

12. This is the correct Watts Bar number according to the Westinghouse PLS document, Page 14 1.B.i.1.A.6.

13. The 12 hour CHANNEL CHECK has been deleted from Functions 6.f. and 6.g. (motor and turbine - driven AFW pump suction transfer on suction pressure - low). A CHANNEL CHECK cannot be performed on process bistables.

14. SR 3.3.2.7 has been added to the Surveillance table. This SR is an 18 month SLAVE RELAY TEST for certain relays for SI and Phase B Containment Isolation. Testing of these slave relays could cause unstable or unsafe plant conditions. Details for each slave relay are explained in the TVA correspondence to the NRC, dated November 9, 1984.

    A Note has been added to SR 3.3.2.5 to exclude the SI and Containment Isolation that are tested by SR 3.3.2.7.

    The Surveillance table, Function table, and Bases have been renumbered.

## 3.3 INSTRUMENTATION

### 3.3.3 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.3 The PAM instrumentation for each Function in Table 3.3.3-1 shall be OPERABLE.

(8) ----------NOTE----------
Source Range Neutron Flux is required OPERABLE below P-6 (Intermediate Range Neutron Flux) interlocks.
---

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

1. LCO 3.0.4 is (C) not applicable.

2. Separate Condition entry is allowed for each Function.

---

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more Functions with one required channel inoperable. | A.1 Restore required channel to OPERABLE status. | 30 days |
| B. Required Action and associated Completion Time of Condition A not met. | B.1 Initiate action in accordance with Specification 5.~~4~~.2.c. ~~8~~ 9 (D) | Immediately |
| C. ---------NOTE--------- Not applicable to hydrogen monitor channels. --- One or more Functions with two or more required channels inoperable. | C.1 Restore one channel to OPERABLE status. | 7 days |

Replace with INSERT 3.3-1a, see next page. (9)

(continued)

## 3.3 INSTRUMENTATION

### 3.3.3 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.3 The PAM instrumentation for each Function in Table 3.3.3-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

---NOTE---
Source Range Neutron Flux is required OPERABLE below P-6 (Intermediate Range Neutron Flux) interlocks.
---

ACTIONS

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

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more Functions with one or more required channels inoperable. | A.1 Enter the Condition referenced in Table 3.3.3-1 for the required channel(s). | Immediately |
| B. One channel inoperable. | B.1 Restore channel to OPERABLE status. | 7 days |
| C. One channel inoperable. | C.1 Restore channel to OPERABLE status. | 30 days |

(continued)

INSERT 3.3-1a

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| D. Two hydrogen monitor channels inoperable. | D.1 Restore one hydrogen monitor channel to OPERABLE status. | 72 hours |
| E. Required Action and associated Completion Time of Condition C or D not met. | E.1 Enter the Condition referenced in Table 3.3.3-1 for the channel(s). C | Immediately |
| F. As required by Required Action E.1 and referenced in Table 3.3.3-1. | F.1 Be in MODE 3. | 6 hours |
| | AND | |
| | F.2 Be in MODE 4. | 12 hours |
| G. As required by Required Action E.1 and referenced in Table 3.3.3-1. | G.1 Initiate action in accordance with Specification 5.~~6~~.2.c. ~~8~~ 9 D | Immediately |

Replace with INSERT 3.3-2a, see next page. 9

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| D. Required Action and associated Completion Time of Condition C not met. | D.1 | Initiate action in accordance with Specification 5.9.2.c. | Immediately |
| E. Two or more channels inoperable. | E.1 | Restore one channel to OPERABLE status. | 7 days |
| F. Two hydrogen monitor channels inoperable. | F.1 | Restore one hydrogen monitor channel to OPERABLE status. | 72 hours |
| G. Two channels inoperable. | G.1 | Restore one channel to OPERABLE status. | 7 days |
| H. Required Action and associated Completion Time of Condition G not met. | H.1 | Initiate action in accordance with Specification 5.9.2.c. | Immediately |
| I. Required Action and associated Completion Time of Conditions B, E, or F not met. | I.1 | Be in MODE 3. | 6 hours |
| | AND | | |
| | I.2 | Be in MODE 4. | 12 hours |

INSERT 3.3-2a

SURVEILLANCE REQUIREMENTS

----------NOTE----------
SR 3.3.3.1 and SR 3.3.3.3 apply to each PAM instrumentation Function in Table 3.3.3-1.
---

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.3.3.1 (E) | Perform CHANNEL CHECK for each required instrumentation channel that is normally energized. | 31 days |
| SR 3.3.3.2 | Perform ACOT on hydrogen monitor channels. | ~~92~~ 184 days (10) |
| SR 3.3.3.~~3~~2 (E) | ----------NOTE---------- Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION. | [18] months |

Table 3.3.3-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

| FUNCTION | REQUIRED CHANNELS | CONDITION ~~REFERENCED FROM REQUIRED ACTION B.1~~ (9) |
|---|---|---|
| 1. Intermediate ~~Power~~ Range Neutron Flux(d) | 2 | C, E |
| 2. Source Range Neutron Flux | 2 | C, E |
| 3. Reactor Coolant System (RCS) Hot Leg Temperature | 1 ~~2~~ per loop | B |
| 4. RCS Cold Leg Temperature | 1 ~~2~~ per loop | B |
| 5. RCS Pressure (Wide Range) | ~~2~~ 3 | C, E |
| 6. Reactor Vessel Water Level(c)(d) | 2 | C, G |
| 7. Containment Sump Water Level (Wide Range) | 2 | C, E |
| 8. Containment Lower Comp Atm. Temperature | 2 | C, E |
| 9. ~~8.~~ Containment Pressure (Wide Range)(d) | 2 | |
| 10. Containment Pressure (Narrow Range) | ~~4~~ 3 | |
| 11. ~~9.~~ Containment Isolation Valve Position(d) | 1 per valve(a) | B |
| 12. ~~10.~~ Containment ~~Area~~ Radiation (High Range) | 2 | C, G |
| 13. ~~11.~~ Hydrogen Monitors | 2 | C, F |
| 14. ~~12.~~ RCS Pressurizer Level | ~~2~~ 3 | C, E |
| 15. ~~13.~~ Steam Generator Water Level (Wide Range)(d) | ~~2~~ per steam generator | C, E |
| 16. ~~14.~~ Steam Generator Water Level (Narrow Range) | ~~2~~ per steam generator | C, E |
| 17. ~~14.~~ ~~Condensate Storage Tank Level~~ AFW valve status(d) | ~~2~~ 1 per valve | C, E |
| 18. ~~15.~~ Core Exit Temperature – Quadrant 1(c) | 2(b) | B |
| 19. ~~16.~~ Core Exit Temperature – Quadrant 2(c) | 2(b) | C, E |
| 20. ~~17.~~ Core Exit Temperature – Quadrant 3(c) | 2(b) | |
| 21. ~~18.~~ Core Exit Temperature – Quadrant 4(c) | 2(b) | |
| 22. ~~19.~~ Auxiliary Feedwater Flow | 2 | |

Insert 1

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

(b) A channel consists of two or more core exit thermocouples (CETs).

(c) The ICCM provides these functions on a plasma display

~~NOTE: Table 3.3.3-1 shall be amended for each unit as necessary to list:~~

~~(1) All Regulatory Guide 1.97, Type A instruments, and~~

~~(2) All Regulatory Guide 1.97, Category I, non Type A instruments in accordance with the unit's Regulatory Guide 1.97, Safety Evaluation Report.~~

(d) Regulatory Guide 1.97, non-Type A, Category I variables.

Insert 1

| | | | |
|---|---|---|---|
| 23. | Reactor Coolant System Subcooling Margin Monitor[c] | 2 | C, E |
| 24. | Refueling Water Storage Tank Level | 2 | F |
| 25. | Steam Generator Pressure | 2 per steam generator | F |
| 26. | Auxiliary Building Passive Sump Level[d] | 2 | G |
| 27. | Steam Line Radiation Monitor | 1 per steam generator | G B |

# B 3.3 INSTRUMENTATION

## B 3.3.3 Post Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

The primary purpose of the PAM Instrumentation is to display unit variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for Design Basis Accidents (DBAs).

The OPERABILITY of the accident monitoring instrumentation ensures that there is sufficient information available on selected unit parameters to monitor and to assess unit status and behavior following an accident.

The availability of accident monitoring instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. These essential instruments are identified by unit specific documents (Ref. ~~1~~ 2) addressing the recommendations of Regulatory Guide 1.97 (Ref. ~~2~~ 1) as required by Supplement 1 to NUREG-0737, "TMI Action Items" (Ref. 3).

The instrument channels required to be OPERABLE by this LCO include two classes of parameters identified during unit specific implementation of Regulatory Guide 1.97 as Type A and Category I variables.

Type A variables are included in this LCO because they provide the primary information required for the control room operator to take specific manually controlled actions for which no automatic control is provided, and which are required for safety systems to accomplish their safety functions for DBAs. Because the list of Type A variables differs widely between units, Table 3.3.3-1 in the accompanying LCO contains no examples of Type A variables, except for those that may also be Category I variables.

A

Those Type A variables listed in Table 3.3.3-1 ~~may~~ are ~~also be~~ Category I variables.

(continued)

BASES

BACKGROUND (continued)

Category I variables (types B and D) are the key variables deemed risk significant because they are needed to: (A)

- Determine whether other systems important to safety are performing their intended functions;

(A) (D)

INSERT B3.3-2

- Provide information to the operators that will enable them to determine the likelihood of a gross breach of the barriers to radioactivity release; and
- Provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public and to estimate the magnitude of any impending threat.

(C) non-Type A These key variables are identified by unit specific Regulatory Guide 1.97 analyses. These analyses also identify the unit specific Type A variables and provide justification for deviating from the NRC proposed list of Category I ~~key~~ variables.

~~Reviewers Note: Table 3.3.3-1 provides a list of variables typical of those identified by unit specific Regulatory Guide 1.97 analyses. Table 3.3.3-1 in unit specific Technical Specifications (TS) shall list all Type A and Category I variables identified by the unit specific Regulatory Guide 1.97 analyses, as amended by the NRC's Safety Evaluation Report (SER).~~

The specific instrument Functions listed in Table 3.3.3-1 are discussed in the LCO section.

APPLICABLE SAFETY ANALYSES

(C) The PAM instrumentation (Ref. 1) ensures the operability of Regulatory Guide 1.97 Type A and Category I key variables so that the control room operating staff can: (C)

- Perform the diagnosis specified in the emergency operating procedures (these variables are restricted to preplanned actions for the primary success path of DBAs ~~, and~~ e.g., ~~loss of coolant accident~~ (LOCA); and (C)
- Take the specified, ~~preplanned,~~ pre-planned manually controlled actions, for which no automatic control is provided,

(continued)

INSERT B.3.3-2

- Provide information to indicate the operation of individual safety systems and other plant systems. These variables are to help the operator make appropriate decisions in using the individual systems in mitigating the consequences of an accident.

APPLICABLE SAFETY ANALYSES (continued)

that are required for safety systems to accomplish their safety function;

- Determine whether systems important to safety are performing their intended functions;
- Determine the likelihood of a gross breach of the barriers to radioactivity release;
- Determine if a gross breach of a barrier has occurred; and
- Initiate action necessary to protect the public and to estimate the magnitude of any impending threat.

PAM instrumentation that meets the definition of Type A in Regulatory Guide 1.97 satisfies the requirements of Criterion 3 of the NRC Policy Statement. Category I PAM instrumentation must be retained in TS because they are intended to assist operators in minimizing the consequences of accidents (Ref. 2). Therefore, Category I variables are important in reducing public risk.

The PAM Instrumentation LCO provides OPERABILITY requirements for Regulatory Guide 1.97 Type A monitors, which provide information required by the control room operators to perform certain manual actions specified in the unit Emergency Operating Procedures. These manual actions ensure that a system can accomplish its safety function, and are credited in safety analyses. Additionally, this LCO addresses Regulatory Guide 1.97 instruments that have been designated Category I, non type A.

The OPERABILITY of the PAM instrumentation ensures there is sufficient information available on selected unit parameters to monitor and assess unit status following an accident. This capability is consistent with the recommendations of Reference 1.

LCO 3.3.3 requires two OPERABLE channels for most Functions. Two OPERABLE channels ensure no single failure prevents operators from getting the information necessary for them to determine the safety status of the unit, and to bring the

(continued)

INSERT A

Another exception to the two channel requirement is RCS hot leg and cold leg temperature. One channel is sufficient because the loop temperatures are normally similar in value and there is other adequate instrumentation to verify abnormal readings in one channel.

A third exception is the steam generator water level wide range. One channel is sufficient because the wide range levels are back up measurements for the narrow range indication (three channels) and auxiliary feedwater flow (two channels).

A forth exception is AFW valve status. This is acceptable for reasons similar to containment isolation valve status.

A fifth exception is steamline radiation monitors. One channel is sufficient because there are other monitors available to verify abnormal readings in one channel.

BASES

LCO (continued)

1, 2. Intermediate Range ~~Power Range~~ and Source Range Neutron Flux

(1)

Intermediate ~~Power~~ Range and Source Range Neutron Flux indication is provided to verify reactor shutdown. The two ranges are necessary to cover the full range of flux that may occur post accident.

Neutron flux is used for accident diagnosis, verification of subcriticality, and diagnosis of positive reactivity insertion.

3, 4. Reactor Coolant System (RCS) Hot and Cold Leg Temperature

(C)

RCS Hot and Cold Leg Temperatures are Category I variables provided for verification of core cooling and long term surveillance.

(D)

(and/or reactor vessel water level)

is used to make decisions to

RCS hot and cold leg temperatures are used to determine RCS subcooling margin. RCS subcooling margin ~~will allow~~ terminat~~ion of~~ safety injection (SI), if still in progress, or reinitiat~~ion of~~ SI if it has been stopped. RCS subcooling margin is also used for unit stabilization and cooldown control.

(6)

In addition, RCS cold leg temperature is used in conjunction with RCS hot leg temperature to verify the unit conditions necessary to establish natural circulation in the RCS.

(A) (D) (D)

Reactor Trip System (RTS)

Reactor outlet temperature inputs to the ~~Reactor Protection System~~ are provided by ~~two~~ one fast response resistance elements and associated transmitters in each loop. The channels provide indication over a range of ~~32°~~ 50°F to 700°F.

(D)

5. Reactor Coolant System Pressure (Wide Range)

(C)

RCS wide range pressure is a Category I variable provided for verification of core cooling and RCS integrity long term surveillance.

(C)

Insert B

RCS pressure is used to verify delivery of SI flow to RCS from at least one train when the RCS pressure is

(D)

(continued)

INSERT B

Wide-range RCS loop pressure is measured by pressure transmitters with a span of 0-3000 psig. Redundant monitoring capability is provided by two trains of instrumentation. Control room indications are provided by panel meters and through the inadequate core cooling monitoring (ICCM) plasma display.

BASES

LCO

5. Reactor Coolant System Pressure (Wide Range) (continued)

below the pump shutoff head. RCS pressure is also used to verify closure of manually closed spray line valves and pressurizer power operated relief valves (PORVs).

In addition to these verifications, RCS pressure is used for determining RCS subcooling margin. RCS subcooling margin will allow termination of SI, if still in progress, or to reinitiation of SI if it has been stopped. RCS pressure can also be used:

- to determine whether to terminate actuated SI or to reinitiate stopped SI;
- to determine when to reset SI and shut off low head SI;
- to manually restart low head SI; A
- as reactor coolant pump (RCP) trip criteria;
- to make a determination on the nature of the accident in progress and where to go next in the procedure.

RCS subcooling margin is also used for unit stabilization and cooldown control.

RCS pressure is also related to three decisions about depressurization. They are:

- to determine whether to proceed with primary system depressurization;
- to verify termination of depressurization; and
- to determine ~~whether~~ C when to close accumulator isolation valves during a controlled cooldown/depressurization.

A final use of RCS pressure is to determine whether to operate the pressurizer heaters.

(continued)

BASES

LCO

5. Reactor Coolant System Pressure (Wide Range) (continued)

C

~~In some units,~~ RCS pressure is a Type A variable because the operator uses this indication to monitor the cooldown of the RCS following a steam generator tube rupture (SGTR) or small break LOCA. Operator actions to maintain a controlled cooldown, such as adjusting steam generator (SG) pressure or level, would use this indication. Furthermore, RCS pressure is one factor that may be used in decisions to terminate RCP operation.

6. Reactor Vessel Water Level

Reactor Vessel Water Level is provided for verification and long term surveillance of core cooling. It is also used for accident diagnosis and to determine reactor coolant inventory adequacy.

~~The Reactor Vessel Water Level Monitoring System provides a direct measurement of the collapsed liquid level above the fuel alignment plate. The collapsed level represents the amount of liquid mass that is in the reactor vessel above the core. Measurement of the collapsed water level is selected because it is a direct indication of the water inventory.~~

Insert C

7. Containment Sump Water Level (Wide Range)

A

Containment Sump Water Level is provided for verification and long term Surveillance of RCS integrity.

Containment Sump Water Level is used to determine:

A

INSERT B3.3-7a

- containment sump level accident diagnosis;
- when to begin the recirculation procedure; and
- whether to terminate SI, if still in progress.

(continued)

## INSERT C

The Reactor Vessel Level Indicating System (RVLIS) provides a direct measurement of the liquid level above the bottom of the reactor vessel. Indication is in percent of this distance (i.e., the reactor vessel bottom is 0% and the vessel top is 100%). It also has a dynamic range vessel liquid content (% LIQ) normalized from 0% to 100%. Normalization corrects the transmitted level information for the RCP operational configuration so that the accurate dynamic % LIQ is indicated regardless of the pattern of pumps running, and the resulting fluid density or void content.

## INSERT B 3.3-7a

- Verify water source for recirculation mode of ECCS operation after a LOCA.
- Determine whether high energy line rupture has occurred inside or outside containment.

BASES

Insert M

LCO (continued)

9 ~~8~~. Containment Pressure (Wide Range)

Insert E

A

Containment Pressure (Wide Range) is provided for verification of RCS and containment OPERABILITY. monitor containment pressure during an accident

Containment pressure is used to verify closure of main steam isolation valves (MSIVs), and containment spray Phase B isolation when High-3 containment pressure is reached.

Insert N

B -- High-High

11 ~~9~~. Containment Isolation Valve Position

CIV Position is provided for verification of Containment OPERABILITY, and Phase A and Phase B isolation. A

When used to verify Phase A and Phase B isolation, the important information is the status of the containment penetrations. The LCO requires one position indicator for each active CIV. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve and prior knowledge of passive valve, or system boundary status. If a normally active CIV is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE. This Function is on a per valve basis and Condition A is entered separately for each inoperable valve indication. A Note to the Required Channels states that the Function is not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. A

12 ~~10~~. Containment ~~Area~~ Radiation (High Range)

B

Containment ~~Area~~ Radiation is provided to monitor for the potential of significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans.

(continued)

INSERT E

Containment Pressure instrumentation consists of two recorded channels on separate power supplies with a range of -5 to +60 psig.

INSERT M

8. Containment Lower Compartment Atmospheric Temperature

The lower compartment temperature monitors will verify the temperatures in the lower compartment after an accident with display in the main control room. The monitoring system consists of two channels with range 0°F to 350°F.

INSERT N

10. Containment Pressure (Narrow Range)

Containment Pressure (Narrow Range) is provided to determine margin to containment design pressure. The narrow range monitors are also used to monitor containment conditions following a break inside containment and verify if the accident is properly controlled. The narrow range instrumentation has a range of -2 to +15 psid.

BASES

LCO

12 ~~10~~. Containment Area Radiation (High Range) (continued)

Containment radiation level is used to determine if a high energy line break (HELB) has occurred, and whether the event is inside or outside of containment.

13 ~~11~~. Hydrogen Monitors (A) a loss of reactor coolant or secondary coolant has occured.

Hydrogen Monitors are provided to detect high hydrogen concentration conditions which represent a potential for containment breach from a hydrogen explosion. This variable is also important in verifying the adequacy of mitigating actions.

Insert G

(A) 14 ~~12~~. Pressurizer Level one factor (a)

Pressurizer Level is used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Knowledge of pressurizer water level is also used to verify the unit conditions necessary to establish natural circulation in the RCS and to verify that the unit is maintained in a safe shutdown condition.

Insert H

(A) 15 ~~13~~, 16 Steam Generator (SG) Water Level ~~(Wide Range)~~

SG Water Level is provided to monitor operation of decay heat removal via the SGs. The Category I indication of SG level is the ~~extended startup~~ range level instrumentation. ~~The extended startup range level covers a span of ≥ 6 to ≤ 394 inches above the lower tubesheet. The measured differential pressure is displayed in inches of water at 68°F.~~ wide

(A) 12 to 554

~~Temperature compensation of this indication is performed manually by the operator. Redundant monitoring capability is provided by two trains of instrumentation. The uncompensated level signal is input to the unit computer, a control room indicator, and the Emergency Feedwater Control System.~~

(continued)

## INSERT G

Hydrogen concentration is also used to determine whether or not to start the hydrogen recombiners. Containment hydrogen instrumentation consists of two channels on separate power supplies with a range of 0-10% (by volume) hydrogen concentration.

## INSERT H

Pressurizer Level instrumentation consists of the three differential pressure transmitters and associated instrumentation used to measure pressurizer level. The channels provide indication over the entire distance between taps.

BASES

LCO

~~13~~ 15, 16 Steam Generator Water Level ~~(Wide Range)~~ (continued)

Insert I

~~SG Water Level (Wide Range) is used to:~~

- ~~identify the faulted SG following a tube rupture;~~
- ~~verify that the intact SGs are an adequate heat sink for the reactor;~~
- ~~determine the nature of the accident in progress (e.g., verify an SGTR); and~~
- ~~verify unit conditions for termination of SI during secondary unit HELBs outside containment.~~

A

~~At some units,~~ operator action is based in part on the control room indication of SG level. The RCS response during a design basis small break LOCA is dependent on the break size. For a certain range of break sizes, the boiler condenser mode of heat transfer is necessary to remove decay heat. Extended startup range level is a Type A variable because the operator must manually raise and control SG level to establish boiler condenser heat transfer. Operator action is initiated on a loss of subcooled margin. Feedwater flow is increased until the indicated extended startup range level reaches the ~~boiler condenser setpoint.~~ desired Level.

Insert O

D

14. Condensate Storage Tank (CST) Level

CST Level is provided to ensure water supply for auxiliary feedwater (AFW). The CST provides the ~~ensured~~ safety grade water supply for the AFW System. The CST consists of two identical tanks connected by a common outlet header. Inventory is monitored by a 0 to 144 inch level indication for each tank. CST Level is displayed on a control room indicator, strip chart recorder, and unit computer. In addition, a control room annunciator alarms on low level.

At some units, CST Level is considered a Type A variable because the control room meter and annunciator are considered the primary indication used by the operator.

(continued)

## INSERT I

Steam generator level (Narrow Range) may be used to help identify the faulted steam generator following a tube rupture and verify that the intact steam generators are an adequate heat sink for the reactor. Narrow range steam generator level is also needed to make a determination on the nature of the accident in progress, e.g., verify a steam generator tube rupture. Narrow range steam generator water level is used when verifying plant conditions for termination of SI during secondary plant high energy line breaks outside containment.

## INSERT O

17. <u>AFW Valve Status</u>

The status of each AFW valve is monitored with indication in the control room. There is one channel per valve which indicates fully open or fully closed position for each valve.

AFW valve status is monitored to give verification to the operator that automatic transfer to ERCW has taken place.

BASES

LCO

D

~~14. Condensate Storage Tank (CST) Level (continued)~~

~~The DBAs that require AFW are the loss of electric power, steam line break (SLB), and small break LOCA. The CST is the initial source of water for the AFW System. However, as the CST is depleted, manual operator action is necessary to replenish the CST or align suction to the AFW pumps from the hotwell.~~

A ~~15, 16, 17, 18.~~ 18, 19, 20, 21. Core Exit Temperature

INSERT B3.3-11a

Core Exit Temperature is provided for verification and long term surveillance of core cooling.

B ICCM detection

~~An evaluation was made of the minimum number of valid core exit thermocouples (CET) necessary for measuring core cooling. The evaluation determined the reduced complement of CETs necessary to detect initial core recovery and trend the ensuing core heatup. The evaluations account for core nonuniformities including incore effects of the radial decay power distribution and excore effects of condensate runback in the hot legs and nonuniform inlet temperatures. Based on these evaluations, adequate core cooling is ensured with two valid Core Exit Temperature channels per quadrant. with two CETs per required channel. The CET pair are oriented radially to permit evaluation of core radial decay power distribution.~~

ICCM detection

INSERT PG. B 3.3-11b A

~~(see next pg.)~~

Core Exit Temperature is used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Core Exit Temperature is also used for unit stabilization and cooldown control.

Two OPERABLE channels of Core Exit Temperature are required in each quadrant to provide indication of radial distribution of the coolant temperature rise across representative regions of the core. Power distribution symmetry was considered in determining the specific number and locations provided for diagnosis of local core problems. Therefore, two randomly selected thermocouples are not sufficient to meet the two thermocouples per channel requirement in any quadrant. The two thermocouples in each channel must meet the additional requirement that one is located near the center of the core and the other near

(continued)

INSERT B 3.3-11a

The ICCM is used to monitor the core exit thermocouples. There are two isolated systems, with each system monitoring at least four thermocouples per quadrant. The plasma display gives the average quadrant value, the high value, and the low value.

INSERT B 3.3-11b

Control room indications are provided through the ICCM plasma display, which is the primary indication used by the operator during an accident. Therefore, the accident monitoring specification deals specifically with this portion of the instrument channel.

BASES

LCO ~~15, 16, 17, 18.~~ 18, 19, 20, 21.

Core Exit Temperature (continued)

the core perimeter, such that the pair of Core Exit Temperatures indicate the radial temperature gradient across their core quadrant. Unit specific evaluations in response to Item II.F.2 of NUREG-0737 (Ref. 3) should have identified the thermocouple pairings that satisfy these requirements. Two sets of two thermocouples ensures a single failure will not disable the ability to determine the radial temperature gradient.

D

22 ~~19~~.

Auxiliary Feedwater Flow

AFW Flow is provided to monitor operation of decay heat removal via the SGs.

The AFW Flow to each SG is determined from a differential pressure measurement calibrated for a range of 0 to 1200 gpm. Redundant monitoring capability is provided by two independent trains of instrumentation for each SG. Each differential pressure transmitter provides an input to a control room indicator and the unit computer. Since the primary indication used by the operator during an accident is the control room indicator, the PAM specification deals specifically with this portion of the instrument channel.

A

AFW flow is used three ways:

- to verify ~~delivery of~~ AFW flow to the SGs;
- to determine whether to terminate SI if still in progress in conjunction with SG water level (narrow range);
- to regulate AFW flow so that the SG tubes remain covered.

~~At some units, AFW flow is a Type A variable because operator action is required to throttle flow during an SLB accident in order to prevent the AFW pumps from operating in runout conditions. AFW flow is also used by the operator to verify that the AFW System is~~

2

Insert L

(continued)

INSERT L

23. Reactor Coolant System Subcooling Margin Monitor

The RCS subcooling margin monitor is used to determine the temperature margin to saturation of the primary coolant. Control room indications are provided through the ICCM plasma display. The ICCM is the primary indication used by the operator during an accident.

24. Refueling Water Storage Tank Level

RWST water level is used to verify the water source availability to the ECCS and Containment Spray Systems. It may also provide an indication of time for initiating cold leg recirculation from the sump following a LOCA.

25. Steam Generator Pressure

Steam pressure is used to determine if a high energy secondary line rupture has occurred and the availability of the steam generators as a heat sink. It is also used to verify that a faulted steam generator is isolated. Steam pressure may be used to ensure proper cooldown rates or to provide a diverse indication for natural circulation cooldown.

26. Auxiliary Building Passive Sump Level

Auxiliary Building Passive Sump Level monitors the sump level in the auxiliary building during a LOCA. The purpose is to verify that radioactive water does not leak to the auxiliary building. The Auxiliary Building Passive Sump Level monitor consists of two channels on separate power supply. One channel is recorded. The calibrated range of the two monitors are 0" to 60".

27. Steam Line Radiation Monitor

The steam line radiation monitors are used to detect primary to secondary leakage and monitor radioactivity release.

Two channels are required to be OPERABLE for most functions. Two OPERABLE channels ensure no single failure prevents the operators from having the information necessary to determine the safety status of the plant and to bring the plant to and maintain it in a safe condition following an accident.

BASES

LCO

~~9.19. Auxiliary Feedwater Flow (continued)~~

~~delivering the correct flow to each SG. However, the primary indication used by the operator to ensure an adequate inventory is SG level.~~

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1, 2, and 3. These variables are related to the diagnosis and preplanned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6, unit conditions are such that the likelihood of an event that would require PAM instrumentation is low; therefore, the PAM instrumentation is not required to be OPERABLE in these MODES.

See INSERT B 3.3-13a

ACTIONS

Note 1 has been added in the ACTIONS to exclude the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require unit shutdown. This exception is acceptable due to the passive function of the instruments, the operator's ability to respond to an accident using alternate instruments and methods, and the low probability of an event requiring these instruments.

Note 2 has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.3-1. The Completion Time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

9

INSERT B3.3-13b + B3.3-13c

~~A.1~~

9

~~Condition A applies when one or more Functions have one required channel that is inoperable. Required Action A.1 requires restoring the inoperable channel to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channel (or in the case of a Function that has only~~

(continued)

INSERT B 3.3-13a

A Note modifies the APPLICABILITY to recognize that the Source Range Neutron Flux Channel is not required OPERABLE above the P-6 (Intermediate Range Neutron Flux) interlock.

A.1

Condition A is applicable to all PAM Instrumentation Functions. Condition A addresses the situation where one or more required channels for one or more functions are inoperable at the same time. The Required Action is to refer to Table 3.3.3-1 and to take the Required Actions for the Functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

B.1

Condition B applies when one channel is inoperable for those PAM instrumentation Functions that only require one channel. Required Action B.1 requires restoring one channel in the Function(s) to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring the operation of PAM instrumentation during this interval.

C.1

Condition C applies when one channel is inoperable for those PAM instrumentation Functions that require two or more channels. Required Action C.1 requires restoring the inoperable channel to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channel, the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the relatively low probability of an event requiring the operation of PAM instrumentation during this interval.

D.1

Condition D applies when the Required Action and associated Completion Time for Condition C are not met. This Required Action specifies immediate initiation of actions in Specification 5.9.2.c, "Special Reports," that require a written report, approved by the onsite review committee, to be submitted to the NRC. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative actions. This action is appropriate in lieu of a shutdown requirement since alternative actions are identified before loss of functional capability. Also, given the small likelihood of unit conditions that would require information provided by this instrumentation.

E.1

Condition E applies when two or more channels are inoperable in the same Function. Required Action E.1 requires restoring one channel in the Function(s) to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation and the availability of alternate means to obtain the required information. Continuous operation with two required channels inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur.

F.1

Condition F applies when two hydrogen monitor channels are inoperable. Required Action F.1 requires restoring one hydrogen monitor channel to OPERABLE status within 72 hours. The 72 hour Completion Time is reasonable based on the backup capability of the Post Accident Sampling System to monitor the hydrogen concentration for evaluation of core damage and to provide information for operator decisions. Also, it is unlikely that a LOCA (that would cause core damage) would occur during this time.

G.1

Condition G applies when two channels are inoperable for Containment Radiation High Range or Reactor Vessel Water Level. Required Action G.1 requires restoring one channel in the Function(s) to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation.

H.1

Alternate means of monitoring Reactor Vessel Water Level and Containment Area Radiation may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allotted time. If these alternate means are used, the Required Action is not to shut down the unit but to follow the directions of Specification 5.9.2.c, in the Administrative Controls section of the TS. The report provided to the NRC should discuss the alternate means used, describe the degree to which the alternate means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels.

I.1 and I.2

If the Required Action and associated Completion Time of Conditions B, E, or F are not met, the unit must be brought to a MODE where the requirements of this LCO do not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and MODE 4 within 12 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

ACTIONS

9 A.1 (continued)

one required channel, other non Regulatory Guide 1.97 instrument channels to monitor the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval.

B.1

89 D

Condition B applies when the Required Action and associated Completion Time for Condition A are not met. This Required Action specifies initiation of actions in Specification 5.~~6~~.2.c, "Special Reports," that require a written report, approved by the [onsite review committee], to be submitted to the NRC immediately. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative actions. This action is appropriate in lieu of a shutdown requirement since alternative actions are identified before loss of functional capability and given the likelihood of unit conditions that would require information provided by this instrumentation.

C.1

Condition C applies when one or more Functions have two inoperable required channels (i.e., two channels inoperable in the same Function). Required Action C.1 requires restoring one channel in the Function(s) to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation and the availability of alternate means to obtain the required information. Continuous operation with two required channels inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur. Condition C is modified by a Note that excludes hydrogen monitor channels.

(continued)

BASES

ACTIONS (continued)

D.1

Condition D applies when two hydrogen monitor channels are inoperable. Required Action D.1 requires restoring one hydrogen monitor channel to OPERABLE status within 72 hours. The 72 hour Completion Time is reasonable based on the backup capability of the Post Accident Sampling System to monitor the hydrogen concentration for evaluation of core damage and to provide information for operator decisions. Also, it is unlikely that a LOCA (that would cause core damage) would occur during this time.

E.1

Condition E applies when the Required Action and associated Completion Time of Condition C or D are not met. Required Action E.1 requires entering the appropriate Condition referenced in Table 3.3.3-1 for the channel(s) immediately. The applicable Condition referenced in the Table is Function dependent. Each time an inoperable channel has not met any Required Action of Condition C or D, and the associated Completion Time has expired, Condition E is entered for that channel and provides for transfer to the appropriate subsequent Condition.

F.1 and F.2

If the Required Action and associated Completion Time of Conditions C or D are not met and Table 3.3.3-1 directs entry into Condition F, the unit must be brought to a MODE where the requirements of this LCO do not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and MODE 4 within 12 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

G.1

At ~~this unit~~ Watts Bar, alternate means of monitoring Reactor Vessel Water Level and Containment Area Radiation have been

(continued)

BASES

ACTIONS

9 ~~G.1~~ (continued)

~~developed and tested. These alternate means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allotted time. If these alternate means are used, the Required Action is not to shut down the unit but rather to follow the directions of Specification 5.6.2.c, in the Administrative Controls section of the TS. The report provided to the NRC should discuss the alternate means used, describe the degree to which the alternate means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels.~~

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that SR 3.3.3.1 and SR 3.3.3.3 apply to each PAM instrumentation Function in Table 3.3.3-1:

SR 3.3.3.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross instrumentation failure has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels where feasible. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or even something more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared to similar unit instruments located throughout the unit.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including isolation, indication and readability. If a channel is outside the match criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1 (continued)

are within the match criteria, it is an indication that the channels are OPERABLE.

(E) As specified in the SR, a CHANNEL CHECK is only required for those channels which are normally energized.

The Frequency of 31 days is based on operating experience that demonstrates that channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO Required Channels.

SR 3.3.3.2

A ~~A~~CCOT is performed on each hydrogen monitor ~~every~~ 92 days to ensure the entire channel will ~~perform~~ the intended function. The 92 ~~day~~ Frequency is based on the known reliability of the hydrogen monitors, and has been ~~shown~~ to be acceptable through operating experience. 184

184

(10)

(E) This SR is modified by a Note which excludes neutron detectors. The calibration method for neutron detectors is specified in the Bases of LCO 3.3.1, "Reactor Trip System Instrumentation."

SR 3.3.3.3

A CHANNEL CALIBRATION is performed every [18] months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to measured parameter with the necessary range and accuracy. The Frequency is based on operating experience and consistency with the typical industry refueling cycle.

REFERENCES

1. Regulatory Guide 1.97, ~~[date]~~, Revision 2, December 1980.
2. [Unit specific document (e.g., FSAR, NRC Regulatory Guide 1.97 SER letter).]
3. NUREG-0737, Supplement 1.

(5)

NUREG-0847, Safety Evaluation Report, Supplement No. 9, June 16 1992, section 7.5.2, "Postaccident Monitoring System."

# JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.3.3

## Generic Justifications

A. Modified to reflect WBN plant specific design.

B. Modified to reflect WBN plant specific terminology.

C. Editorial change to improve clarity.

D. Deletion or modification of information which is generic in nature and has been modified to be plant specific to WBN.

E. This change reflects comments made by the industry to the NRC at the RSTS P&R review meeting held in Irvine, CA July 13-20, 1992.

F. Bases revised to reflect on changes made to the LCO.

## Specific Justifications

1. Modified to reflect WBN use if Intermediate Range rather than Power Range to monitor neutron flux.

2. Additional post-accident instrument channels have been added to table 3.3.3-1 to include all those Category I channels identified in Enclosure 2 of TVA October 29, 1991 letter from J.H. Garrity to NRC responding to NRC request for additional information.

3. As indicated in TVA October 29, 1991 letter from J.H. Garrity to NRC responding to NRC request for additional information, the condensate storage tank is not the "primary" source of Auxiliary Feedwater.

4. Not used.

5. Change to specify WBN specific reference information and format.

6. Since RCS subcooling margin is not an interlock the wording has been changed to move correctly discuss it as a variable that is used to make decisions.

7. Because Watts Bar has Functions with more than two channels. Condition C needs to read "two or more."

   The RSTS was written for functions with only one or two required channels. Therefore, Condition C only needed to read "two required channels."

8. A Note has been added to the Applicability to point out that the Source Range Neutron Flux channel is not OPERABLE about the P-6/Intermediate Range Neutron Flux) interlock. The channel is disabled by removing voltage from the monitors.

9. The PAMs ACTIONS table has been modified to correct a breakdown in the RSTS for single channel systems. The RSTS as written would allow indefinite operation in Condition B. The Required Action would never direct a shutdown under Condition E. To correct this problem using the existing structure introduced new problems with the way this particular LCO's ACTIONS table is structured. Therefore, the ACTIONS table and Table 3.3.3-1 were revised to work in a manner consistent with LCO 3.3.1 and 3.3.2. This resolves the structural problem with the RSTS and eliminates the confusion of having these three LCO tables work differently.

10. Although the industry has proposed deleting the Channel Operational Test for the hydrogen monitors, WBN has elected to retain this SR and extend the surveillance interval to 184 days.

## 3.3 INSTRUMENTATION

### 3.3.4 Remote Shutdown System

LCO 3.3.4 The Remote Shutdown System Functions in Table 3.3.4-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

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

---

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more required Functions inoperable. | A.1 Restore required Function to OPERABLE status. | 30 days |
| B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3.<br>AND<br>B.2 Be in MODE 4. | 6 hours<br><br>12 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.3.4.1 | Perform CHANNEL CHECK for each required instrumentation channel that is normally energized. | 31 days |
| SR 3.3.4.2 | Verify each required control circuit and transfer switch is capable of performing the intended function. | [18] months |
| SR 3.3.4.3 | ----------NOTE----------<br>Neutron detectors are excluded from CHANNEL CALIBRATION.<br>----------<br><br>Perform CHANNEL CALIBRATION for each required instrumentation channel. | [18] months |
| SR 3.3.4.4 | Perform TADOT of the reactor trip breaker open/closed indication. | 18 months |

Table 3.3.4-1 (page 1 of 1)
Remote Shutdown System Instrumentation and Controls

NOTE

This table is for illustration ~~purposes only~~. It does not ~~attempt to encompass~~ every Function used at every unit, but does contain the ~~types of Functions~~ commonly found.

| FUNCTION/INSTRUMENT OR CONTROL PARAMETER | LOCATION | REQUIRED NUMBER OF FUNCTIONS |
|---|---|---|
| 1. Reactivity Control | | |
| a. Source Range Neutron Flux | | [1] |
| b. Reactor Trip Breaker Position | | [1 per trip breaker] |
| c. Manual Reactor Trip | | ~~[1]~~ 2 1 |
| 2. Reactor Coolant System (RCS) Pressure Control | | |
| a. Pressurizer Pressure or RCS Wide Range Pressure | | [1] |
| b. Pressurizer Power Operated Relief Valve (PORV) Control and Block Valve Control | | [1, controls must be for PORV & block valves on same line] |
| 3. Decay Heat Removal via Steam Generators (SGs) | | |
| a. RCS Hot Leg Temperature | | [1 per loop] |
| ~~b. RCS Cold Leg Temperature~~ | | ~~[1 per loop]~~ 2 |
| c. AFW Controls ~~Condensate Storage Tank Level~~ | | [1] |
| d. SG Pressure | | [1 per SG] |
| e. SG Level or AFW Flow | | [1 per SG] |
| 4. RCS Inventory Control | | |
| a. Pressurizer Level | | [1] |
| b. Charging Pump Controls | | [1] |

A

# B 3.3 INSTRUMENTATION

## B 3.3.4 Remote Shutdown System

BASES

BACKGROUND

The Remote Shutdown System provides the control room operator with sufficient instrumentation and controls to place and maintain the unit in a safe shutdown condition from a location other than the control room. This capability is necessary to protect against the possibility that the control room becomes inaccessible. A safe shutdown condition is defined as MODE 3. With the unit in MODE 3, the Auxiliary Feedwater (AFW) System and the steam generator (SG) safety valves or the SG atmospheric dump valves (ADVs) can be used to remove core decay heat and meet all safety requirements. The long term supply of water for the AFW System and the ability to borate the Reactor Coolant System (RCS) from outside the control room allows extended operation in MODE 3.

In the event that the control room becomes inaccessible, the operators can establish control ~~at the remote shutdown panel~~ in the Auxiliary Control Room, and place and maintain the unit in MODE 3. Not all controls and necessary transfer switches are located ~~at the remote shutdown panel~~. Some controls and transfer switches will have to be operated locally at the switchgear, motor control panels, or other local stations. The unit automatically reaches MODE 3 following a unit shutdown and can be maintained safely in MODE 3 for an extended period of time.

The OPERABILITY of the remote shutdown control and instrumentation functions ensures there is sufficient information available on selected unit parameters to place and maintain the unit in MODE 3 should the control room become inaccessible.

APPLICABLE SAFETY ANALYSES

The Remote Shutdown System is required to provide equipment at appropriate locations outside the control room with a capability to promptly shut down and maintain the unit in a safe condition in MODE 3.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The criteria governing the design and specific system requirements of the Remote Shutdown System are located in 10 CFR 50, Appendix A, GDC 19 (Ref. 1).

The Remote Shutdown System is considered an important contributor to the reduction of unit risk to accidents and as such it has been retained in the Technical Specifications as indicated in the NRC Policy Statement.

LCO

The Remote Shutdown System LCO provides the OPERABILITY requirements of the instrumentation and controls necessary to place and maintain the unit in MODE 3 from a location other than the control room. The instrumentation and controls ~~typically~~ required are listed in Table 3.3.4-1. ~~in the accompanying LCO.~~

Watts Bar

~~Reviewer Note~~ For channels that fulfill GDC 19 requirements, the number of OPERABLE channels required depends upon the unit licensing basis as described in the NRC ~~unit specific~~ Safety Evaluation Report (SER). Generally, two divisions are required OPERABLE. However, only one channel per a given Function is required if the unit has justified such a design, and NRC's SER accepted the justification.

The controls, instrumentation, and transfer switches are required for:

- Core reactivity control (initial and long term);
- RCS pressure control;
- Decay heat removal via the AFW System and the SG safety valves or SG ADVs;
- RCS inventory control via charging flow; and
- Safety support systems for the above Functions, including service water, component cooling water, and onsite power, including the diesel generators.

A Function of a Remote Shutdown System is OPERABLE if all instrument and control channels needed to support the Remote Shutdown System Function are OPERABLE. In some cases,

(continued)

BASES

LCO (continued)

△

Table 3.3.4-1 may indicate that the required information or control capability is available from several alternate sources. In these cases, the Function is OPERABLE as long as one channel of any of the alternate information or control sources is OPERABLE.

The remote shutdown instrument and control circuits covered by this LCO do not need to be energized to be considered OPERABLE. This LCO is intended to ensure the instruments and control circuits will be OPERABLE if unit conditions require that the Remote Shutdown System be placed in operation.

APPLICABILITY

The Remote Shutdown System LCO is applicable in MODES 1, 2, and 3. This is required so that the unit can be placed and maintained in MODE 3 for an extended period of time from a location other than the control room.

This LCO is not applicable in MODE 4, 5, or 6. In these MODES, the facility is already subcritical and in a condition of reduced RCS energy. Under these conditions, considerable time is available to restore necessary instrument control functions if control room instruments or controls become unavailable.

ACTIONS

△

Note 1 is included which excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into an applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require a unit shutdown. This exception is acceptable due to the low probability of an event requiring the Remote Shutdown System and because the equipment can generally be repaired during operation without significant risk of spurious trip.

△

Note 2 has been added to the ACTIONS to clarify the application of Completion Time rules. Separate Condition entry is allowed for each Function listed on Table 3.3.4-1. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

(continued)

BASES

ACTIONS (continued)

A.1

Condition A addresses the situation where one or more required Functions of the Remote Shutdown System are inoperable. This includes any Function listed in Table 3.3.4-1, as well as the control and transfer switches.

The Required Action is to restore the required Function to OPERABLE status within 30 days. The Completion Time is based on operating experience and the low probability of an event that would require evacuation of the control room.

B.1 and B.2

△ If the Required Action and associated Completion Time of Condition A is not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit 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 unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of even something more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

△ Agreement criteria are determined by the unit staff based on a combination of the channel instrument uncertainties, including indication and readability. If the channels are within the match criteria, it is an indication that the channels are OPERABLE. If the channels are normally off

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1 (continued)

scale during times when Surveillance is required, the CHANNEL CHECK will verify only that they are off scale in the same direction. Offscale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.

As specified in the Surveillance, a CHANNEL CHECK is only required for those channels which are normally energized.

The Frequency of 31 days is based upon operating experience which demonstrates that channel failure is rare.

SR 3.3.4.2

SR 3.3.4.2 verifies each required Remote Shutdown System control circuit and transfer switch performs the intended function. This verification is performed from the ~~remote shutdown panel~~ Auxiliary Control Room and locally, as appropriate. This will ensure that if the control room becomes inaccessible, the unit can be placed and maintained in MODE 3 from the ~~remote shutdown panel~~ Auxiliary Control Room and the local control stations. 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. (However, this Surveillance is not required to be performed only during a unit outage.) Operating experience demonstrates that remote shutdown control channels usually pass the Surveillance test when performed at the [18] month Frequency.

SR 3.3.4.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The test verifies that the channel responds to measured parameters with the necessary range and accuracy.

The Frequency of [18] months is based upon operating experience and consistency with the typical industry refueling cycle.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.4.4

SR 3.3.4.4 is the performance of a TADOT every 18 months. This test should verify the OPERABILITY of the reactor trip breakers (RTBs) open and closed indication on the remote shutdown panel, by actuating the RTBs. The Frequency of 18 months was chosen because the RTBs cannot be exercised while the facility is at power. The Frequency is based upon operating experience and consistency with the typical industry refueling outage.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 19.

## JUSTIFICATIONS FOR DEVIATIONS FROM STS LCO 3.3.4

A. This change reflects comments made by the industry to the NRC at the RSTS P&R review meeting held in Irvine, CA July 13-20,1992.

1. The "Manual Reactor Trip" has been deleted. The reactor is tripped prior to evacuation from the main control room.

2. TVA has recommended to use T-sat instead of T-cold in assessing natural circulation cooling in the RCS. This recommendation has been evaluated and accepted by the NRC staff (reference TVA 9105230.136).

Δ. A triangle is used to indicate a place with editorial differences.

## 3.3 INSTRUMENTATION

### 3.3.5 Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation

LCO 3.3.5 ~~[Four]~~ Three channels per bus of the loss of voltage Function and ~~[four]~~ Three channels per bus of the degraded voltage Function and shall be OPERABLE.

two channels per bus of the diesel generator starting Function and two channels per bus of the load shedding Function.

APPLICABILITY: MODES 1, 2, 3, and 4,
When associated DG is required to be OPERABLE by LCO 3.8.2, "AC Sources – Shutdown."

ACTIONS

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

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more Function with one channel per bus inoperable. | A.1 --------NOTE--------- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ---------- Place channel in trip. | For loss of voltage or degraded voltage Functions<br>6 hours |
| B. One or more Function with two or more channels per bus inoperable. | B.1 Restore all but one channel to OPERABLE status. | 1 hour |

Stet

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| ~~f~~. Required Action and associated Completion Time not met. | ~~f~~.1 Enter applicable Condition(s) and Required Action(s) for the associated DG made inoperable by LOP DG start instrumentation. | Immediately |

Stet

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| (A) ~~SR 3.3.5.1 Perform CHANNEL CHECK.~~ | ~~12 hours~~ |
| SR 3.3.5.~~2~~ 1 Perform TADOT. | ~~[31~~ days~~]~~ |
| SR 3.3.5.~~3~~ 2 (D) Verify system actuation Response Time is within limit. | ~~[18]~~ months on a STAGGERED TEST BASIS |

Stet

(continued)

NOTE
~~Relay~~ Verification of relay setpoints not required. (2)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | FREQUENCY |
|---|---|
| SR 3.3.5.~~4~~ 2 Perform CHANNEL CALIBRATION with ~~either~~ [setpoint Allowable Value ~~or Trip Setpoint and Allowable Value~~] as follows: | [18] months |
| a. ~~Loss of voltage Allowable Value ≥ [2912] volts with a time delay of [0.8] ± [ ] second.~~ ~~Loss of voltage Trip Setpoint ≥ [2975] volts with a time delay of [0.8] ± [ ] second.~~ | |
| b. ~~Degraded voltage Allowable Value ≥ [3683] volts with a time delay of [20] ± [ ] seconds.~~ ~~Degraded voltage Trip Setpoint ≥ [3746] volts with a time delay of [20] ± [ ] seconds.~~ | |

Replace with

a. Loss of voltage Allowable Value [6003±69] volts with a time ~~relay~~ ~~with~~ 0.25 to 0.5 second ~~time~~ delay of for trip of the 6.9 kV shutdown board supply breaker.

b. Degraded voltage Allowable Value [~~6560~~ ±33] volts with a time delay of 6±0.5 seconds for trip of the 6.9 kV shutdown board supply breaker.

c. Diesel Generator start Allowable Value 4830±96.6 volts with an internal time delay of 0.5 seconds nominal at zero volts.

d. Load shedding Allowable Value 4830±96.6 volts with an internal time delay of 3 seconds nominal at zero volts.

## B 3.3 INSTRUMENTATION

### B 3.3.5 Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation

BASES

BACKGROUND

Three degraded voltage

The DGs provide a source of emergency power when offsite power is either unavailable or is insufficiently stable to allow safe unit operation. Undervoltage protection will generate a LOP start in the event a loss of voltage or degraded voltage condition occurs in the switchyard. There are ~~two~~ four LOP start signals, one for each ~~4.16~~ 6.9 kV ~~vital~~ Shutdown (B) bus.

(one per phase)

~~Four undervoltage~~ relays ~~with inverse time characteristics~~ are provided on each ~~4160~~ 6900V Class 1E ~~instrument~~ bus for the purpose of detecting a sustained undervoltage condition ~~or a loss of bus voltage~~. The relays are combined in a two out of ~~four~~ three logic to generate a ~~LOP~~ supply breaker trip signal if the voltage is below ~~75%~~ 98% for ~~a short time or below 90% for a long time.~~ ~~The LOP start actuation is described in FSAR, Section 8.3~~, ~~"Electrical"~~ ~~(Ref. 1)~~. 6 seconds

Insert B (1)

Trip Setpoints and Allowable Values

The Trip Setpoints used in the relays are based on the analytical limits presented in FSAR, ~~Chapter~~ (B) Section 15, "Accident Analysis" (Ref. 2). The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account.

The actual nominal Trip Setpoint entered into the relays is normally still more conservative than that required by the Allowable Value. If the measured setpoint does not exceed the Allowable Value, the relay is considered OPERABLE.

Setpoints adjusted in accordance with the Allowable Value ensure that the consequences of accidents will be acceptable, providing the unit is operated from within the LCOs at the onset of the accident and that the equipment functions as designed.

Allowable Values and/or Trip Setpoints are specified for each Function in the LCO. Nominal Trip Setpoints are also specified in the unit specific setpoint calculations. The nominal setpoints are selected to ensure that the setpoint measured by the surveillance procedure does not exceed the

(continued)

# Insert B

(1)

Additionally, three undervoltage relays (one per phase) are provided on each 6900v Class 1E bus for the purpose of detecting a loss of voltage condition. The relays are combined in a two out of three logic to generate a supply breaker trip signal if the voltage is below 87% for 0.25 to 0.5 seconds.

Once the supply breakers have been opened, an induction disk type relay, which has a voltage setpoint of 70% of 6.9kV (nominal, decreasing) and an internal time delay of 0.5 seconds at zero volts, will start the diesel generator. Another induction disk type relay, which has a voltage setpoint of 70% of 6.9kV (nominal, decreasing) and an internal time delay of 3 seconds, at zero volts, will initiate load shedding of the 6.9kV shutdown board and close the 480V shutdown board current limiting reactor bypass breaker. The LOP start actuation is described in FSAR Section 8.0, "Electric Power" (Ref. 1) (B)

BASES

BACKGROUND

Trip Setpoints and Allowable Values (continued)

Allowable Value if the relay is performing as required. If the measured setpoint does not exceed the Allowable Value, the relay is considered OPERABLE. Operation with a Trip Setpoint less conservative than the nominal Trip Setpoint, but within the Allowable Value, is acceptable provided that operation and testing is consistent with the assumptions of the unit specific setpoint calculation. Each Allowable Value and/or Trip Setpoint specified is more conservative than the analytical limit assumed in the transient and accident analysis in order to account for instrument uncertainties appropriate to the trip function. These uncertainties are defined in the "~~Unit Specific RTS/ESFAS~~ Setpoint Methodology ~~Study~~" (Ref. 3).

B Westinghouse for Protection System Watts Bar 1 and 2

APPLICABLE SAFETY ANALYSIS

The LOP DG Start Instrumentation is required for the Engineered Safety Features (ESF) Systems to function in any accident with a loss of offsite power. Its design basis is that of the ESF Actuation System (ESFAS).

Accident analyses credit the loading of the DG based on the loss of offsite power during a loss of coolant accident (LOCA). The actual DG start has historically been associated with the ESFAS actuation. The DG loading has been included in the delay time associated with each safety system component requiring DG supplied power following a loss of offsite power. The analysis assumes a non mechanistic DG loading, which does not explicitly account for each individual component of loss of power detection and subsequent actions.

A time delay includes contributions from the DG start, DG loading, and Safety Injection (SI) System component actuation. The response of the DG to a LOP must be demonstrated to fall within this analysis response time when including the contributions of all portions of the delay.

The required channels of LOP DG Start Instrumentation, in conjunction with the ESF Systems powered from the DGs, provide unit protection in the event of any of the analyzed accidents discussed in Reference 2, in which a loss of offsite power is assumed.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The delay times assumed in the safety analysis for the ESF equipment include the 10 second DG start delay, and the appropriate sequencing delay, ~~if applicable~~. The response times for ESFAS actuated equipment in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," include the appropriate DG loading and sequencing delay.

The LOP DG Start Instrumentation channels satisfy Criterion 3 of the NRC Policy Statement.

LCO

The LCO for the LOP DG Start Instrumentation requires that Three ~~[four]~~ channels per bus of both the loss of voltage and the degraded voltage Functions and two channels per bus of the diesel generator starting and the load shedding Functions shall be OPERABLE in MODES 1, 2, 3, and 4 when the LOP DG Start Instrumentation supports safety systems associated with the ESFAS. In MODES 5 and 6, the ~~[four]~~ 8 channels must be OPERABLE whenever the associated DG is required to be OPERABLE to ensure that the automatic start of the DG is available when needed. Loss of LOP DG Start Instrumentation Function could result in the delay of safety systems initiation when required. This could lead to unacceptable consequences during accidents. During the loss of offsite power the DG powers the motor driven auxiliary feedwater pumps. Failure of these pumps to start would leave only one turbine driven pump, as well as an increased potential for a loss of decay heat removal through the secondary system.

APPLICABILITY

The LOP DG Start actuation Function is required in MODES 1, 2, 3, and 4 because ESF Functions are designed to provide protection in these MODES. Actuation in MODE 5 or 6 is required whenever the required DG must be OPERABLE so that it can perform its Function on an LOP or degraded power to the vital bus.

ACTIONS

In the event a channel's Trip Setpoint is found nonconservative with respect to the Allowable Value, or the channel is found inoperable, then the Function that channel provides must be declared inoperable and the LCO Condition entered for the particular protection Function affected.

(continued)

BASES

ACTIONS (continued)

Because the required channels are specified on a per bus basis, the Condition may be entered separately for each bus as appropriate.

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in the LCO. The Completion Time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies to the LOP DG Start Function with one ~~loss of voltage or degraded voltage channel~~ per bus inoperable.

or one out of one logic for initiation of subsequent DG start and bus load shedding

If one channel is inoperable, Required Action A.1 requires that channel be placed in trip in 6 hours. With a channel in trip, the LOP DG Start Instrumentation channels are configured to provide a one out of three logic to initiate a trip of the incoming offsite power.

A Note is added to allow bypassing an inoperable channel for up to 4 hours for surveillance testing of other channels. This allowance is made where bypassing the channel does not cause an actuation and where at least two other channels are monitoring that parameter.

The specified Completion Time and time allowed for bypassing one channel are reasonable considering the Function remains fully OPERABLE on every bus and the low probability of an event occurring during these intervals.

B.1

Condition B applies when more than one ~~undervoltage or more than one degraded voltage~~ channel on a single bus is inoperable.

Required Action B.1 requires restoring all but one channel to OPERABLE status. The 1 hour Completion Time should allow ample time to repair most failures and takes into account

(continued)

BASES

ACTIONS

B.1 (continued)

the low probability of an event requiring a LOP start occurring during this interval.

~~C~~.1 ← Stet

Condition ~~C~~ applies to each of the LOP DG Start Functions when the Required Action and associated Completion Time ~~for~~ ~~cf~~ Condition A ~~or~~ B are not met.
Stet

In these circumstances the Conditions specified in LCO 3.8.1, "A~~c~~ Sources – Operating," or 3.8.2, "A~~c~~ Sources – Shutdown," for the DG made inoperable by failure of the LOP DG start instrumentation are required to be entered immediately. The actions of those LCOs provide for adequate compensatory actions to assure unit safety.

SURVEILLANCE REQUIREMENTS

~~SR 3.3.5.1~~

~~Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or even something more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.~~

A

~~Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the match criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.~~

~~The Frequency is based upon operating experience that demonstrates channel failure is rare. Thus, performance of the CHANNEL CHECK ensures that undetected overt channel~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS

(A)

~~SR 3.3.5.1 (continued)~~

~~failure is limited to 12 hours. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.~~

SR 3.3.5.~~2~~ 1

SR 3.3.5.~~2~~ 1 is the {performance of a TADOT. This test is performed every ~~[31~~ days}. The test checks trip devices that provide actuation signals directly, bypassing the analog process control equipment. ~~relay trip setpoints are verified and adjusted as necessary.~~ For these tests, the Frequency is based upon the known reliability of the relays and controls, the multichannel redundancy available, and has been shown to be acceptable through operating experience.

(2)

The SR is modified by a Note that excludes verification of ~~relay~~ setpoints for relays. Relay setpoints are verified during a CHANNEL CALIBRATION.

(B)

Engineered Safety Features

~~SR 3.3.5.3~~ 2

~~This SR ensures the individual channel LOP DG Start Instrumentation actuation response times are less than or equal to the maximum values assumed in the accident analysis. Response time testing acceptance criteria are included in Technical Requirements Manual, Section ~~15,~~ 3.3.2 "Response ~~Time~~ Times" (Ref. 4).~~

(D)

~~ESF RESPONSE TIME tests are conducted on an [18] month STAGGERED TEST BASIS. The [18] month Frequency is consistent with the typical refueling cycle and is based upon unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.~~

SR 3.3.5.~~4~~ 32

SR 3.3.5.~~4~~ 32 is the performance of a CHANNEL CALIBRATION. A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.~~4~~ 3 (continued)

The setpoints, as well as the response to a loss of voltage and a degraded voltage test, shall include a single point verification that the trip occurs within the required time delay, as shown in Reference 1.

A CHANNEL CALIBRATION is performed every [18] months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop including the sensor. The test verifies that the channel responds to measured parameter within the necessary range and accuracy.

The Frequency of [18] months is based upon operating experience and consistency with the typical industry refueling cycle and is justified by the assumption of an [18] month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. FSAR, Section [8.3].
2. FSAR, Chapter [15].
3. Unit Specific RTS/ESFAS Setpoint Methodology Study.
4. Technical Requirements Manual, Section 15, "Response Times."

Replace with

Watts Bar

1. FSAR, Section 8.~~3~~2, "Electrical." Power
2. FSAR, Section 15, "Accident Analysis."
3. "Westinghouse Setpoint Methodology for Protection System Watts Bar 1 and 2."
4. Technical Requirements Manual, Section 3.3.2, Engineered Safety Feature Response Times.

B

D

## JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.3.5

### Generic Justifications

A. Modified to reflect WBN plant specific design.

B. Modified to reflect WBN plant specific terminology.

C. Deletion or modification of information which is generic in nature and has been modified to be plant specific to WBN.

D. This change reflects comments made by the industry to the NRC at the RSTS P&R review meeting held in Irvine, CA July 13-20,1992.

### Specific Justifications

1. Modified to reflect WBN plant specific design. WBN has three degraded voltage and three undervoltage relays which are combined in a two out of three logic to generate a trip of the supply breaker. Once the supply breaker is opened on a bus, two sets of relays with independent functions actuate to strip the bus and start the diesel. The bus stripping relays consist of four relays in a one out of two taken twice logic, and the diesel start relays consist of two relays in a one out of two logic.

2. These SRs are TADOTs that apply to undervoltage and degraded voltage relays to the DGs. The setpoint is not verified when performing a TADOT. These devices are simply triggered and actuation verified. Setpoint verification requires elaborate bench calibration and is done every 18 months.

Attachment C

August 1992 Draft WBN Technical Specifications

① GLOBAL ACOT TO COT



## 3.3 INSTRUMENTATION

### 3.3.6 Containment ~~Purge and Exhaust~~ VENT Isolation Instrumentation (B)

LCO 3.3.6 The Containment ~~Purge and Exhaust~~ VENT Isolation instrumentation for each Function in Table 3.3.6-1 shall be OPERABLE. (B)

APPLICABILITY: MODES 1, 2, 3, and 4,
During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

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

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One radiation monitoring channel inoperable. | A.1 Restore the affected channel to OPERABLE status. | 4 hours |
| (Only) (E) ----------NOTE---------- Applicable in MODE 1, 2, 3, or 4. ---------- B. One or more Functions with one or more manual or automatic actuation trains inoperable. OR | B.1 Enter applicable Conditions and Required Actions ~~for affected valves~~ of (E) LCO 3.6.3, "Containment Isolation Valves," made inoperable by (E) isolation instrumentation. for Containment Purge and Exhaust isolation valves | Immediately |

(continued)

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| B. (continued)<br><br>(A) Two ~~or more~~ radiation monitoring channels inoperable.<br><br>OR<br><br>Required Action and associated Completion Time of Condition A not met. | | |
| (Only) (E)<br>----------NOTE----------<br>Applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment.<br>----------<br><br>C. One or more Functions with one or more manual or automatic actuation trains inoperable.<br><br>OR<br><br>(A) Two ~~or more~~ radiation monitoring channels inoperable.<br><br>OR<br><br>Required Action and associated Completion Time for Condition A not met. | C.1 Place and maintain containment purge and exhaust valves in closed position.<br><br>OR<br><br>C.2 Enter applicable Conditions and Required Actions ~~for affected valves~~ (E) of LCO 3.9.4, "Containment Penetrations," made inoperable by isolation instrumentation. (E) for Containment Purge and Exhaust isolation valves | Immediately<br><br><br><br>Immediately |

SURVEILLANCE REQUIREMENTS

----------NOTE----------
Refer to Table 3.3.6-1 to determine which SRs apply for each Containment ~~Purge and Exhaust~~ VENT (B) Isolation Function.
---

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.3.6.1 | Perform CHANNEL CHECK. | 12 hours |
| SR 3.3.6.2 | Perform ACTUATION LOGIC TEST. | 31 days on a STAGGERED TEST BASIS |
| SR 3.3.6.3 | Perform MASTER RELAY TEST. | 31 days on a STAGGERED TEST BASIS |
| SR 3.3.6.4 | Perform ACOT. (1) | 92 days |
| SR 3.3.6.5 | Perform SLAVE RELAY TEST. | [92] days |
| STET SR 3.3.6.6 (A) | Perform ESF RESPONSE TIME test. | [18] months on a STAGGERED TEST BASIS |
| SR 3.3.6.~~7~~ 6 | Perform TADOT. | [18] months |
| SR 3.3.6.~~8~~ 7 | Perform CHANNEL CALIBRATION. | [18] months |

Table 3.3.6-1 (page 1 of 1)
Containment Purge and Exhaust Isolation Instrumentation

| FUNCTION | REQUIRED CHANNELS | SURVEILLANCE REQUIREMENTS | TRIP SETPOINT |
|---|---|---|---|
| 1. Manual Initiation | 2 | SR 3.3.6.~~7~~ 6 | N/A |
| 2. Automatic Actuation Logic and Actuation Relays | 2 trains | SR 3.3.6.2<br>SR 3.3.6.3<br>SR 3.3.6.5<br>~~SR 3.3.6.6~~ | N/A |
| 3. Containment PURGE EXHAUST Radiation MONITORS | 2 | | |
| a. Gaseous | [1] | SR 3.3.6.1<br>SR 3.3.6.4<br>SR 3.3.6.~~8~~ 7 | ≤ [2 x background] |
| b. Particulate | [1] | SR 3.3.6.1<br>SR 3.3.6.4<br>SR 3.3.6.8 | ≤ [2 x background] |
| c. Iodine | [1] | SR 3.3.6.1<br>SR 3.3.6.4<br>SR 3.3.6.8 | ≤ [2 x background] |
| d. Area Radiation | [1] | SR 3.3.6.1<br>SR 3.3.6.4<br>SR 3.3.6.8 | ≤ [2 x background] |
| 4. ~~Containment Isolation—Phase A~~ Safety Injection (SI) | Refer to LCO 3.3.2, "ESFAS Instrumentation," Function ~~3.a.~~ 1, for all initiating Functions and requirements. | | |

Step

B 3.3 INSTRUMENTATION

B 3.3.6 Containment ~~Purge and Exhaust~~ VENT Isolation Instrumentation (B)

BASES

BACKGROUND

Containment ~~Purge~~ VENT Isolation closes the containment isolation valves in the ~~Mini Purge System and the Shutdown~~ CONTAINMENT Purge System. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. The ~~Mini~~ CONTAINMENT Purge System may be in use during reactor operation and ~~the Shutdown Purge System will be in use~~ with the reactor shutdown. (B) (B) (A)

Containment ~~Purge~~ VENT Isolation ~~also~~ IS initiated on an automatic Safety Injection (SI) Function ~~through the Containment~~ or ~~Phase A Isolation Function or~~ by manual actuation of Phase A Isolation. The Bases for LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," discuss these other modes of initiation. (A) (B)

(A) REDUNDANT AND INDEPENDENT GASEOUS RADIOACTIVITY MONITORS MEASURE THE RADIOACTIVITY LEVELS OF THE CONTAINMENT PURGE EXHAUST, EACH OF WHICH WILL INITIATE ITS ASSOCIATED TRAIN OF AUTOMATIC CONTAINMENT VENT ISOLATION UPON DETECTION OF HIGH GASEOUS RADIOACTIVITY.

~~Four radiation monitoring channels are also provided as input to the Containment Purge Isolation Function. The four channels measure containment radiation at two locations. One channel is a containment area gamma monitor, and the other three measure radiation in a sample of the containment purge exhaust. The three purge exhaust radiation detectors are of three different types: gaseous, particulate, and iodine monitors. All four detectors will respond to most events that release radiation to containment. However, analyses have not been conducted to demonstrate that all credible events will be detected by more than one monitor. Therefore, for the purposes of this LCO the four channels are not considered redundant. Instead, they are treated as four one out of one Functions. Since the purge exhaust monitors constitute a sampling system, various components such as sample line valves, sample line heaters, sample pumps, and filter motors are required to support monitor OPERABILITY.~~

~~Each of~~ The purge systems has inner and outer containment isolation valves in its supply and exhaust ducts. ~~A high radiation signal from any one of the four channels initiates containment purge isolation, which closes both inner and outer containment isolation valves in the Mini Purge System and the Shutdown Purge System.~~ (A) The ~~Mini~~ CONTAINMENT Purge System may be

(continued)

BASES

BACKGROUND (continued)

in use during reactor operation and ~~the Shutdown Purge System will be in use~~ with the reactor shutdown. ~~These~~ THIS (A) systems ~~are~~ is described in the Bases for LCO 3.6.3, "Containment Isolation Valves." The Containment Purge (B) EXHAUST ~~Isolation~~ Radiation Monitoring Instrumentation isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident.

APPLICABLE SAFETY ANALYSES

The safety analyses assume that the containment remains intact with penetrations unnecessary for core cooling isolated early in the event, approximately 60 seconds. ~~The isolation of the purge valves has not been analyzed mechanistically in the dose calculations, although its rapid isolation is assumed.~~ The containment ~~purge~~ isolation VENT (B) radiation monitors act as backup to the SI signal to ensure closing of the purge and exhaust valves. They are also the primary means for automatically isolating containment in the event of a fuel handling accident during shutdown. Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accidental offsite radiological doses are below 10 CFR 100 (Ref. 1) limits.

(A) THE CONTAINMENT ISOLATION VALVES FOR THE CONTAINMENT PURGE SYSTEM CLOSE WITHIN FOUR SECONDS FOLLOWING ACTUATION OF AN ISOLATION SIGNAL.

air system supply (B)

The Containment ~~Purge and Exhaust~~ VENT Isolation Instrumentation satisfies Criterion 3 of the NRC Policy Statement. (B)

LCO

The LCO requirements ensure that the instrumentation necessary to initiate Containment ~~Purge and Exhaust~~ VENT (B) Isolation, listed in Table 3.3.6-1, is OPERABLE.

1. Manual Initiation

The LCO requires two channels OPERABLE. The operator can initiate Containment Purge Isolation at any time by using either of two switches in the control room. Either switch actuates both trains. This action will cause actuation of all components in the same manner as any of the automatic actuation signals. THESE MANUAL SWITCHES ALSO INITIATE A PHASE A ISOLATION SIGNAL. (A)

(continued)

BASES

LCO

1. Manual Initiation (continued)

The LCO on Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

Each channel consists of one ~~push button~~ SELECTOR SWITCH (B) and the interconnecting wiring to the actuation logic cabinet.

2. Automatic Actuation Logic and Actuation Relays

The LCO requires two trains of Automatic Actuation Logic and Actuation Relays OPERABLE to assure that no single random failure can prevent automatic actuation.

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b, SI, and ESFAS Function 3.a, Containment Phase A Isolation. (A) The Applicable MODES and conditions specified for the (B) vent containment ~~purge~~ isolation portion of ~~these~~ the SI Functions is ~~are~~ different and less restrictive than those for the ~~to Phase A isolation and~~ SI roles. (A) If one or more of the SI ~~or Phase A isolation~~ Functions becomes inoperable in such a manner that only the Containment (B) VENT ~~Purge~~ Isolation Function is affected, the Conditions applicable to the ~~ip~~ SI ~~and Phase A isolation~~ Functions need not be entered. The less restrictive Actions specified for inoperability of the Containment ~~Purge~~ VENT (B) Isolation Functions specify sufficient compensatory measures for this case.

3. Containment Radiation

The LCO specifies ~~four~~ TWO (A) required channels of radiation monitors to ensure that the radiation monitoring instrumentation necessary to initiate Containment (B) VENT ~~Purge~~ Isolation remains OPERABLE.

For sampling systems, channel OPERABILITY involves more than OPERABILITY of the channel electronics. OPERABILITY may also require correct valve lineups,

(continued)

BASES

LCO

3. Containment Radiation (continued)

(A) and sample pump operation, ~~and filter motor operation~~ as well as detector OPERABILITY if these supporting features are necessary for trip to occur under the conditions assumed by the safety analyses.

(A) 4. Safety Injection (SI) ~~Containment Isolation - Phase A~~

Refer to LCO 3.3.2, Function ~~3.a.~~ 1, for all initiating Functions and requirements.

APPLICABILITY

Containment ~~Purge~~ VENT Manual Isolation, Automatic Logic, and Radiation Monitoring Instrumentation input Functions are required OPERABLE in MODES 1, 2, 3, and 4 and during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. Under these conditions the potential exists for an accident that could release fission product radioactivity into containment. Therefore, the containment purge and exhaust isolation instrumentation must be OPERABLE in these MODES. (B)

(B) While in MODES 5 and 6 without fuel handling in progress, the CONTAINMENT VENT ~~Purge Valve~~ Isolation System need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within the limits of Reference 1.

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module ~~sufficient to~~ exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of an ~~A~~COT, when the process instrumentation is set up for adjustment to bring it within specification. (1) If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, (8)

(continued)

BASES

ACTIONS (continued)

the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.6-1 (2) IN THE ACCOMPANYING LCO. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A is applicable to the failure of one containment purge isolation radiation monitor channel. Since the ~~four~~ TWO (A) containment radiation monitors (A) ARE BOTH GASEOUS DETECTORS ~~measure different parameters~~, failure of a single channel may result in loss of the REDUNDANCY. ~~radiation monitoring Function for certain events.~~ Consequently, the failed channel must be restored to OPERABLE status. The 4 hours allowed to restore the affected channel is justified by the low likelihood of events occurring during this interval, and recognition that one or more of the remaining channels will respond to most events.

B.1

Condition B is applicable to the Containment ~~Purge~~ VENT (B) Isolation Function and addresses the train orientation of the Solid State Protection System (SSPS) and the master and slave relays for this Function. It also addresses the failure of multiple radiation monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1. Condition B is applicable in MODES 1, 2, 3, and 4.

If a train is inoperable, multiple channels are inoperable, or the Required Action and associated Completion Time of Condition A is not met, operation may continue as long as the Required Actions for the applicable Conditions of LCO 3.6.3 are met for each valve made inoperable by failure of isolation instrumentation.

(continued)

BASES

ACTIONS

B.1 (continued)

A Note is added stating that Conditions are only applicable in MODES 1, 2, 3, and 4.

C.1

Condition C is applicable to the Containment ~~Purge~~ VENT (B) Isolation Function and addresses the train orientation of the SSPS and the master and slave relays for this Function. It also addresses the failure of multiple radiation monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1. If a train is inoperable, multiple channels are inoperable, or the Required Action and associated Completion Time of Condition A is not met, operation may continue as long as the Required Action to place and maintain Containment Purge and Exhaust Valves in their closed position is met or the applicable Conditions of LCO 3.9.4, "Containment Penetrations," are met for each valve made inoperable by failure of isolation instrumentation. The Completion Time for these Required Actions is Immediately.

A Note states that Condition C is applicable during CORE ALTERATIONS and during movement of irradiated fuel assemblies within containment.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.6-1 determines which Surveillances apply to which Containment Purge and Exhaust Isolation Functions.

SR 3.3.6.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1 (continued)

of the channels or even something more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

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

The Frequency is based upon operating experience that demonstrates channel failure is rare. Thus, performance of the CHANNEL CHECK ensures that undetected overt channel failure is limited to 12 hours. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.6.2

SR 3.3.6.2 is the performance of an ACTUATION LOGIC TEST. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection Function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and there is an intact voltage signal path to the master relay coils. This test is performed every 31 days on a STAGGERED TEST BASIS. The Surveillance interval is justified in ~~WCAP-10271-P-A, Supplement 2, Rev. 1 (Ref. 2).~~ (1)

SR 3.3.6.3

SR 3.3.6.3 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.3 (continued)

operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity. This test is performed every 31 days on a STAGGERED TEST BASIS. The Surveillance interval is justified in Reference 2.

SR 3.3.6.4

(1) (3) An ACOT is performed every 92 days on each required channel to ensure the entire channel will perform the intended function. The Frequency is based on the staff recommendation for increasing the availability of radiation monitors according to NUREG-1366 (Ref. 3). This test verifies the capability of the instrumentation to provide the containment vent (B) ~~purge and exhaust system~~ isolation. The setpoint shall be left consistent with the ~~current~~ unit specific (8) calibration procedure tolerance. The Frequency is based upon the known reliability of the monitoring equipment and has been shown to be acceptable through operating experience.

SR 3.3.6.5

SR 3.3.6.5 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation MODE (4) is either allowed to function or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment that may not be operated in the design mitigation MODE is prevented from operation by the SLAVE RELAY TEST circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay. This test is performed every [92] days. The Frequency is justified in Reference 2.

LOWER CASE

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.6

This SR ensures the individual channel purge and exhaust isolation times are less than or equal to the maximum values assumed in the accident analysis. Response time testing acceptance criteria are included in ~~the Technical Requirements Manual, Section 15, "Response Times"~~ (Ref. 4). Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the ~~T~~trip setpoint value at the sensor, to the point at which the containment purge and exhaust isolation valves close.

ESF RESPONSE TIME tests are conducted on an {18} month STAGGERED TEST BASIS. Testing of the isolation valves, which make up the bulk of the response time, is included. The {18} month Frequency is consistent with the typical refueling cycle and is based upon unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

SR 3.3.6.~~7~~6 7

SR 3.3.6.~~7~~ 7 is the performance of a TADOT. This test is a check of the Manual Actuation Functions and is performed every {18} months. Each Manual Actuation Function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.).

The test also includes trip devices that provide actuation signals directly to the SSPS, bypassing the analog process control equipment. For these tests, the relay trip setpoints are verified and adjusted as necessary. The Frequency is justified in Reference ~~4~~ 2.

SR 3.3.6.~~8~~ 8

A CHANNEL CALIBRATION is performed every {18} months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop including the sensor.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.8 (continued)

The test verifies that the channel responds to measured parameter within the necessary range and accuracy.

The Frequency is based upon operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

INSERT

(see next page)

1. 10 CFR 100.11, 1973.

2. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.

3. NUREG-1366, [date].

4. WATTS BAR Technical Requirements Manual, Section ~~15~~ 3.3.2, ENGINEERED SAFETY FEATURE Response Times.

## REFERENCES INSERT


1. 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," 1973.

2. WCAP-10271-P-A, Supplement 2, Rev. 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," March 1987.

3. NUREG-1366.

4. Watts Bar Technical Requirements Manual, Section 3.3.2, Engineered Safety Feature Response Times.

# JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.3.6

## Generic Justifications

A. Modified to reflect WBN plant specific design.

B. Modified to reflect WBN plant specific terminology.

C. This change reflects comments made by the industry to the NRC at the RSTS P&R review meeting held in Irvine, CA July 13-20,1992.

## Specific Justifications

1. The defined term "ACOT" in Section 1.1, Definitions, has been changed to "COT" and has been changed in this Instrumentation specification to reflect the Definitions change.

2. Editorial change. The phrase "in the accompanying LCO" has been added to clarify the location of the referenced Table.

3. Editorial change. The word "function" has been capitalized, "Function", consistent with the normal practice in Technical Specifications.

4. Editorial change. The word "MODE" should not be capitalized in this usage, since it is <u>not</u> referring to a MODE of Applicability as defined in Section 1.1, Definitions.

5. Editorial change. The term "trip setpoint" has been capitalized, "Trip Setpoint", consistent with the normal practice in Technical Specifications.

6. Editorial change. The reference which provides justification for the Completion Time is Reference 2 (NUREG-1366), not Reference 4 (response times). The reference has been corrected.

7. Change to specify WBN specific reference information and format.

8. Editorial change to delete "calibration". Other procedures, e.g. ACOT procedure also look at setpoint tolerance.

GLOBAL ACOT TO COT



## 3.3 INSTRUMENTATION

### 3.3.7 Control Room Emergency ~~Filtration~~ Ventilation System (CRE~~FS~~V) Actuation Instrumentation (1)

LCO 3.3.7 The CRE~~FS~~V actuation instrumentation for each Function in Table 3.3.7-1 shall be OPERABLE.

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

ACTIONS

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

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more Functions with one channel or train inoperable. | (3) A.1 ~~--------NOTE-------- Place in toxic gas protection mode if autoswapover to toxic gas protection mode is inoperable. ----------~~ Place one CRE~~FS~~V train in emergency [radiation protection] mode. | 7 days |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| B. One or more Functions with two channels or two trains inoperable. | ~~----------NOTE---------- Place in the toxic gas protection mode if autoswapover to toxic gas protection mode is inoperable. ----------~~ | |
| | B.1.1. Place one CREFS train in emergency [radiation protection] mode. | Immediately |
| | AND | |
| | B.1.2 Enter applicable Conditions and Required Actions for one CREFS train made inoperable by inoperable CREFS actuation instrumentation. | Immediately |
| | OR | |
| | B.2 Place both trains in emergency [radiation protection] mode. | Immediately |
| C. Required Action and associated Completion Time for Condition A or B not met in MODE 1, 2, 3, or 4. | C.1 Be in MODE 3. | 6 hours |
| | AND | |
| | C.2 Be in MODE 5. | 36 hours |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| D. Required Action and associated Completion Time for Condition A or B not met during movement of irradiated fuel assemblies or during CORE ALTERATIONS. | D.1 Suspend CORE ALTERATIONS. | Immediately |
| | AND | |
| | D.2 Suspend movement of irradiated fuel assemblies. | Immediately |
| ~~E. Required Action and associated Completion Time for Condition A or B not met in MODE 5 or 6.~~ | ~~E.1 Initiate action to restore one CREFS train to OPERABLE status.~~ | ~~Immediately~~ |

SURVEILLANCE REQUIREMENTS

----------NOTE----------
Refer to Table 3.3.7-1 to determine which SRs apply for each CREFS Actuation Function.
---

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.3.7.1 | Perform CHANNEL CHECK. | 12 hours |
| SR 3.3.7.2 | Perform ACOT. | 92 days |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.3.7.3 | Perform ACTUATION LOGIC TEST. | 31 days on a STAGGERED TEST BASIS |
| SR 3.3.7.4 | Perform MASTER RELAY TEST. | 31 days on a STAGGERED TEST BASIS |
| SR 3.3.7.5 | Perform SLAVE RELAY TEST. | [92] days |
| SR 3.3.7.~~6~~3 | Perform TADOT. | [18] months |
| SR 3.3.7.~~7~~4 | Perform CHANNEL CALIBRATION. | [18] months |

Table 3.3.7-1 (page 1 of 1)
CREFS Actuation Instrumentation

| FUNCTION | REQUIRED CHANNELS | SURVEILLANCE REQUIREMENTS | TRIP SETPOINT |
|---|---|---|---|
| 1. Manual Initiation | 2 trains | SR 3.3.7.~~8~~ 3 | N/A |
| 7 2. Automatic Actuation Logic and Actuation Relays | 2 trains | SR 3.3.7.3<br>SR 3.3.7.4<br>SR 3.3.7.5 | N/A |
| Z ~~3~~ 4. Control Room Radiation | | | |
| ~~a. Control Room Atmosphere~~ | ~~[2]~~ | ~~SR 3.3.7.1~~<br>~~SR 3.3.7.2~~<br>~~SR 3.3.7.7~~ | ~~≤ [2] mR/hr~~ |
| b. Control Room Air Intakes | ~~[2]~~ 2 | SR 3.3.7.1<br>SR 3.3.7.2<br>SR 3.3.7.~~7~~ 4 | ≤ ~~[2]~~ mR/hr TBD |
| 3 ~~4~~. Safety Injection | Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiating Functions and requirements. | | |

B 3.3 INSTRUMENTATION

B 3.3.7 Control Room Emergency ~~Filtration~~ Ventilation System (CREFS) Actuation Instrumentation

BASES

BACKGROUND

The CREFS provides an enclosed control room environment from which the unit can be operated following an uncontrolled release of radioactivity. During normal operation, the Control ~~Auxiliary~~ Building Ventilation System provides control room ventilation. Upon receipt of an actuation signal the system initiates filtered ventilation and pressurization of the control room. This system is described in the Bases for LCO 3.7.10, "Control Room Emergency Filtration System."

The actuation instrumentation consists of redundant radiation monitors downstream from the normal pressurizing fans ~~in the air intakes and a control room area.~~ A high radiation signal from its associated any ~~of these~~ detectors will initiate ~~both~~ trains ~~of~~ CREFS. The control room operator can also initiate CREFS trains by manual switches in the control room. CREFS is also actuated by a Safety Injection (SI) signal. The SI Function is discussed in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."

APPLICABLE SAFETY ANALYSES

The control room must be kept habitable for the operators stationed there during accident recovery and post accident operations.

△ The CREFS acts to terminate the supply of unfiltered outside air to the control room, initiate filtration and pressurize the control room. These actions are necessary to ensure the control room is kept habitable for the operators stationed there during accident recovery and post accident operations by minimizing the radiation exposure of control room personnel.

In MODES 1, 2, 3, and 4, the radiation monitor actuation of CREFS is a backup for the SI signal actuation. This ensures initiation of CREFS during a loss of coolant accident or steam generator tube rupture.

The radiation monitor actuation of CREFS in MODES 5 and 6, and during movement of irradiated fuel assemblies, is the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

primary means to ensure control room habitability in the event of a fuel handling~~, or waste gas decay tank rupture~~ accident.

The CREFS actuation instrumentation satisfies Criterion 3 of the NRC Policy Statement.

LCO

The LCO requirements ensure that the instrumentation necessary to initiate CREFS is OPERABLE.

1. Manual Initiation

The LCO requires two channels OPERABLE. The operator can initiate CREFS at any time by using either of two switches in the control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO on Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

~~Each channel consists of one push button and the interconnecting wiring to the actuation logic cabinet.~~

2. Automatic Actuation Logic and Actuation Relays

The LCO requires two trains of Actuation Logic and relays OPERABLE to assure that no single random failure can prevent automatic actuation.

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b., SI, in LCO 3.3.2. The Applicable MODES and conditions specified for the CREFS portion of these Functions are different and less restrictive than those for their SI roles. If one or more of the SI Functions becomes inoperable in such a manner that only the CREFS Function is affected, the Conditions applicable to their SI isolation Function need not be entered. The less

(continued)

BASES

LCO

2. Automatic Actuation Logic and Actuation Relays (continued)

restrictive Actions specified for inoperability of the CREFS Functions specify sufficient compensatory measures for this case.

2 ~~3~~. Control Room Radiation

The LCO specifies ~~two required Control Room Atmosphere Radiation Monitors and~~ two required Control Room Intake Radiation Monitors to ensure that the radiation monitoring instrumentation necessary to initiate CREFS remains OPERABLE.

For sampling systems, channel OPERABILITY involves more than OPERABILITY of channel electronics. OPERABILITY may also require correct valve lineups, sample pump operation, and filter motor operation as well as detector OPERABILITY if these supporting features are necessary for trip to occur under the conditions assumed by the safety analyses.

3 ~~4~~. Safety Injection

Refer to LCO 3.3.2, Function 1, for all initiating Functions and requirements.

APPLICABILITY

These CREFS Functions must be OPERABLE in MODES 1, 2, 3, 4, and during CORE ALTERATIONS and movement of irradiated fuel assemblies. ~~The Functions must also be OPERABLE in MODES [5 and 6] when required for a waste gas decay tank rupture accident~~ to ensure a habitable environment for the control room operators.

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather

(continued)

BASES

ACTIONS (continued)

than a total loss of function. This determination is generally made during the performance of an ~~A~~COT, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS indicating that separate Condition entry is allowed for each Function. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.7-1 in the accompanying LCO. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from time the Condition was entered for that Function.

A.1

This action addresses the actuation logic train Function of CREFS, the radiation monitor channel Functions, and manual channel Functions.

If one train is inoperable, or one radiation monitoring channel is inoperable in one or more Functions, 7 days are permitted to restore it to OPERABLE status. The 7 day Completion Time is the same as is allowed if one train of the mechanical portion of the system is inoperable. The basis for this Completion Time is the same as provided in LCO 3.7.10. If the channel/train cannot be restored to OPERABLE status, one CREFS train must be placed in the emergency radiation protection mode of operation. This accomplishes the actuation instrumentation Function and places the unit in a conservative mode of operation.

~~The Required Action for Condition A is modified by a Note, which requires placing one CREFS train in the toxic gas protection mode instead of the [radiation protection] mode of operation if the autoswapover to toxic gas protection mode is inoperable. This ensures the CREFS train is placed in the most conservative mode of operation relative to the OPERABILITY of the associated actuation instrumentation.~~

(continued)

BASES

ACTIONS (continued)

B.1.1, B.1.2, and B.2

Condition B applies to the failure of two CREFS actuation trains, two radiation monitor channels or two manual channels. The Required Action is to place one CREFS train in the emergency [radiation protection] mode of operation immediately. This accomplishes the actuation instrumentation Function that may have been lost and places the unit in a conservative mode of operation. The Applicable Conditions and Required Actions of LCO 3.7.10 must also be entered for the CREFS train made inoperable by the inoperable actuation instrumentation. This ensures appropriate limits are placed upon train inoperability as discussed in the Bases for LCO 3.7.10.

Alternatively, both trains may be placed in the [radiation protection] mode. This ensures the CREFS function is performed even in the presence of a single failure.

3

~~The Required Actions for Condition B are modified by a Note, which requires placing one CREFS train in the toxic gas protection mode instead of the [radiation protection] mode of operation if the autoswapover to toxic gas protection mode is inoperable. This ensures the CREFS train is placed in the most conservative mode of operation relative to the OPERABILITY of the associated actuation instrumentation.~~

C.1 and C.2

Condition C applies when the Required Action and associated Completion Time for Condition A or B have not been met and the unit is in MODE 1, 2, 3, or 4. The unit must be brought to a MODE in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1 and D.2

Condition D applies when the Required Action and associated Completion Time for Condition A or B have not been met

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

during CORE ALTERATIONS or when irradiated fuel assemblies are being moved. Movement of irradiated fuel assemblies and CORE ALTERATIONS must be suspended immediately to reduce the risk of accidents that would require CREFS actuation.

3

~~E.1~~

~~Condition E applies when the Required Action and associated Completion Time for Condition A or B have not been met in MODE 5 or 6. Actions must be initiated to restore the inoperable train(s) to OPERABLE status immediately to ensure adequate isolation capability in the event of a waste gas decay tank rupture.~~

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.7-1 determines which Surveillances apply to which CREFS Functions.

SR 3.3.7.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or even something more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.7.1 (continued)

The Frequency is based upon operating experience that demonstrates channel failure is rare. Thus, performance of the CHANNEL CHECK ensures that undetected overt channel failure is limited to 12 hours. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.7.2

An ACOT is performed once every 92 days on each required channel to ensure the entire channel will perform the intended function. This test verifies the capability of the instrumentation to provide the CREFS actuation. The setpoints shall be left consistent with the unit specific calibration procedure tolerance. The Frequency is based upon the known reliability of the monitoring equipment and has been shown to be acceptable through operating experience.

SR 3.3.7.3

SR 3.3.7.3 is the performance of an ACTUATION LOGIC TEST. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and there is an intact voltage signal path to the master relay coils. This test is performed every 31 days on a STAGGERED TEST BASIS. The Frequency is justified in ~~WCAP-10271-P-A, Supplement 2, Rev. 1~~ (Ref. ~~1~~ 2).

SR 3.3.7.4

SR 3.3.7.4 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.7.4 (continued)

coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity. This test is performed every 31 days on a STAGGERED TEST BASIS. The Frequency is justified in Reference 1.

7

SR 3.3.7.5

SR 3.3.7.5 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation MODE is either allowed to function or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment that may not be operated in the design mitigation MODE is prevented from operation by the SLAVE RELAY TEST circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay. This test is performed every [92] days. The Frequency is justified in Reference 1.

SR 3.3.7.~~6~~ 3

SR 3.3.7.6 is the performance of a TADOT. This test is a check of the Manual Actuation Functions and is performed every [18] months. Each Manual Actuation Function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.).

7

The test also includes trip devices that provide actuation signals directly to the Solid State Protection System, bypassing the analog process control equipment. For these tests, the relay trip setpoints are verified and adjusted as necessary. The Frequency is justified in Reference 1.

SR 3.3.7.~~7~~ 4

A CHANNEL CALIBRATION is performed every [18] months, or approximately at every refueling. CHANNEL CALIBRATION is a

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.7.7 (continued)

complete check of the instrument loop including the sensor. The test verifies that the channel responds to measured parameter within the necessary range and accuracy.

The Frequency is based upon operating experience and is consistent with the typical industry refueling cycle.

REFERENCES 5 ~~1. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.~~

1. Watts Bar FSAR, Section 9.4, Control Room Area Ventilation System.

2. WCAP-10271-P-A, Supplement 2, Rev. 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," ~~dated June 1990.~~ March 1987

## JUSTIFICATIONS FOR DEVIATIONS FROM STS 3.3.7

1. The proper terminology for Watts Bar, Unit 1 is "Ventilation" System.

2. In general, the CREVS does not need to be OPERABLE in Modes 5 and 6 for Watts Bar. The only exception to this is when irradiated fuel is being moved or when CORE ALTERATIONS take place.

3. The CREVS for Watts Bar does not have a "toxic gas protection mode".

4. The radiation monitors are only located in the air intakes to the control room and not in control room itself.

5. Change to specify WBN specific reference information and format.

6. Editorial change to delete "calibration". Other procedures, e.g. ACOT procedure also look at setpoint tolerance.

7. SRs 3.3.7.3, 3.3.7.4, and 3.3.7.5 have been deleted because old Fuction 4, SI, references LCO 3.3.2, Function 1 (SI) and the Surveillances for LCO 3.3.2, Function 1, test the Actuation Logic and Relays.

   The control room ventilation actuation by radiation monitors is separate from the SSPS, therefore there is no MASTER RELAY TEST and SLAVE RELAY TEST. The relays for control room ventilation by radiation monitors are tested during the COT.

   Due to these deletions the remaining SRs have been renumbered in the Surveillance table and Table 3.3.7-1.

8. Modified to reflect WBN plant specific terminology/design.

Δ. A triangle is used to indicate a place with editorial differences.

## 3.3 INSTRUMENTATION

### 3.3.8 ~~Fuel~~ Auxiliary Building ~~Air Cleanup~~ Gas Treatment System (~~FBACS~~ ABGTS) Actuation Instrumentation (B)

LCO 3.3.8 The ~~FBACS~~ ABGTS actuation instrumentation for each Function in Table 3.3.8-1 shall be OPERABLE. (B)

APPLICABILITY: [MODES 1, 2, 3, and 4,] (2)
During movement of irradiated fuel assemblies in the fuel ~~building~~. Handling area. (B)

ACTIONS

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

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more Functions with one channel or train inoperable. | A.1 Place one ~~FBACS~~ ABGTS train in operation. (B) | 7 days |
| B. One or more Functions with two channels or two trains inoperable. | B.1.1 Place one ~~FBACS~~ ABGTS train in operation. (B) | Immediately |
| | AND | |
| | B.1.2 Enter applicable Conditions and Required Actions of LCO 3.7.~~13~~12 "~~Fuel~~ Auxiliary Building ~~Air Cleanup~~ Gas Treatment System ~~(FBACS)~~ ABGTS," for one train made inoperable by inoperable actuation instrumentation. (3) (B) | Immediately |
| | OR | |
| | | (continued) |

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| B. (continued) | B.2 | Place both trains in emergency {radiation protection} mode. | Immediately |
| C. Required Action and associated Completion Time for Condition A or B not met during movement of irradiated fuel assemblies in the fuel building. | C.1 | Suspend movement of irradiated fuel assemblies in the fuel building. | Immediately |
| D. Required Action and associated Completion Time for Condition A or B not met in MODE 1, 2, 3, or 4. | D.1<br>AND<br>D.2 | Be in MODE 3.<br><br>Be in MODE 5. | 6 hours<br><br>36 hours |

SURVEILLANCE REQUIREMENTS

----------NOTE----------
Refer to Table 3.3.8-1 to determine which SRs apply for each ABGTS ~~FBACS~~ Actuation Function.
---

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.3.8.1 | Perform CHANNEL CHECK. | 12 hours |
| SR 3.3.8.2 | Perform ACOT. | 92 days |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | FREQUENCY |
|---|---|
| ~~SR 3.3.8.3 Perform ACTUATION LOGIC TEST.~~ ~~8~~ 6 | ~~31 days on a STAGGERED TEST BASIS~~ |
| SR 3.3.8.~~4~~3 Perform TADOT. | [18] months |
| SR 3.3.8.~~5~~4 Perform CHANNEL CALIBRATION. | [18] months |

Table 3.3.8-1 (page 1 of 1)
~~FBACS~~ Actuation Instrumentation (B)
ABGTS

| FUNCTION | REQUIRED CHANNELS | SURVEILLANCE REQUIREMENTS | TRIP SETPOINT |
|---|---|---|---|
| 1. Manual Initiation | 2 | SR 3.3.8.~~2~~3 | N/A |
| 2. Automatic Actuation Logic and Actuation Relays | 2 trains | SR 3.3.8.3 | N/A |
| ~~3.~~ 2 Fuel ~~Building~~ POOL AREA Radiation MONITORS | 2 | | |
| ~~a. Gaseous~~ | ~~[2]~~ | SR 3.3.8.1<br>SR 3.3.8.2<br>SR 3.3.8.~~5~~4 | ≤ ~~[X]~~ TBD mR/hr (A) (B) |
| ~~b. Particulate~~ | ~~[2]~~ | ~~SR 3.3.8.1~~<br>~~SR 3.3.8.2~~<br>~~SR 3.3.8.5~~ | ~~≤ [2] mR/hr~~ |
| ~~3. Safety Injection~~ Containment Isolation Phase A | Refer to LCO 3.3.2, Function ~~A~~ 3a, for all initiating Functions and requirements. | | |

B 3.3 INSTRUMENTATION

B 3.3.8 AUXILIARY ~~Fuel~~ Building GAS TREATMENT ~~Air Cleanup~~ System (ABGTS ~~FBACS~~) Actuation Instrumentation (B)

BASES

BACKGROUND

(B) exhaust of air from (B) GAS TREATMENT ~~of~~ the fuel ~~building~~ (A) HANDLING AREA, ECCS PUMP ROOMS, AND PENETRATION ROOMS automatically following receipt of a high radiation signal ~~(gaseous or particulate)~~ or a ~~safety injection (SI)~~ signal. Initiation may also be performed manually as needed from the main control room.

The ABGTS ~~FBACS~~ ensures that radioactive materials in the fuel building atmosphere following a fuel handling accident or a loss of coolant accident (LOCA) are filtered and adsorbed prior to exhausting to the environment. The system is described in the Bases for LCO 3.7.~~13~~12, "~~Fuel~~ AUXILIARY Building ~~Air Cleanup~~ System." The system initiates filtered ~~ventilation~~ (B) FUEL POOL AREA

High ~~gaseous and particulate~~ AREA radiation ABGTS ~~each~~ monitored by either of two monitors provides ~~FBACS~~ initiation. Each ABGTS ~~FBACS~~ train is initiated by high radiation detected by a channel dedicated to that train. There are a total of two channels, one for each train. ~~Each channel contains a gaseous and particulate monitor.~~ High radiation detected by any monitor or ~~an SI~~ signal from the Engineered Safety Features Actuation System (ESFAS) initiates ~~fuel~~ AUXILIARY building isolation and starts the ~~FBACS.~~ These actions function to prevent exfiltration of contaminated air by initiating filtered ventilation, which imposes a negative pressure on the ~~fuel building.~~ ~~Since the radiation monitors include an air sampling system, various components such as sample line valves, sample line heaters, sample pumps, and filter motors are required to support monitor OPERABILITY.~~

(A) PHASE A ISOLATION Containment (B)

(A) PHASE A (B) ABGTS (A)(B) AUXILIARY BUILDING SECONDARY CONTAINMENT ENCLOSURE (ABSCE)

APPLICABLE SAFETY ANALYSES

The ABGTS ~~FBACS~~ ensures that radioactive materials in the ~~fuel building~~ ABSCE atmosphere following a fuel handling accident or a LOCA are filtered and adsorbed prior to being exhausted to the environment. This action reduces the radioactive content in the (B) AUXILIARY ~~fuel~~ building exhaust following a LOCA or fuel handling accident so that offsite doses remain within the limits specified in 10 CFR 100 (Ref. 1).

The ABGTS ~~FBACS~~ actuation instrumentation satisfy Criterion 3 of the NRC Policy Statement. (B)

(continued)

BASES (continued)

LCO

The LCO requirements ensure that instrumentation necessary to initiate ~~FBACS~~ ABGTS (B) is OPERABLE.

Two channels of the ~~FBACS~~ ABGTS (B) manual initiation, two trains of automatic actuation logic, and two channels of ~~each~~ AREA (A) radiation monitors are required OPERABLE to ensure automatic isolation and initiation of filtered ventilation. ~~FBACS~~ ABGTS (B) actuation instrumentation is OPERABLE when all channel and train components necessary to provide ~~a FBACS~~ AN ABGTS (B) actuation signal are functional and in service.

(5) SPECIFIED IS ASSUMED TO BE ABOVE LOCAL BACKGROUND RADIATION AND IS MORE CONSERVATIVE THAN THE ANALYTICAL LIMIT ASSUMED IN THE SAFETY ANALYSIS IN ORDER TO

Only the Trip Setpoint is specified for ~~each FBACS~~ THE FUEL POOL AREA RADIATION MONITORS (A) Function in the LCO. The Trip Setpoint ~~limits~~ account for instrument uncertainties, which are defined in the Unit Specific Setpoint Calibration Procedure (Ref. 2). △

APPLICABILITY

The manual and Phase A ~~automatic~~ ABGTS ~~FBACS~~ Actuation Instrumentation (B) trains must be OPERABLE in MODES {1, 2, 3, and 4} ~~and when moving irradiated fuel assemblies in the fuel building,~~ to ensure the FBACS operates to remove fission products associated with leakage after a LOCA or a fuel handling accident.

High radiation initiation of the ABGTS ~~FBACS~~ is also required in (B) MODES {1, 2, 3, and 4} to remove fission products caused by post LOCA Emergency Core Cooling Systems leakage. The high radiation Function must also be OPERABLE in any MODE during movement of irradiated fuel assemblies in the fuel building to ensure automatic initiation of the ~~FBACS~~ ABGTS (B) when the potential for a fuel handling accident exists.

(B) ABGTS While in MODES 5 and 6 without fuel handling in progress, the ~~FBACS~~ instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and manual operator action is sufficient to prevent offsite dose limits from being exceeded.

ACTIONS

The most common cause of channel inoperability is outright failure or drift ~~of the bistable or process module~~ (A) sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a

(continued)

BASES

ACTIONS (continued)

total loss of function. This determination is generally made during the performance of an ACOT, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the tolerance specified by the ~~calibration~~ procedure, the channel must be declared inoperable immediately and the appropriate Condition entered. (A) (9)

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.8-1 in the accompanying LCO. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

(6) From the Phase A isolation

A.1

This action addresses the actuation logic train Function ~~of the Solid State Protection System (SSPS)~~, the radiation monitor channel Functions, and the manual channel Function. Condition A applies to the failure of a single actuation logic train, radiation monitor channel, or manual channel. If one channel or train is inoperable, a period of 7 days is allowed to restore it to OPERABLE status. If the train cannot be restored to OPERABLE status, one ~~FBACS~~ ABGTS train must be placed in operation. This accomplishes the actuation instrumentation Function and places the unit in a conservative mode of operation. The 7 day Completion Time is the same as is allowed if one train of the mechanical portion of the system is inoperable. The basis for this time is the same as that provided in LCO 3.7.~~13~~12. (B) (3)

(6) Signals from the Phase A isolation (4)

B.1.1, B.1.2, B.2

Condition B applies to the failure of two ~~FBACS~~ ABGTS actuation logic ~~trains~~, two radiation monitors, or two manual channels. The Required Action is to place one ~~FBACS~~ ABGTS train in operation immediately. This accomplishes the actuation instrumentation function that may have been lost and places the unit in a conservative mode of operation. The applicable Conditions and Required Actions of LCO 3.7.~~13~~12 must also be entered for the ~~FBACS~~ ABGTS train made inoperable by (B) (3) (B)

(continued)

BASES

ACTIONS

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

the inoperable actuation instrumentation. This ensures appropriate limits are placed on train inoperability as discussed in the Bases for LCO 3.7.~~13~~.2 (3)

(B) Alternatively, both trains may be placed in the emergency [radiation protection] mode. This ensures the ~~FBACS~~ ABGTS (B) Function is performed even in the presence of a single failure.

C.1

Condition C applies when the Required Action and associated Completion Time for Condition A or B have not been met and irradiated fuel assemblies are being moved in the fuel building. Movement of irradiated fuel assemblies in the fuel building must be suspended immediately to eliminate the potential for events that could require ~~FBACS~~ ABGTS actuation. (B)

(7) Performance of these actions shall not preclude moving a Component to a safe position.

D.1 and D.2

Condition D applies when the Required Action and associated Completion Time for Condition A or B have not been met and the unit is in MODE 1, 2, 3, or 4. The unit must be brought to a MODE in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. △

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.8-1 determines which Surveillances apply to which (B) ~~FBACS~~ ABGTS Actuation Functions.

SR 3.3.8.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1 (continued)

CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or even something more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

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

The Frequency is based upon operating experience that demonstrates channel failure is rare. Thus, performance of the CHANNEL CHECK ensures that undetected overt channel failure is limited to 12 hours. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.8.2

(1) ~~An~~ ACOT is performed once every 92 days on each required channel to ensure the entire channel will perform the (4) intended ~~function~~. This test verifies the capability of the instrumentation to provide the ~~FBACS~~ actuation. The ABGTS (B) (9) setpoints shall be left consistent with the unit specific calibration procedure tolerance. The Frequency of 92 days is based upon the known reliability of the monitoring equipment and has been shown to be acceptable through operating experience.

(6) (8) SR 3.3.8.3

SR 3.3.8.3 is the performance of an ACTUATION LOGIC TEST. The actuation logic is tested every 31 days on a STAGGERED

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.3 (continued)

(6) (B) TEST BASIS. All possible logic combinations, with and without applicable permissive, are tested for each protection function. The Frequency is based upon the known reliability of the relays and controls, the multichannel redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.8.4

SR 3.3.8.4 is the performance of a TADOT. This test is a check of the manual actuation functions and is performed every 18 months. Each manual actuation function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (e.g., pump starts, valve cycles). The Frequency is based upon operating experience and is consistent with the typical industry refueling cycle. (B)

SR 3.3.8.5

A CHANNEL CALIBRATION is performed every {18} months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop including the sensor. The test verifies that the channel responds to measured parameter within the necessary range and accuracy. The Frequency is based upon operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. 10 CFR 100.11, 1973.
2. WATTS BAR Unit Specific Setpoint Calibration Procedure. (B)

# JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.3.8

## Generic Justifications

Δ Minor editorial difference not marked up.

A. Modified to reflect WBN plant specific design.

B. Modified to reflect WBN plant specific terminology.

## Specific Justifications

1. The defined term "ACOT" in Section 1.1, Definitions, has been changed to "COT" and has been changed in this Instrumentation specification to reflect the Definitions change.

2. MODES 1, 2, 3, and 4 are applicable to the Auxiliary Building Gas Treatment System, since the system must actuate in response to a LOCA.

3. Editorial change. Reference to LCO 3.7.13 has been changed to LCO 3.7.12, consistent with the WBN specification numbering.

4. Editorial change. The word "function" has been capitalized, "Function", consistent with the normal practice in Technical Specifications.

5. Additional detail describing the Trip Setpoint value has been added to clarify the basis for the value.

6. SR 3.3.8.3 has been deleted, Function 2 of Table 3.3.8-1 has been deleted, and a Function 3 has been added. These changes have been made to reflect Watts Bar design. The ABGTS isolation from radiation monitors and manual does not interface with the SSPS, only the Phase A isolation does.

   ABGTS is actuated by an SI Phase A signal. Therefore, the actuation logic and relays are tested in LCO 3.3.2.

   Renumbering and Bases changes have also been made.

7. Wording has been added to the Bases discussion of Condition C. This wording clarifies the intent of Required Action and is consistent with the Refueling section 3.9.

8. This change reflects comments made by the industry to the NRC at the RSTS proof and review meeting in Irvine, CA, July 13 - 20, 1992.

9. Editorial change to delete "calibration". Other procedures, e.g. ACOT procedure also look at setpoint tolerance.

## 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 ≥ [~~2200~~] [2216] psig;

b. RCS average temperature ≤ [~~581~~] [592.9] °F; and

c. RCS total flow rate ≥ [~~284,000~~] 390,000 gpm.

1

APPLICABILITY: MODE 1.

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

a. THERMAL POWER ramp ~~in excess of~~ > 5% RTP per minute; or

b. THERMAL POWER step ~~in excess of~~ > 10% RTP.

---

A

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 (1) | Verify pressurizer pressure is ≥ [~~2200~~] [2216] psig. | 12 hours |
| SR 3.4.1.2 (1) | Verify RCS average temperature is ≤ [~~581~~] [592.9] °F. | 12 hours |
| SR 3.4.1.3 (1) | Verify RCS total flow rate is ≥ [~~284,000~~] [400,140] gpm. | 12 hours |
| SR 3.4.1.4 (1) | ----------NOTE---------- Not required to be ~~met~~ Performed (A) until 24 hours after ≥ ~~reaching~~ 90% RTP. ---------- Verify by precision heat balance that RCS total flow rate is ≥ [~~284,000~~] 390,000 gpm. | [18] (2) months |

# B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady-state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed.

The RCS pressure limit is consistent with operation within the nominal operational envelope. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit. A lower pressure will cause the reactor core to approach DNB limits.

The RCS coolant average temperature limit is consistent with full power operation within the nominal operational envelope. Indications of temperature are averaged to determine a value for comparison to the limit. A higher average temperature will cause the core to approach DNB limits.

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. Flow rate indications are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the core to approach DNB limits.

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB-limited event.

APPLICABLE SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for DNB-limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will

(continued)

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)."

The pressurizer pressure limit of [~~2200~~] 2216 psig and the RCS average temperature limit of [581]°F 592.9 correspond to analytical limits of [2205] psig and [595]°F used in the safety analyses, ~~with~~ which include allowance for measurement uncertainty. [2204 psia] 594.7 The RCS DNB parameters satisfy Criterion 2 of the NRC Policy Statement. (1)

LCO

Replace with INSERT "LCO" (attached) (A)

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 [2.0]%.

The LCO numerical values for pressure, temperature, and flow rate are given for the measurement location but have not been adjusted for instrument error.

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady-state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB-limited transient. In all other MODES, the power level is low enough that DNBR is not a concern. (3)

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short-term operational

(continued)

INSERT "LCO"

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 [~~2.0~~] 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 [~~2.1~~] [2.6] % for no fouling.

ED

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.

The LCO numerical values for pressure, temperature, and flow rate are given for the measurement location but have not been adjusted for instrument error.

BASES

APPLICABILITY (continued)

transients such as a THERMAL POWER ramp increase ~~in excess of~~ > 5% RTP per minute or a THERMAL POWER step increase ~~in excess of~~ > 10% RTP. These conditions represent short-term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels less than 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations. A

Another set of limits on DNB-related parameters is provided in SL 2.1.1, "Reactor Core Safety Limits." Those limits are less restrictive than the limits of this LCO, but violations of a Safety Limit (SL) merits a stricter, more severe Required Action. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded. A 3

ACTIONS

A.1

RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within LCO limits, action must be taken to restore parameter(s).

RCS total flow rate is not a controllable parameter and is not expected to vary during steady-state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced, as required by Required Action B.1, to restore DNB margin and eliminate the potential for violation of the accident analysis bounds.

The 2-hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause for the off-normal condition, and to restore the readings within limits and is based on plant operating experience.

B.1

If Required Action A.1 is not met within the associated 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 2 within 6 hours. In

(continued)

BASES

ACTIONS

B.1 (continued)

MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds. The Completion Time of 6 hours is reasonable, to reach the required plant conditions in an orderly manner in conjunction with an even control of steam generator heat removal.

3 A

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12-hour Surveillance Frequency for pressurizer pressure is sufficient to ensure the pressure can be restored to a normal operation, steady-state condition following load changes and other expected transient operations. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.2

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12-hour Surveillance Frequency for RCS average temperature is sufficient to ensure the temperature can be restored to a normal operation, steady-state condition following load changes and other expected transient operations. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess for 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

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.

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 required to be ~~not within~~ 24 hours after ~~reaching~~ 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, Chapter 15, "Accident Analyses," ~~FSAR, Section [15]~~ Sections 15.2 and 15.3.4.

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 3.4.1

A. This change reflects comments made by the industry to the NRC at the RSTS P&R meeting held in Irvine, CA July 13 - July 20, 1992.

1. Change to reflect Watts Bar specific parameter values.

2. Format change to delete brackets that identify plant specific information/values.

3. Change to correct errors in the STS.

4. Change to specify WBN specific reference information and format.

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.2 RCS Minimum Temperature for Criticality

1

LCO 3.4.2 Each RCS loop average temperature ($T_{avg}$) shall be ≥ ~~[541]~~ 551 °F.

APPLICABILITY: MODE 1 [with $T_{avg}$ in one or more RCS loops < ~~[551]~~ 561 °F], 2
MODE 2 [with $T_{avg}$ in one or more RCS loops < ~~[551]~~ 561 °F] ~~and~~ with A
$K_{eff}$ ≥ 1.0.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. $T_{avg}$ in one or more RCS loops not within limit. | A.1 Be in MODE 3. | 30 minutes |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE 1 | FREQUENCY |
|---|---|
| SR 3.4.2.1 Verify RCS $T_{avg}$ in each loop ≥ ~~[541]~~ 551 °F. | Within 15 minutes prior to achieving criticality<br>AND<br>1 2<br>-----NOTE------<br>Only required if [$T_{avg}$ - $T_{ref}$ deviation, low-low $T_{avg}$] alarm not reset<br>----------<br>30 minutes |

A and any RCS loop Tavg < 557°F

# 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.

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

APPLICABLE SAFETY ANALYSES

The minimum temperature for criticality limitation provides a small band, 6°F, for critical operation below hot zero power (HZP). This band allows critical operation slightly

3 A

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

3 A

below HZP temperature 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 temperature and the minimum temperature for criticality. A

hot zero power

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

557 1

All low power safety analyses assume initial RCS loop temperatures ≥ the HZP temperature of ~~547~~°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.

3

The temperature for 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 present a challenge to, the integrity of a fission product barrier. A 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.

(continued)

BASES (continued)

APPLICABILITY

In MODES 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. In addition, reactor criticality is permitted only when any RCS loop average temperature is > ~~[541]~~ 551 °F and the $T_{avg}$ - $T_{ref}$ deviation ~~or low-low $T_{avg}$~~ alarm is ~~in an alarm state.~~ reset. In the range of ~~[541]~~ 551 °F, at the minimum temperature for criticality, to ~~547~~ 557 °F, there is a potential for RCS loop average temperature to fall below the LCO requirement. Below ~~547~~ 557 °F, $T_{ref}$ is essentially constant and equal to ~~547~~ 557 °F ($T_{no\ load}$). Therefore, a $T_{avg}$ - $T_{ref}$ deviation ~~or low-low $T_{avg}$~~ alarm would be due to the movement of RCS loop average temperature below $T_{no\ load}$ and would alarm ~~1~~ 3 °F above the minimum temperature for criticality. As power level increases, each RCS loop average temperature increases to a value far enough above ~~547~~ 557 °F that the potential for RCS loop average temperatures to fall below ~~[541]~~ 551 °F is so diminished that this LCO is no longer applicable.

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 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.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

551 (1)

When the Tavg-Tref deviation alarm is not reset and any RCS loop Tavg < 557°F.

RCS loop average temperature is required to be verified at or above ~~[541]~~°F within 15 minutes prior to achieving criticality and every 30 minutes ~~thereafter~~. 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.

557 (1)

The Note modifies the SR. When any RCS loop average temperature is less than ~~[547]~~°F and the $\Delta T_{avg} - T_{ref}$ deviation, low-low $T_{avg}$ ] 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.

(1)

REFERENCES

(4)

1. ~~FSAR, Section [15.0.3]~~
Watts Bar FSAR, Chapter 15, "Accident Analyses,"

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 3.4.2

A. This change reflects comments made by the industry to the NRC at the RSTS P&R meeting held in Irvine, CA July 13 - July 20, 1992.

1. Change to reflect WBN specific parameter values or information.

2. Format change to delete brackets that identify plant specific information/values.

3. Change to delete text duplication in the STS.

4. Change to specify WBN specific reference information and format.

5. Change to correct error(s) in the STS.

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.3 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.3 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. ---------NOTE--------- Required Action A.2 shall be completed whenever this Condition is entered. ---------- Requirements of LCO not met, in MODES 1, 2, 3 or 4. | A.1 Restore parameter(s) to within limits. | 30 minutes |
| | AND | |
| | A.2 Determine RCS is acceptable for continued operation. | 72 hours |
| B. Required Action and associated Completion Time of Condition A not met. | B.1 Be in MODE 3. | 6 hours |
| | AND | |
| | B.2 Be in MODE 5 with RCS pressure < ~~(500)~~ psig. (1) | 36 hours |
| C. ----NOTE---- Required Action C.2 shall be Completed whenever this Condition is entered. ---------- Requirements of LCO not met anytime in other than MODES 1, 2, 3, or 4. | C.1 Initiate action to restore parameter(s) to within limits | Immediately |
| | AND | |
| | C.2 Determine RCS is acceptable for continued operation | Prior to entering MODE 4 |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.3.1 (A) | ----------NOTE---------- Only required to be ~~met~~ performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ---------- | |
| | Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the PTLR. | 30 minutes |

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

## B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The PTLR contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature (Ref. 1).

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2) requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 3).

The neutron embrittlement effect on the material toughness is reflected by increasing the nil-ductility reference temperature ($RT_{NDT}$) as exposure to neutron fluence increases.

(continued)

BASES

BACKGROUND (continued)

The actual shift in the $RT_{NDT}$ of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 4) and Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99 (Ref. 6).

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limit curve includes the Reference 2 requirement that it be no less than 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a non-isolable leak or loss-of-coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 7), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate-of-change conditions that might cause undetected flaws to propagate and cause non-ductile failure of the RCPB, an unanalyzed condition. Reference 1 establishes the methodology for determining the P/T limits. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

The two elements of this LCO are:

a. The limit curves for heatup, cooldown, and ISLH testing; and

b. Limits on the rate of change of temperature.

The LCO limits apply to all components of the RCS, except the pressurizer. These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to non-ductile failure.

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and ISLH testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follow:

a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature;

(continued)

BASES

LCO (continued)

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T Limits LCO provides a definition of acceptable operation for prevention of non-ductile failure in accordance with 10 CFR 50, Appendix G (Ref. 2). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, their applicability is at all times in keeping with the concern for non-ductile failure. The limits do not apply to the pressurizer.

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; LCO 3.4.2, "RCS Minimum Temperature for Criticality"; and Safety Limit 2.1, "Safety Limits," also provide operational restrictions for pressure and temperature and maximum pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for non-ductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

ACTIONS

The actions of this LCO consider the premise that a violation of the limits occurred during normal plant maneuvering. Severe violations caused by abnormal transients, at times accompanied by equipment failures, may also require additional actions from emergency operating procedures.

A.1 and A.2

Operation outside the P/T limits during MODES 1, 2, 3 or 4. must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses. (A)

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

The 30-minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with preanalyzed transients in the stress analyses, new 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.

The 72-hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event-specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

Condition A is modified by a Note requiring Required Action A.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 A.1 is insufficient because higher-than-analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. In reduced

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

pressure and temperature conditions, the possibility of propagation with undetected flaws is decreased.

If the required restoration activity cannot be accomplished within 30 minutes, Required Action B.1 and Required Action B.2 must be implemented to reduce pressure and temperature.

If the required evaluation for continued operation cannot be accomplished within 72 hours or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Required Action B.1 and Required Action B.2. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions.

Pressure and temperature are reduced by bringing the plant to MODE 3 within 6 hours and to MODE 5 with RCS pressure < [500] psig within 36 hours.

1

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.

A

INSERT (attached)

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 permit~~s~~ 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.

A'

This SR is modified by a Note that only requires this SR to be ~~not~~ performed only during system heatup, cooldown, and ISLH testing. No

(continued)

INSERT

C.1 and C.2

Actions must be initiated immediately to correct operation outside of the P/T limits at times other than when in MODES 1, 2, 3, or 4, so that the RCPB is returned to a condition that has been verified by stress analysis.

The immediate Completion Time reflects the urgency of initiating action to restore the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished ~~in this time~~ in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed prior to entry into MODE 4. Several methods may be used, including comparison with preanalyzed 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.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1 (continued)

SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

REFERENCES

1. ~~WCAP-7924-A, April 1975.~~
2. ~~10 CFR 50, Appendix G.~~
3. ~~ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.~~
4. ~~ASTM E 185-82, July 1982.~~
5. ~~10 CFR 50, Appendix H.~~
6. ~~Regulatory Guide 1.99, Revision 2, May 1988.~~
7. ~~ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.~~

1. TVA Calc WBN-MTB-027 R0, "Pressure-Temperature Limits Based on Reg Guide 1.99 R2 for submittal to NRC" ~~(B46890501559)~~.
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."

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 3.4.3

A. This change reflects comments made by the industry to the NRC at the RSTS P&R meeting held in Irvine, CA July 13 - July 20, 1992.

1. Format change to delete brackets that identify plant specific information/values.

2. Change to specify WBN specific reference information and format.

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.4 RCS Loops – MODES 1 and 2

LCO 3.4.4 [Four] 2 RCS loops shall be OPERABLE and in operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Requirements of LCO not met. | A.1 Be in MODE 3. | 6 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.4.1 | Verify each RCS loop is in operation. | 12 hours |
| SR 3.4.4.2 | Verify steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program. | In accordance with the Steam Generator Tube Surveillance Program A |

Relocated to SR 3.4.13.2

# B 3.4 REACTOR COOLANT SYSTEM (RCS)

## B 3.4.4 RCS Loops – MODES 1 and 2

BASES

BACKGROUND

The primary function of the RCS is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs) to the secondary plant.

The secondary functions of the RCS include:

a. Moderating the neutron energy level to the thermal state, to increase the probability of fission;

b. Improving the neutron economy by acting as a reflector;

c. Carrying the soluble neutron poison, boric acid;

d. Providing a second barrier against fission-product release to the environment; and

e. Removing the heat generated in the fuel due to fission-product decay following a unit shutdown.

The reactor coolant is circulated through [four] loops connected in parallel to the reactor vessel, each containing an SG, a reactor coolant pump (RCP), and appropriate flow and temperature instrumentation for both control and protection. The reactor vessel contains the clad[ded] fuel. The SGs provide the heat sink to the isolated secondary coolant. The RCPs circulate the coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage. This forced circulation of the reactor coolant ensures mixing of the coolant for proper boration and chemistry control.

APPLICABLE SAFETY ANALYSES

Safety analyses contain various assumptions for the design bases accident initial conditions including RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of RCS loops in service.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Both transient and steady-state analyses have been performed to establish the effect of flow on the departure from nucleate boiling (DNB). The transient and accident analyses for the plant have been performed assuming [four] RCS loops are in operation. The majority of the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are most important to RCP operation are the [four]-pump coastdown, single pump locked rotor, single pump (broken shaft or coastdown), and rod withdrawal events (Ref. 1).

Steady-state DNB analysis has been performed for the [four] RCS loop operation. For [four] RCS loop operation, the steady-state DNB analysis, which generates the pressure and temperature Safety Limit (i.e., the departure from nucleate boiling ratio (DNBR) limit) assumes a maximum power level of ~~109%~~ 118 RTP. This is the design overpower condition for [four] RCS loop operation. The value for the accident analysis setpoint of the nuclear overpower (high flux) trip is ~~107%~~ 118 % and is based on an analysis assumption that bounds possible instrumentation errors. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.

The plant is designed to operate with all RCS loops in operation to maintain DNBR above the safety limit, during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant.

RCS loops – MODES 1 and 2 satisfy Criterion 2 of the NRC Policy Statement.

LCO

The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, [four] pumps are required ~~to be~~ at rated power.

An OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an

(continued)

BASES

LCO (continued)

OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program.

APPLICABILITY

In MODES 1 and 2, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, and 5.

Operation in other MODES is covered by:

LCO 3.4.5, "RCS Loops – MODE 3";
LCO 3.4.6, "RCS Loops – MODE 4";
LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level"; and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level."

ACTIONS

A.1

If the requirements of the LCO are not met, the Required Action is to reduce power and bring the plant to MODE 3. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits.

The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging safety systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

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

A

Relocate to 3.4.13

SR 3.4.~~4~~13.2

This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions.

REFERENCES

3

1. ~~FSAR, Section [ ].~~ Watts Bar FSAR, Chapter 15, "Accident Analyses."

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 3.4.4

A. This change reflects comments made by the industry to the NRC at the RSTS P&R meeting held in Irvine, CA July 13 - July 20, 1992.

1. Change to reflect WBN specific parameter values.

2. Format change to delete brackets that identify plant specific information/values.

3. Change to specify WBN specific reference information and format.

4. To correct error in STS.

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.5 RCS Loops – MODE 3

Rod Control System is capable of rod withdrawal

LCO 3.4.5 [Two] RCS loops shall be OPERABLE, and either:

2

a. [Two] RCS loops shall be in operation when the ~~reactor trip breakers are closed~~; or

b. One RCS loop shall be in operation when the ~~reactor trip breakers are open.~~ rod control system is not capable of rod withdrawal.

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

All reactor coolant pumps may be de-energized for ≤ 1 hour per 8-hour period provided:

a. No operations are permitted that would cause reduction of the RCS boron concentration; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature.

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APPLICABILITY: MODE 3.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One required RCS loop inoperable. | A.1 Restore required RCS loop to OPERABLE status. | 72 hours |
| B. Required Action and associated Completion Time of Condition A not met. | B.1 Be in MODE 4. | 12 hours |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| C. One required RCS loop not in operation, and reactor trip breakers closed and Rod Control System capable of rod withdrawal | C.1 Restore required RCS loop to operation. | 1 hour |
| | OR | |
| | C.2 Open reactor trip breakers. Deenergize all CRDMs. | 1 hour |
| D. All ~~[Two]~~ RCS loops inoperable. | D.1 Open reactor trip breakers. Deenergize all CRDMs | Immediately |
| OR | AND | |
| No RCS loop in operation. | D.2 Suspend all operations involving a reduction of RCS boron concentration. | Immediately |
| | AND | |
| | D.3 Initiate action to restore one RCS loop to OPERABLE status and operation. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.5.1 | Verify required number of RCS loops are in operation. | 12 hours |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.5.2 | Verify steam generator secondary-side water levels ≥ ~~[17]~~% for required OPERABLE RCS loops. | 12 hours |
| SR 3.4.5.3 | ----------NOTE---------- Only required to be met if reactor trip breakers are open. ---------- Verify correct breaker alignment and indicated power are available to the required pump that is not in operation. | 7 days |

# B 3.4 REACTOR COOLANT SYSTEM (RCS)

## B 3.4.5 RCS Loops – MODE 3

BASES

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BACKGROUND

In MODE 3, the primary function of the reactor coolant is removal of decay heat and transfer of this heat, via the steam generator (SG), to the secondary plant fluid. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

The reactor coolant is circulated through [four] RCS loops, connected in parallel to the reactor vessel, each containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The reactor vessel contains the clad~~ded~~ fuel. The SGs provide the heat sink. The RCPs circulate the water through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage.

In MODE 3, RCPs are used to provide forced circulation for heat removal during heatup and cooldown. The MODE 3 decay heat removal requirements are low enough that a single RCS loop with one RCP running is sufficient to remove core decay heat. However, [two] RCS loops are required to be OPERABLE to ensure redundant capability for decay heat removal.

---

APPLICABLE SAFETY ANALYSES

Whenever the reactor trip breakers are in the closed position and the control rod drive mechanisms (CRDM) energized, ~~the possibility of~~ an inadvertent rod withdrawal from subcritical, resulting in a power excursion, is possible. Such a transient could be caused by a malfunction of the rod control system. In addition, the possibility of a power excursion due to the ejection of an inserted control rod is possible with the breakers closed or open. Such a transient could be caused by the mechanical failure of a control rod drive mechanism housing.

Therefore, in MODE 3 with reactor trip breakers in the closed position and Rod Control System capable of rod withdrawal, accidental control rod withdrawal from subcritical is postulated and requires at least [two] RCS loops to be OPERABLE and in operation to ensure that the accident analyses limits are met. For those conditions when the ~~reactor trip breakers are open,~~ rod control system is not capable of rod withdrawal, two RCS loops are

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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.

3

2

LCO

A and Rod Control System capable of rod withdrawal

The purpose of this LCO is to require that at least [two] RCS loops be OPERABLE. In MODE 3 with the reactor trip breakers in the closed position, [two] RCS loops must be in operation. [Two] RCS loops are required to be in operation in MODE 3 with reactor trip breakers closed 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.

5

3

With the reactor trip breakers in the open position, 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.

A Or the CRDMs deenergized, the rod control system is not capable of rod withdrawl, therefore

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. Another test performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow.

4

(continued)

BASES

LCO (continued)

4

~~The no-flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits the de-energizing of the pumps in order to perform this test and validate the assumed analysis values. As with the validation of the pump coastdown curve, this test should be performed only once unless the flow characteristics of the RCS are changed.~~ The 1-hour time period specified is adequate to perform the desired tests, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of the Note is permitted provided the following conditions are met, along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration, thereby maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

A

3

An OPERABLE RCS loop consists of one OPERABLE RCP and one OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required. (and which has the minimum water level specified in SR 3.4.5.2)

APPLICABILITY

In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. The most stringent condition of the LCO, that is, two RCS loops OPERABLE and two RCS loops in operation, applies to MODE 3 with reactor trip breakers in the closed position. The least stringent condition, that is, two RCS loops OPERABLE and one RCS loop in operation, applies to MODE 3 with the reactor trip breakers open.

(continued)

BASES

APPLICABILITY (continued)

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops – MODES 1 and 2";
LCO 3.4.6, "RCS Loops – MODE 4";
LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level"; and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level."

ACTIONS

A.1

If one required RCS loop is inoperable, redundancy for heat removal is lost. The Required Action is restoration of the required RCS loop to OPERABLE status within the Completion Time of 72 hours. This time allowance is a justified period to be without the redundant, non-operating loop because a single loop in operation has a heat transfer capability greater than that needed to remove the decay heat produced in the reactor core and because of the low probability of a failure in the remaining loop occurring during this period.

B.1

If restoration is not possible within 72 hours, the unit must be brought to MODE 4. In MODE 4, the unit may be placed on the Residual Heat Removal System. The additional Completion Time of 12 hours is compatible with required operations to achieve cooldown and depressurization from the existing plant conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

A

and Rod Control System capable of rod withdrawal

A

deenergize all CRDMs by

or deenergize the motor generator sets.

If the required RCS loop is not in operation, and the reactor trip breakers are closed, the Required Action is either to restore the required RCS loop to operation or to opening the reactor trip breakers. When the reactor trip breakers are in the closed position, it is postulated that a power excursion could occur in the event of an inadvertent control rod withdrawal. This mandates having the heat

and Rod Control System capable of rod withdrawal

A

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

to restore the required RCS loop to operation or deenergize all CRDMs (A)

(2)

transfer capacity of two RCS loops in operation. If only one loop is in operation, the reactor trip breakers must be opened. The Completion Times of 1 hour ~~for opening the breakers~~ is adequate to perform ~~this~~ these (A) operations in an orderly manner without exposing the unit to risk for an undue time period.

D.1, D.2, and D.3

(1)

(A) all CRDMs must be deenergized by opening

If ~~[two]~~ ALL RCS loops are inoperable or no RCS loop is in operation, except as during conditions permitted by the Note in the LCO section, or deenergizing the MG sets (A) the reactor trip breakers ~~must be opened~~, all operations involving a reduction of RCS boron concentration must be suspended, and action to restore one of the RCS loops to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and opening the reactor trip breakers (A) removes the possibility of an inadvertent rod withdrawal. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification every 12 hours that the required number of loops are in operation. Verification includes flow rate, temperature, and 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 RCS loop performance.

SR 3.4.5.2

(1)

SR 3.4.5.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary-side narrow-range water level is ≥ ~~[17]~~ 10% for required ~~OPERABLE~~ (A) RCS loops. If the SG secondary-side narrow-range water

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.5.2 (continued) (1)

level is < ~~[17]~~ 10%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of the decay heat. The 12-hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.

SR 3.4.5.3

Verification that the required number of RCS pumps are OPERABLE ensures that safety analyses limits are met. The requirement also ensures that additional 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.

The Note indicates that the SR is not required to be met when reactor trip breakers are closed because both RCPs are in operation and not in standby. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable through operating experience. (A)

REFERENCES None.

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 3.4.5

A. This change reflects comments made by the industry to the NRC at the RSTS P&R meeting held in Irvine, CA July 13 - July 20, 1992.

1. Change to reflect WBN specific parameter values.

2. Format change to delete brackets that identify plant specific information/values.

3. Change to correct errors in the STS.

4. Watts Bar is not performing cold-no-flow rod drop time tests. The rod drop test will be performed at hot full flow conditions.

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.6 RCS Loops – MODE 4

LCO 3.4.6 Two loops consisting of any combination of RCS loops and residual heat removal (RHR) loops shall be OPERABLE, and one loop shall be in operation.

---NOTES---

4 ~~1. All reactor coolant pumps (RCPs) and RHR pumps may be de-energized for ≤ 1 hour per 8-hour period provided:~~

~~a. No operations are permitted that would cause reduction of the RCS boron concentration; and~~

~~b. Core outlet temperature is maintained at least 10°F below saturation temperature.~~

2. No RCP shall be started with any RCS cold leg temperature ≤ ~~[275]~~ 310 °F unless the secondary-side water temperature of each steam generator (SG) is ≤ ~~[50]~~ 2 °F above each of the RCS cold leg temperatures.

APPLICABILITY: MODE 4.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One required RCS loop inoperable.<br><br>AND<br><br>Two RHR loops inoperable. | A.1 Initiate action to restore a second loop to OPERABLE status. | Immediately |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| B. One required RHR loop inoperable.<br><br>AND<br><br>Two required RCS loops inoperable. | B.1 Be in MODE 5. | 24 hours |
| C. Required RCS or RHR loops inoperable.<br><br>OR<br><br>No RCS or RHR loop in operation. | C.1 Suspend all operations involving a reduction of RCS boron concentration.<br><br>AND<br><br>C.2 Initiate action to restore one loop to OPERABLE status and operation. | Immediately<br><br><br><br>Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.6.1 | Verify one RHR or RCS loop is in operation. | 12 hours |
| SR 3.4.6.2 | Verify SG secondary-side water levels ≥ ~~[17]~~ 10 % NARROW-RANGE for required OPERABLE A RCS loops. | 12 hours |

1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.6.3 | Verify correct breaker alignment and indicated power are available to the required pump that is not in operation. | 7 days |

# B 3.4 REACTOR COOLANT SYSTEM (RCS)

## B 3.4.6 RCS Loops – MODE 4

BASES

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BACKGROUND

In MODE 4, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat to either the steam generator (SG) secondary-side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

The reactor coolant is circulated through four RCS loops connected in parallel to the reactor vessel, each loop containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The RCPs circulate the coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and to prevent boric acid stratification.

In MODE 4, either RCPs or RHR loops can be used to provide forced circulation. The intent of this LCO is to provide forced flow from at least one RCP or one RHR loop for decay heat removal and transport. The flow provided by one RCP loop or RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that two paths be available to provide redundancy for decay heat removal.

---

APPLICABLE SAFETY ANALYSES

In MODE 4, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RCS and RHR loops provide this circulation.

RCS loops – MODE 4 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 two loops be OPERABLE in MODE 4 and that one of these loops be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS

(continued)

BASES

LCO (continued)

loops and RHR loops. Any one loop in operation provides enough flow to remove the decay heat from the core with forced circulation. An additional loop is required to be OPERABLE to provide redundancy for heat removal.

Note 1 permits all RCPs or RHR pumps to be de-energized for ≤ 1 hour per 8-hour period. The purpose of the Note is to permit tests that are designed to validate various accident analyses values. One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no-flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits the de-energizing of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The 1-hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

4

Utilization of Note 1 is permitted provided the following conditions are met along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

2

The note ~~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 an RCP with any RCS cold leg temperature ≤ ~~275~~ 310°F. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started. 1

An OPERABLE RCS loop is comprised of an OPERABLE RCP and an OPERABLE SG in accordance with the Steam Generator Tube

(continued)

BASES

LCO (continued)

Surveillance Program, which has the minimum water level specified in SR 3.4.6.2.

Similarly for the RHR System, an OPERABLE RHR loop is comprised of an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RCPs and RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

APPLICABILITY

In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of either RCS or RHR provides sufficient circulation for these purposes. However, two loops consisting of any combination of RCS and RHR loops are required to be OPERABLE to meet single failure considerations.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops – MODES 1 and 2";
LCO 3.4.5, "RCS Loops – MODE 3";
LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level"; and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level."

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

(continued)

BASES

ACTIONS

B.1 (continued)

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 unit must be brought to MODE 5 within 24 hours. Bringing the unit 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 ~~300~~ 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.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This SR requires verification every 12 hours that one RCS or RHR 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 RCS and RHR loop performance.

(continued)

BASES

1

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.6.2

10

SR 3.4.6.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary-side narrow-range water level is ≥ ~~[17]~~%. If the SG secondary-side narrow-range water level is < ~~[17]~~%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. The 12-hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

SR 3.4.6.3

Verification that the required pump is OPERABLE ensures that an additional RCS or RHR pump 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 pump. 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.

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 3.4.6

A. This change reflects comments made by the industry to the NRC at the RSTS P&R meeting held in Irvine, CA July 13 - July 20, 1992.

1. Change to reflect WBN specific parameter values.

2. Format change to delete brackets that identify plant specific information/values.

3. Change to correct errors in the STS.

4. Watts Bar will not be performing cold-no-flow rod drop tests. The rod drop test will be conducted at hot full flow conditions.

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.7 RCS Loops – MODE 5, Loops Filled

LCO 3.4.7 One residual heat removal (RHR) loop shall be OPERABLE and in operation, and either:

a. One additional RHR loop shall be OPERABLE; or

b. The secondary-side water level of at least [two] steam generators (SGs) shall be ≥ ~~[17]~~ 10 % NARROW RANGE.

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

~~1. The RHR pump of the loop in operation may be de-energized for ≤ 1 hour per 8-hour period provided:~~

~~a. No operations are permitted that would cause reduction of the RCS boron concentration; and~~

~~b. Core outlet temperature is maintained at least 10°F below saturation temperature.~~

1 ~~2~~. One required RHR loop may be inoperable for up to 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

2 ~~3~~. No reactor coolant pump shall be started with one or more RCS cold leg temperatures ≤ ~~[275]~~ 310 °F unless the secondary-side water temperature of each SG is ≤ [50] °F above each of the RCS cold leg temperatures.

3 ~~4~~. All RHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.

---

APPLICABILITY: MODE 5 with RCS loops filled.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One required RHR loop inoperable.<br><br>AND<br><br>Required SGs secondary-side water levels not within limits. | A.1 Initiate action to restore a second RHR loop to OPERABLE status<br><br>OR<br><br>A.2 Initiate action to restore required SG secondary-side water levels to within limits. | Immediately<br><br><br><br>Immediately |
| B. Required RHR loops inoperable.<br><br>OR<br><br>No RHR loop in operation. | B.1 Suspend all operations involving a reduction of RCS boron concentration.<br><br>AND<br><br>B.2 Initiate action to restore one RHR loop to OPERABLE status and operation. | Immediately<br><br><br><br>Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| SR 3.4.7.1 Verify one RHR loop is in operation. | 12 hours |

(continued)

SURVEILLANCE REQUIREMENTS (continued) A

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.7.2 | ----------NOTE---------- Only required to be met if fewer than two RHR loops are OPERABLE. ---------- Verify SG secondary-side water level is ≥ ~~[17]~~ 10 % in required SGs. NARROW RANGE 1 | 12 hours |
| SR 3.4.7.3 | A ----------NOTE---------- Only required to be met if secondary-side water level is < ~~[17]~~ 10 % in more than ~~[two]~~ 1 SGs. NARROW RANGE ---------- 2 Verify correct breaker alignment and indicated power are available to the required RHR pump that is not in operation. | 7 days |

# B 3.4 REACTOR COOLANT SYSTEM (RCS)

## B 3.4.7 RCS Loops – MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is to remove decay heat and transfer this heat to the steam generators (SGs) or 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.

In MODE 5 with RCS loops filled, the reactor coolant is circulated by means of two RHR loops connected to the 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 ~~[17]~~% 10 to provide an alternate method for decay heat removal.

NARROW RANGE

1

(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.

E*A

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 ≥ [17]%. 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 ≥ [17]%. Should the operating RHR loop fail, the SGs could be used to remove the decay heat.

Note 1 permits all RHR pumps to be de-energized ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests designed to validate various accident analyses values. One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits de-energizing of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not likely during this short period with no forced flow.

(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.

3 A

RCS loops – MODE 5 (loops filled) have been identified in the NRC Policy Statement as important contributors to risk reduction.

LCO

10 1 NARROW RANGE.

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 ≥ ~~[17]%~~. 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 ≥ ~~[17]%.~~ Should the operating RHR loop fail, the SGs could be used to remove the decay heat.

1 10% NARROW RANGE.

4

~~Note 1 permits all RHR pumps to be de-energized ≤ 1 hour per 8-hour period. The purpose of the Note is to permit tests designed to validate various accident analyses values. One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no-flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits de-energizing of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The 1-hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not likely during this short period with no forced flow.~~

(continued)

BASES

LCO (continued)

~~Utilization of Note 1 is permitted provided the following conditions are met, along with any other conditions imposed by initial startup test procedures:~~

~~a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and~~

4

~~b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.~~

Note 1 ~~2~~ 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 ~~3~~ requires 2 ~~that~~ the secondary-side water temperature of each SG be ≤ ~~[50]~~ 310 °F above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) with an RCS cold leg temperature ≤ ~~[275]~~ °F. This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

Note 3 ~~4~~ 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 function provided by the RHR loops.

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. An OPERABLE SG can perform as a heat sink when it has an adequate water level and is OPERABLE in accordance with the Steam Generator Tube Surveillance Program.

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from

(continued)

BASES

APPLICABILITY (continued)

the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary-side water level of at least [two] SGs is required to be ≥ ~~[17]%.~~ 10% NARROW RANGE.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops – MODES 1 and 2";
LCO 3.4.5, "RCS Loops – MODE 3";
LCO 3.4.6, "RCS Loops – MODE 4";
LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level"; and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level."

ACTIONS

A.1 and A.2

10% NARROW RANGE,

If one required RHR loop is inoperable and the required SGs have secondary-side water levels < ~~[17]%,~~ redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore the required SG secondary-side water levels. Either Required Action A.1 or Required Action A.2 will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no RHR loop is in operation, except during conditions permitted by Note 1, or if no loop is OPERABLE, all operations involving a reduction of RCS boron concentration must be suspended and action to restore one RHR loop to OPERABLE status and operation must be initiated. Avoidance of a Boron dilution requires forced circulation for proper mixing and to preserve the margin to criticality in this type of operation. The immediate Completion Times reflect the importance of maintaining operation for heat removal.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification every 12 hours that the required 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 3.4.7.2

1 RANGE

Verifying that at least two SGs are OPERABLE by ensuring their secondary-side narrow-range water levels are ≥ ~~[17]%~~ 10% NARROW ensures an alternate decay heat removal method in the event that the second RHR loop is not OPERABLE. The Note requires the Surveillance to be met when the LCO requirement is being met by use of the SGs. If both RHR loops are OPERABLE, this Surveillance is not needed. The 12-hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level. A

SR 3.4.7.3

Verification that a second RHR pump is OPERABLE ensures that an additional pump 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 RHR pump. The Note requires the Surveillance to be met when the LCO requirement is being met by the use of RHR loops. If secondary-side water level is ≥ ~~[17]%~~ in at least two SGs, this Surveillance is not needed. 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. A

10% NARROW RANGE 1

REFERENCES None.

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 3.4.7

A. This change reflects comments made by the industry to the NRC at the RSTS P&R meeting held in Irvine, CA July 13 - July 20, 1992.

1. Change to reflect WBN specific parameter values.

2. Format change to delete brackets that identify plant specific information/values.

3. Change to correct errors in the STS.

4. Watts Bar will not be performing cold-no-flow rod drop tests. The rod drop test will be conducted at hot full flow conditions.

# 3.4 REACTOR COOLANT SYSTEM (RCS)

## 3.4.8 RCS Loops – MODE 5, Loops Not Filled

LCO 3.4.8 Two residual heat removal (RHR) loops shall be OPERABLE and one RHR loop shall be in operation. (A)

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

1. All RHR pumps may be de-energized for ≤ 15 minutes (1 hour) ~~when switching from one loop to another~~ provided:

   (1) (3) a. ~~The maximum RCS~~ Core outlet temperature is ≤ ~~160°F;~~ maintained ~~at least~~ > 10°F below saturation temperature; (A)

   b. No operations are permitted that would cause a reduction of the RCS boron concentration; and

   c. No draining operations to further reduce the RCS water volume are permitted.

2. One RHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

---

APPLICABILITY: MODE 5 with RCS loops not filled.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One RHR loop inoperable. | A.1 Initiate action to restore RHR loop to OPERABLE status. | Immediately |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| B. Required RHR loops inoperable.<br><br>OR<br><br>No RHR loop in operation. | B.1 Suspend all operations involving reduction in RCS boron concentration.<br><br>AND<br><br>B.2 Initiate action to restore one RHR loop to OPERABLE status and operation. | Immediately<br><br><br><br><br>Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.8.1 | Verify one RHR loop is in operation. | 12 hours |
| SR 3.4.8.2 | Verify correct breaker alignment and indicated power are available to the required RHR pump that is not in operation. | 7 days |

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.8 RCS Loops – MODE 5, Loops Not Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops not filled, the primary function of the reactor coolant is the removal of decay heat generated in the fuel, and the transfer of this heat to the component cooling water via the residual heat removal (RHR) heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

In MODE 5 with loops not filled, only RHR pumps can be used for coolant circulation. The number of pumps in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR pump for decay heat removal and transport and to require that two paths be available to provide redundancy for heat removal.

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. The flow provided by one RHR loop is adequate for heat removal and for boron mixing.

RCS loops in MODE 5 (loops not 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 two RHR loops be OPERABLE and one of these loops be in operation. An OPERABLE loop is one that has the capability of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the RHR System unless forced flow is used. A minimum of one running RHR pump meets the LCO requirement for one loop in operation. An additional RHR loop is required to be OPERABLE to meet single failure considerations.

(continued)

BASES

1 hour (A) (1)

LCO (continued)

Note 1 permits all RHR pumps to be de-energized for ≤ 15 minutes when switching from one loop to another. The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short (and temperature is maintained ≤ [160]°F). The Note prohibits boron dilution or draining operations when RHR forced flow is stopped.

Core outlet (3)

(A) at least 10°F below saturation

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

An OPERABLE RHR loop is comprised of an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the RHR System.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops – MODES 1 and 2";
LCO 3.4.5, "RCS Loops – MODE 3";
LCO 3.4.6, "RCS Loops – MODE 4";
LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level"; and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level."

ACTIONS

A.1

If only one RHR loop is OPERABLE and in operation, redundancy for RHR is lost. Action must be initiated to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

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.

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.

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 3.4.8

A. This change reflects comments made by the industry to the NRC at the RSTS P&R meeting held in Irvine, CA July 13 - July 20, 1992.

1. Format change to delete brackets that identify plant specific information/values.

2. Change to correct errors in the STS.

3. Change to make consistent with LCO 3.4.7 and with STS Revision 4a which allowed monitoring at core outlet temperature with respect to saturation temperature.

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ [92]% of span, equivalent to ≤ ~~[1240]~~ 1656 cubic feet; and

b. Two groups of pressurizer heaters OPERABLE with the capacity of each group ≥ ~~[125]~~ 150 kW ~~[and capable of being powered from an emergency power supply]~~.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. Pressurizer water level not within limit. | A.1 | Be in MODE 3 with reactor trip breakers open. | 6 hours |
| | AND | | |
| | A.2 | Be in MODE 4. | 12 hours |
| B. One required group of pressurizer heaters inoperable. | B.1 | Restore required group of pressurizer heaters to OPERABLE status. | 72 hours |
| C. Required Action and associated Completion Time of Condition B not met. | C.1 | Be in MODE 3. | 6 hours |
| | AND | | |
| | C.2 | Be in MODE 4. | 12 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.9.1 | Verify pressurizer water level ≤ [92]% of ~~span.~~ indicated level (A) (2) | 12 hours |
| SR 3.4.9.2 | Verify capacity of each required group of pressurizer heaters ≥ ~~[125]~~ 150 kW. (1) | 92 days |
| SR 3.4.9.3 | Verify required pressurizer heaters are capable of being powered from an emergency power supply. | [18] months (4) |

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady-state operation, and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The pressure control components addressed by this LCO include the pressurizer water level, the required heaters, and their controls and emergency power supplies. Pressurizer safety valves and pressurizer power-operated relief valves are addressed by LCO 3.4.10, "Pressurizer Safety Valves," and LCO 3.4.11, "Pressurizer Power-Operated Relief Valves (PORVs)," respectively.

The intent of the LCO is to ensure that a steam bubble exists in the pressurizer prior to power operation to minimize the consequences of potential overpressure transients. The presence of a steam bubble is consistent with analytical assumptions. Relatively small amounts of non-condensable gases can inhibit the condensation heat transfer between the pressurizer spray and the steam, and diminish the spray effectiveness for pressure control.

Electrical immersion heaters, located in the lower section of the pressurizer vessel, keep the water in the pressurizer at saturation temperature and maintain a constant operating pressure. A minimum required available capacity of pressurizer heaters ensures that the RCS pressure can be maintained. The capability to maintain and control system pressure is important for maintaining subcooled conditions in the RCS and ensuring the capability to remove core decay heat by either forced or natural circulation of reactor coolant. Unless adequate heater capacity is available, the hot, high-pressure condition cannot be maintained indefinitely and still provide the required subcooling margin in the primary system. Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to

(continued)

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 non-condensible 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. (5)

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 (1) for the pressurizer to be OPERABLE with a water volume ≤ ~~[1240]~~ 1656 cubic feet, which is equivalent to ~~[92]~~% of span (indicated level) (2), 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 (1) of OPERABLE pressurizer heaters, each with a capacity ≥ ~~[125]~~ 150 kW, ~~capable of being powered from either the offsite power source or the emergency power supply.~~ (4) 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 near the operating

(continued)

BASES

LCO (continued)

conditions, a wide margin to subcooling can be obtained in the loops. The exact design value of ~~125~~ 150 kW is derived from the use of ~~seven~~ twelve heaters rated at ~~17.9~~ 12.5 kW each. The amount needed to maintain pressure is dependent on the heat losses.

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup.

In MODES 1, 2, and 3, there is need to maintain the availability of pressurizer heaters, capable of being powered from an emergency power supply. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Residual Heat Removal (RHR) System is in service, and therefore, the LCO is not applicable.

ACTIONS

A.1 and A.2

Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the nominal upper limit, even with the plant at steady-state conditions. Normally the plant will trip in this event since the upper limit of this LCO is the same as the Pressurizer Water Level – High Trip.

If the pressurizer water level is not within the limit, action must be taken to restore the plant to operation within the bounds of the safety analyses. To achieve this status, the unit must be brought to MODE 3, with the reactor trip breakers open, within 6 hours and to MODE 4 within 12 hours. This takes the unit out of the applicable MODES

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

and restores the unit to operation within the bounds of the safety analyses.

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.

B.1

If one required group of pressurizer heaters is inoperable, restoration is required within 72 hours. The Completion Time of 72 hours is reasonable considering the anticipation that a demand caused by loss of offsite power would be unlikely in this period. Pressure control may be maintained during this time using normal station-powered heaters.

C.1 and C.2

If one group of pressurizer heaters are inoperable and cannot be restored in the allowed Completion Time of Required Action B.1, 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 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 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that during steady-state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. The Frequency of 12 hours corresponds to verifying the parameter each shift. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess level for any deviation and verify that operation is within

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1 (continued)

safety analyses assumptions. Alarms are also available for early detection of abnormal level indications.

SR 3.4.9.2

The SR is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated pressurizer heaters are verified to be at their design rating. This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance. The Frequency of 92 days is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.

4

~~SR 3.4.9.3~~

~~This SR is not applicable if the heaters are permanently powered by Class 1E power supplies.~~

~~This Surveillance demonstrates that the heaters can be manually transferred from the normal to the emergency power supply and energized. The Frequency of 18 months is based on a typical fuel cycle and is consistent with similar verifications of emergency power supplies.~~

REFERENCES

~~1. FSAR, Section [ ].~~

~~2. NUREG-0737, November 1980.~~

3

1. Watts Bar FSAR, Chapter 15, "Accident Analyses."

2. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 3.4.9

A. This change reflects comments made by the industry to the NRC at the RSTS P&R meeting held in Irvine, CA July 13 - July 20, 1992.

1. Change to reflect WBN specific parameter values or information.

2. Format change to delete brackets that identify plant specific information/values.

3. Change to specify WBN specific reference information and format.

4. Change to delete text in the STS since required pressurizer heaters at WBN are normally powered from a Class 1E source.

5. Editorial change to correct punctuation.

## 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 ~~any~~ all RCS cold leg temperature > ~~[275]~~ 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.<br><br>OR<br><br>Two or more pressurizer safety valves inoperable. | B.1 Be in MODE 3.<br><br>AND<br><br>B.2 Be in MODE 4 with ~~all~~ at least one RCS cold leg temperature~~s~~ ≤ ~~[275]~~ 310 °F. | 6 hours<br><br><br>12 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.10.1 | Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. | In accordance with the Inservice Testing Program |

# B 3.4 REACTOR COOLANT SYSTEM (RCS)

## B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. The pressurizer safety valves are totally enclosed pop-type, spring-loaded, self-actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), [2735 psig], which is 110% of the design pressure. (2)

Because the safety valves are totally enclosed and self-actuating, they are considered independent components. The relief capacity for each valve, ~~300,000~~ 420,000 lb/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves or increase in the pressurizer relief tank temperature or level. (1)

Overpressure protection is required in MODES 1, 2, 3, 4, and 5; however, in MODE 4, with one or more RCS cold leg temperatures ≤ ~~[275]~~ 310 °F, and MODE 5 and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, ~~"Low Temperature Overpressure Protection (LTOP) System."~~ "Cold Overpressure Mitigation System (COMS)." (1)

The upper and lower pressure limits are based on the ± 1%-tolerance requirement (Ref. 1) for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that

(continued)

BASES

BACKGROUND (continued)

the RCS pressure will be limited to 110% of design pressure. The consequences of exceeding the American Society of Mechanical Engineers (ASME) pressure limit (Ref. 1) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

APPLICABLE SAFETY ANALYSES

All accident and safety analyses in the FSAR (Ref. 2) that require safety valve actuation assume operation of three pressurizer safety valves to limit increases in RCS pressure. The overpressure protection analysis (Ref. 3) is also based on operation of [three] safety valves. Accidents that could result in overpressurization if not properly terminated include:

a. Uncontrolled rod withdrawal from full power;

b. Loss of reactor coolant flow;

c. Loss of external electrical load;

d. Loss of normal feedwater;

e. Loss of all AC power to station auxiliaries; and

f. Locked Rotor.

Detailed analyses of the above transients are contained in Reference 2. Safety valve actuation is required in events c, d, and e (above) to limit the pressure increase. Compliance with this LCO is consistent with the design bases and accident analyses assumptions.

Pressurizer safety valves satisfy Criterion 3 of the NRC Policy Statement.

LCO

The [three] pressurizer safety valves are set to open at the RCS design pressure (2500 psig), and within the ASME specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits are based on the ± 1%-tolerance

(continued)

BASES

LCO (continued)

requirements (Ref. 1) for lifting pressures above 1000 psig. The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

In MODES 1, 2, and 3, and portions of MODE 4 above the ~~LTOP~~ COMS arming temperature, OPERABILITY of [three] valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 and portions of MODE 4 are conservatively included, although the listed accidents may not require the safety valves for protection.

The LCO is not applicable in MODE 4 at or below ~~[275]~~ 310 °F or in MODE 5 because ~~LTOP~~ COMS is provided. Overpressure protection is not required in MODE 6 with reactor vessel head detensioned.

The Note allows entry into MODES 3 and 4 with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The [54]-hour exception is based on 18-hour outage time for each of the [three] valves. The 18-hour period is derived from operating experience that hot testing can be performed in this timeframe.

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS Overpressure Protection System. An inoperable safety valve

(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

310 (1)

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 at least one cold leg ≤ ~~[275]~~ °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. (4) At or below ~~[275]~~ °F, 310 (1) overpressure protection is provided by the ~~LTOP System.~~ COMS 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. (2)

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested one at a time and (1) 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.

REFERENCES

See INSERT from Page B3.4-48A (3)

~~1. ASME, *Boiler and Pressure Vessel Code*, Section III.~~
~~2. FSAR, Chapter [15].~~
~~3. WCAP-7769, Rev. 1, June 1972.~~
~~4. ASME, *Boiler and Pressure Vessel Code*, Section XI.~~

(3)

INSERT

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

2. Watts Bar FSAR, ~~Section~~ Chapter 15, "Safety Analyses."

3. WCAP-7769, Rev. 1, "Topical Report on Overpressure Protection for Westinghouse Pressurized Water Reactors," June 1972.

4. Thomas E. Murley (NRC) letter to W. S. Wilgus dated May 9, 1988, forwarding the NRC Staff review of the "Nuclear Steam Supply System Vendor Owners Groups' Application of the Commission's Interim Policy Statement Criteria to the Standard Technical Specifications."

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

4~~6~~. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 3.4.10

A. This change reflects comments made by the industry to the NRC at the RSTS P&R meeting held in Irvine, CA July 13 - July 20, 1992.

1. Change to reflect WBN specific parameter values or information.

2. Format change to delete brackets that identify plant specific information/values.

3. Change to specify WBN specific reference information and format.

4. Change to correct error in STS.

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.11 Pressurizer Power-Operated Relief Valves (PORVs)

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

A

---NOTE---
2. LCO 3.0.4 is not applicable.
---

ACTIONS

---NOTE---
1. Separate Condition entry is allowed for each PORV.
---

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. One or more PORVs inoperable and capable of being manually cycled. | A.1 | Close and maintain power to associated block valve. | 1 hour |
| 1 B. One ~~[or two]~~ PORV~~[s]~~ inoperable and not capable of being manually cycled. | B.1 | Close associated block valve~~[s]~~. | 1 hour |
| | AND | | |
| | B.2 | Remove power from associated block valve~~[s]~~. | 1 hour |
| | AND | | |
| | B.3 | Restore PORV~~[s]~~ to OPERABLE status. | 72 hours |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| C. One block valve inoperable. | C.1 | Place associated PORV in manual control. | 1 hour |
| | AND | | |
| | C.2 | Restore block valve to OPERABLE status. | 72 hours |
| D. Required Action and associated Completion Time of Condition A, B, or C not met. | D.1 | Be in MODE 3. | 6 hours |
| | AND | | |
| | D.2 | Be in MODE 4. | 12 hours |
| (1) E. Two ~~[or three]~~ PORVs inoperable and not capable of being manually cycled. | E.1 | Close associated block valves. | 1 hour |
| | AND | | |
| | E.2 | Remove power from associated block valves. | 1 hour |
| | AND | | |
| | E.3 | Be in MODE 3. | 6 hours |
| | AND | | |
| | E.4 | Be in MODE 4. | 12 hours |
| (1) F. Two ~~More than one~~ block valve(s) inoperable. | F.1 | Place associated PORVs in manual control. | 1 hour |
| | AND | | |

(continued)

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| F. (continued) | F.2 Restore one block valve to OPERABLE status. ~~[if three block valves are inoperable].~~ (1) | 2 hours |
| | AND | |
| | F.3 Restore remaining block valve(s) to OPERABLE status. | 72 hours |
| G. Required Action~~s~~ and associated Completion Time~~s~~ of Condition F not met. (A) | G.1 Be in MODE 3. | 6 hours |
| | AND | |
| | G.2 Be in MODE 4. | 12 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.11.1 | ----------NOTE---------- Not required to be met with block valve closed in accordance with the Required Action of Condition B or E. ---------- | |
| | Perform a complete cycle of each block valve. | 92 days |
| (A) SR 3.4.11.2 | Perform a CHANNEL CALIBRATION for each PORV. | [18] months (2) |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.11.~~6~~ 2 (A) | ----------NOTE---------- Only required to be performed in MODE 3. ---------- Perform a complete cycle of each PORV. | (2) [18] months |
| (4) SR 3.4.11.~~4~~ 3 | Perform a complete cycle of each solenoid air control valve and check valve on the air accumulators in PORV control systems. | [18] months |
| SR 3.4.11.5 | Verify PORVs and block valves are capable of being powered from emergency power sources. | [18] months |

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Pressurizer Power-Operated Relief Valves (PORVs)

BASES

(1)

BACKGROUND

The pressurizer is equipped with two types of devices for pressure relief: pilot-operated solenoid pressurizer safety valves and PORVs. The PORVs are ~~air-operated~~ valves that are controlled to open at a specific set pressure when the pressurizer pressure increases and closes when the pressurizer pressure decreases. The PORVs may also be manually operated from the control room.

Block valves, which are normally open, are located between the pressurizer and the PORVs. The block valves are used to isolate the PORVs in case of excessive leakage or a stuck-open PORV. Block valve closure is accomplished manually using controls in the control room. A stuck-open PORV is, in effect, a small-break loss-of-coolant accident (LOCA). As such, block valve closure terminates the RCS depressurization and coolant inventory loss.

The PORVs and their associated block valves may be used by plant operators to depressurize the RCS to recover from certain transients if normal pressurizer spray is not available. Additionally, the series arrangement of the PORVs and their block valves permit performance of surveillances on the valves during power operation.

The PORVs may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.

The PORVs, their block valves, and their controls are powered from the vital buses that normally receive power from offsite power sources, but are also capable of being powered from emergency power sources in the event of a loss of offsite power. Two PORVs and their associated block valves are powered from two separate safety trains (Ref. 1).

(1)

The plant has two PORVs, each having a relief capacity of 210,000 lb/hr at ~~2335~~ 2324 psig. The functional design of the PORVs is based on maintaining pressure below the Pressurizer Pressure—High reactor trip setpoint following a step reduction of 50% of full load with steam dump. In addition,

(continued)

BASES

BACKGROUND (continued)

the PORVs minimize challenges to the pressurizer safety valves and also may be used for low temperature overpressure protection (LTOP). See LCO 3.4.12, "~~Low Temperature~~ Cold Overpressure ~~(LTOP) Protection~~ Mitigation System." (COMS) (1)

APPLICABLE SAFETY ANALYSES

Plant operators employ the PORVs to depressurize the RCS in response to certain plant transients if normal pressurizer spray is not available. For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes that manual operator actions are required to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray is unavailable to reduce RCS pressure. The PORVs are assumed to be used for RCS depressurization, which is one of the steps performed to equalize the primary and secondary pressures in order to terminate the primary to secondary break flow and the radioactive releases from the affected steam generator.

The PORVs are used in safety analyses for events that result in increasing RCS pressure for which departure from nucleate boiling ratio (DNBR) criteria are critical. By assuming PORV manual actuation, the primary pressure remains below the high pressurizer pressure trip setpoint; thus, the DNBR calculation is more conservative. Events that assume this condition include a turbine trip and the loss of normal feedwater (Ref. 2). (5)

Pressurizer PORVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation to mitigate the effects associated with a SGTR.

By maintaining two PORVs and their associated block valves OPERABLE, the single-failure criterion is satisfied. The block valves are available to isolate the flow path through either a failed open PORV or a PORV with excessive leakage. Satisfying the LCO helps minimize challenges to fission-product barriers.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, the PORV and its block valve are required to be OPERABLE to limit the potential for a small-break LOCA through the flow path. The most likely cause for a PORV small-break LOCA is a result of a pressure-increase transient that causes the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2. The PORVs are also required to be OPERABLE in MODES 1, 2, and 3 to minimize challenges to the pressurizer safety valves.

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the LCO is applicable in MODES 1, 2, and 3. The LCO is not applicable in MODE 4 when both pressure and core energy are decreased and the pressure surges become much less significant. The PORV setpoint is reduced for LTOP in MODES 4, 5, and 6 with the reactor vessel head in place. LCO 3.4.12 addresses the PORV requirements in these MODES.

A

The exception for LCO 3.0.4, Note 2, permits entry into MODES 1, 2, and 3 to perform cycling of the PORVs or block valves to verify their OPERABLE status. Testing is not performed in lower MODES.

ACTIONS

A Note 1 has been added to clarify that all pressurizer PORVs are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis).

A.1

With the PORVs inoperable and capable of being manually cycled, either the PORVs must be restored or the flow path isolated within 1 hour. The block valves should be closed but power must be maintained to the associated block valves, since removal of power would render the block valve inoperable. Although a PORV may be designated inoperable, it may be able to be manually opened and closed, and therefore, able to perform its function. PORV inoperability may be due to seat leakage, instrumentation problems,

(continued)

BASES

ACTIONS

A.1 (continued)

automatic control problems, or other causes that do not prevent manual use and do not create a possibility for a 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 seat leakage 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).

5

problematic

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

1

If one ~~[on two]~~ PORV~~[s]~~ 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

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

close the block valve is to isolate a stuck-open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck-open PORV at a time that the block valve is inoperable. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period and provides the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted a Completion Time of 72 hours to restore the inoperable block valve to OPERABLE status. The time allowed to restore the block valve is based upon the Completion Time for restoring an inoperable PORV in Condition B, since the PORVs are not capable of mitigating an overpressure event when placed in manual control. If the block valve is restored within the Completion Time of 72 hours, the power will be restored and the PORV restored to OPERABLE status. If it 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.

D.1 and D.2

If the Required Action of Condition A, B, or C is not met, then 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 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. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.12, "~~Low Temperature~~ Cold Overpressure ~~Protection~~ Mitigation ~~(LTOP)~~ System." (COMS) (1)

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

If more than one PORV is inoperable and not capable of being manually cycled, it is necessary to either restore at least one valve within the Completion Time of 1 hour or isolate

(continued)

BASES

ACTIONS

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

the flow path by closing and removing the power to the associated block valves. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time and provides the operator time to correct the situation. If one PORV is restored and one PORV remains inoperable, then the plant will be in Condition B with the time clock started at the original declaration of having two [or three] PORVs inoperable. If no PORVs are restored within the Completion Time, then 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 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. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.12.

F.1 AND F.2

~~F.1, F.2, and F.3~~

If ~~more than one~~ both block valves ~~is~~ are inoperable, it is necessary to either restore the block valves within the Completion Time of 1 hour, or place the associated PORVs in manual control and restore at least one block valve within 2 hours ~~[and restore the remaining block valve within 72 hours]~~. The Completion Times are reasonable, based on the small potential for challenges to the system during this time and provide the operator time to correct the situation.

G.1 and G.2

If the Required Actions of Condition F are not met, then 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 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. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.12.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be closed if needed. The basis for the Frequency of 92 days is the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. 3). If the block valve is closed to isolate a PORV that is capable of being manually cycled, the OPERABILITY of the block valve is of importance, because opening the block valve is necessary to permit the PORV to be used for manual control of reactor pressure. If the block valve is closed to isolate an otherwise inoperable PORV, the maximum Completion Time to restore the PORV and open the block valve is 72 hours, which is well within the allowable limits (25%) to extend the block valve Frequency of 92 days. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to OPERABLE status (i.e., completion of the Required Actions fulfills the SR).

The Note modifies this SR by stating that it is not required to be met with the block valve closed, in accordance with the Required Action of this LCO.

A

SR 3.4.11.2

SR 3.4.11.2 requires the performance of a CHANNEL CALIBRATION for each PORV every [18] months. The CHANNEL CALIBRATION ensures that the PORV setpoint is appropriately maintained below the RCS high pressure trip setpoint, and thus, remote from transient pressure challenges. The calibration also ensures that the PORV setpoint is below the pressurizer safety valve setpoint, thus limiting challenges to the safety valves. The calibration can only be performed during shutdown. The Frequency of [18] months is based on a typical refueling cycle and industry-accepted practice.

SR 3.4.11.~~3~~2

SR 3.4.11.~~3~~2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. The Frequency of [18] months is based on a typical refueling cycle and industry-accepted practice. The Note allows

(continued)

BASES

A

SURVEILLANCE REQUIREMENTS

SR 3.4.11.3 (continued)

entering MODE 3 prior to meeting the requirements of SR 3.4.11.3.

SR 3.4.11.4

Operating the solenoid air control valves and check valves on the air accumulators ensures the PORV control system actuates properly when called upon. The Frequency of [18] months is based on a typical refueling cycle and the Frequency of the other Surveillances used to demonstrate PORV OPERABILITY.

4

SR 3.4.11.5

This Surveillance is not required for plants with permanent 1E power supplies to the valves.

The Surveillance demonstrates that emergency power can be provided and is performed by transferring power from normal to emergency supply and cycling the valves. The Frequency of [18] months is based on a typical refueling cycle and industry-accepted practice.

REFERENCES

1. ~~Regulatory Guide 1.32, U.S. Nuclear Regulatory Commission, February 1977.~~
2. ~~FSAR, Section [15.2].~~
3. ~~ASME, *Boiler and Pressure Vessel Code*, Section XI.~~

3

1. Regulatory Guide 1.32, "Criteria for Safety Related Electric Power Systems for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, February 1977.
2. Watts Bar FSAR, Section 15.2, "Condition II - Faults of Moderate Frequency".
3. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 3.4.11

A. This change reflects comments made by the industry to the NRC at the RSTS P&R meeting held in Irvine, CA July 13 - July 20, 1992.

1. Change to reflect WBN specific parameter values or information.

2. Format change to delete brackets that identify plant specific information/values.

3. Change to specify WBN specific reference information and format.

4. These surveillance are deleted since:

   a. WBN PORVs are not air-operated and do not utilize air accumulators; and

   b. WBN PORVs are normally powered from a Class 1E source.

5. Change to correct error in STS.

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.12 ~~Low Temperature~~ Cold Overpressure ~~Protection (LTOP)~~ Mitigation System (COMS)

LCO 3.4.12 ~~An LTOP System~~ A COMS (1) shall be OPERABLE with a maximum of ~~[one] [high pressure injection (HPI)] pump [and~~ (5) one centrifugal charging pump~~]~~ (2) 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. Two residual heat removal (RHR) suction relief valves with setpoints ≥ [436.5] psig and ≤ [463.5] psig, or]~~ (6)

~~[3.~~ 2. One PORV with a lift setting within the limits specified in the PTLR and ~~one~~ the (1) RHR suction relief valve with a setpoint ≥ ~~[~~436.5~~]~~ psig and ≤ ~~[~~463.5~~]~~ psig~~]~~. (2)

b. The RCS depressurized and an RCS vent ~~of ≥ [2.07] square inches~~ capable of relieving at least 475 GPM water flow. (1)

APPLICABILITY: MODE 4 when any RCS cold leg temperature is ≤ ~~[275]~~ 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.
---

NOTE

Two centrifugal charging pumps may be capable of injecting into the RCS during pump swap operations

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Two or more [HPI] pumps capable of injecting into the RCS. | A.1 Initiate action to verify a maximum of [one] [HPI] pump is capable of injecting into the RCS. | Immediately |
| ~~B.~~ A. Two or more charging pumps capable of injecting into the RCS. | ~~B~~ A.1 Initiate action to verify a maximum of ~~[one]~~ 1 centrifugal 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 ~~[C]~~ B not met. | ~~D~~ C.1 Increase RCS cold leg temperature above ~~[275]~~ 310 °F.<br>OR<br>~~D~~ C.2 Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR. | 12 hours<br><br>12 hours |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| D ~~E~~. One required RCS relief valve inoperable in MODE 4. | D ~~E~~.1 Restore required RCS relief valve~~s~~ to OPERABLE status. (3) | 7 days |
| E ~~F~~. One required RCS relief valve inoperable in MODES 5 or 6. | E ~~F~~.1 Restore required RCS relief valve to OPERABLE status. | 24 hours |
| F ~~G~~. Two required RCS relief valves inoperable.<br><br>OR<br><br>Required Action and associated Completion Time of Condition A, C ~~B~~, D, E, or ~~F~~ not met.<br><br>OR<br>(1) COMS ~~LTOP System~~ inoperable for any reason other than Condition A, B, C, D, E, or ~~F~~. | F ~~G~~.1 Depressurize RCS and establish RCS vent. ~~of ≥ [2.07] square inches.~~ ~~3~~ (1) | 8 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.12.1 | Verify a maximum of [one] [HPI] pump is capable of injecting into the RCS. | Within 15 minutes before decreasing RCS cold leg temperature to ≤ ~~[275]~~ 310 °F<br>AND<br>12 hours |
| SR 3.4.12.~~2~~ 1 | Verify a maximum of one centrifugal charging pump is capable of injecting into the RCS. | ~~Within 15 minutes before decreasing RCS cold leg temperature to ≤ ~~[275]~~ 310 °F AND 12 hours~~ |
| SR 3.4.12.~~3~~ 2 | Verify each accumulator is isolated. | ~~Within 15 minutes before decreasing RCS cold leg temperature to ≤ ~~[275]~~ 310 °F AND 12 hours~~ |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| (6) SR 3.4.12.4 | Verify RHR suction valve is open for each required RHR suction relief valve. | 12 hours |
| SR 3.4.12.~~5~~ 3 | ----------NOTE---------- Only required to be performed when complying with LCO 3.4.12b. ---------- Verify RCS vent ~~≤ [2.07] square inches~~ 3 open: (A) a. For unlocked-open vent ~~valve~~ path(s). (1) b. For locked-open vent ~~valve~~ path(s). (open line or) (stc) | 12 hours for unlocked open vent path(s) (A) (1) AND 31 days for locked open vent path(s) ~~12 hours~~ ~~31 days~~ (A) |
| SR 3.4.12.~~6~~ 4 | Verify PORV block valve is open for each required PORV. | 72 hours |
| (2) SR 3.4.12.~~7~~ 5 (6) | Verify ~~associated~~ both RHR suction isolation valves ~~is~~ are locked open with operator power removed for ~~each~~ the required RHR suction relief valve. | 31 days (2) |
| SR 3.4.12.~~8~~ 6 | ----------NOTE---------- Not required to be met until 12 hours after decreasing RCS cold leg temperature to ≤ ~~[275]~~ 310 °F. (1) ---------- Perform ACOT on each required PORV, excluding actuation. | 31 days |
| SR 3.4.12.~~9~~ 7 | Perform CHANNEL CALIBRATION for each required PORV actuation channel. | ~~[18]~~ 18 months (2) |

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 ~~Low Temperature~~ Cold Overpressure Mitigation ~~Protection (LTOP)~~ System (COMS) (1)

BASES

BACKGROUND (1)

The ~~LTOP System~~ 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 ~~LTOP~~ COMS MODES. (1)

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. (7)

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only ~~during~~ 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] [high pressure injection (HPI)] pump [and~~ one CENTRIFUGAL charging pump (5) and isolating the accumulators. (2) 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.

(continued)

BASES

1 COMS

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 ~~LTOP~~ 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 ~~[HPSI]~~ 5 centrifugal charging pump is actuated by SI.

1 COMS

The ~~LTOP System~~ for pressure relief consists of two PORVs with reduced lift settings, ~~or two residual heat removal (RHR) suction relief valves,~~ or one PORV and ~~one~~ the residual heat removal 6 (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

1 COMS

As designed for the ~~LTOP System~~ COMS, each PORV is signaled to open if the RCS pressure approaches a limit determined by the ~~LTOP~~ COMS actuation logic. The ~~LTOP~~ 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 ~~LTOP~~ COMS 1. 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.

2 RHR Suction Relief Valve Requirements

header 6

During COMS ~~LTOP~~ 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~~s~~ to the inlet~~s~~ of the RHR pumps. While these valves are open ~~and the RHR suction valves are open~~, 6 the RHR suction relief valve~~s~~ ~~are~~ exposed to the RCS and ~~are~~ is able to relieve pressure transients in the RCS. 2

1

2 The RHR suction isolation valves ~~and the RHR suction valves~~ must be open to make the RHR suction relief valve~~s~~ OPERABLE 6 for RCS overpressure mitigation. Auto-closure interlocks are not permitted to cause the RHR suction isolation valves to close. The RHR suction relief valve~~s~~ ~~are~~ is A spring-loaded, bellows-type water relief valve~~s~~ 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. 2

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 1 COMS ~~LTOP~~ 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.

1

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 ~~similarly establishing a vent by opening an RCS vent valve~~. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

Opening the pressurizer manway

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 4) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, and 3, and in MODE 4 with RCS cold leg temperature exceeding ~~[275]~~ 310 °F, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. At about ~~[275]~~ 310 °F and below, overpressure prevention falls to two OPERABLE RCS relief valves or to a depressurized RCS and a sufficient-sized RCS vent. Each of these means has a limited overpressure relief capability.

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the PTLR curves are revised, the ~~LTOP System~~ COMS (1) must be re-evaluated to ensure its functional requirements can still be met using the RCS relief valve method or the depressurized and vented RCS condition.

The PTLR contains the acceptance limits that define the ~~LTOP~~ COMS (1) requirements. Any change to the RCS must be evaluated against the Reference 4 analyses to determine the impact of the change on the ~~LTOP~~ COMS (1) acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

Mass Input Type Transients

a. Inadvertent safety injection; or

b. Charging/letdown flow mismatch.

Heat Input Type Transients

a. Inadvertent actuation of pressurizer heaters;

b. Loss of RHR cooling; or

c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued) (3)

Heat Input Type Transients (continued) (3)

The following are required during the ~~LTOP~~ COMS (1) MODES to ensure that mass and heat input transients do not occur, which either of the ~~LTOP~~ COMS (1) overpressure protection means cannot handle:

a. Deactivating all but ~~[one] [HPI] pump [and~~ one centrifugal charging pump~~]~~ OPERABLE; (5) (2)

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.

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 only ~~one [HPI] pump [and~~ one centrifugal charging pump ~~are]~~ is ~~[are]~~ actuated by SI. Thus, the LCO allows only ~~[one] [HPI] pump [and~~ one centrifugal charging pump~~]~~ (5) (2) OPERABLE during the ~~LTOP~~ COMS (1) MODES. Since neither one RCS relief valve nor the RCS vent can handle a full SI actuation, the LCO also requires the accumulators isolated.

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 (~~[275]~~ 310 (1) °F and below). (2) Fracture mechanics analyses established the temperature of ~~LTOP~~ COMS (1) Applicability at ~~[275]~~ 310 (1) °F.

The consequences of a COMS (3) small-break loss-of-coolant accident (LOCA) in ~~LTOP~~ 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] [HPI] pump [and~~ one centrifugal charging pump~~]~~ OPERABLE and SI actuation enabled. (5) (2)

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Heat Input Type Transients (continued)

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

a. Deactivating all but ~~[one] [HPI] pump [and~~ one centrifugal charging pump~~]~~ OPERABLE;

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.

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 only ~~one [HPI] pump [and~~ one centrifugal charging pump ~~are]~~ is ~~[are]~~ actuated by SI. Thus, the LCO allows only ~~[one] [HPI] pump [and~~ one centrifugal charging pump~~]~~ OPERABLE during the ~~LTOP~~ COMS MODES. Since neither one RCS relief valve nor the RCS vent can handle a full SI actuation, the LCO also requires the accumulators isolated.

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 (~~[275]~~ 310°F and below). Fracture mechanics analyses established the temperature of COMS ~~LTOP~~ Applicability at ~~[275]~~ 310°F.

The consequences of a small-break loss-of-coolant accident (LOCA) in ~~LTOP~~ 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] [HPI] pump [and~~ one centrifugal 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 ~~LTOP System~~, assuming the limiting ~~LTOP~~ transient of SI actuation of ~~[one [HPSI pump and~~ 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.

The PORV setpoints in the PTLR will be updated when the revised P/T limits conflict with the ~~LTOP~~ 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~~s~~ does not have variable pressure and temperature lift setpoints like the PORVs. Analyses must show that ~~one~~ 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 ~~LTOP~~ transient while maintaining RCS pressure less than the P/T limit curve. Assuming all relief flow requirements during the limiting ~~LTOP~~ event, ~~an~~ the RHR suction relief valve will maintain RCS pressure to within the valve rated lift setpoint, plus an accumulation no greater than 10% of the rated lift setpoint.~~]~~ [3%]

~~[Although each~~ The RHR suction relief valve ~~may itself meet single failure criteria, its~~ 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 valve~~s~~ must be analyzed to still accommodate the design basis transients for ~~LTOP.]~~ COMS.

~~[~~The RHR suction relief valve~~s~~ ~~are~~ is considered AN active component~~s~~. Thus, the failure of ~~one~~ this valve is assumed to represent the worst-case single active failure.~~]~~

RCS Vent Performance

With the RCS depressurized, analyses show a vent ~~size of~~ ~~3.0 2.07 square inches~~ capable of relieving at least 475 GPM water flow is capable of mitigating the allowed ~~LTOP~~ COMS overpressure transient. The capacity of ~~a vent this size~~ is 475 GPM greater than the flow of the limiting transient for the ~~LTOP~~ COMS configuration, ~~[~~SI actuation with ~~[one] HPI pump [and~~ one centrifugal charging pump~~]~~ OPERABLE, maintaining RCS pressure less than the maximum pressure on the P/T limit curve.

INSERT "A" (attached)

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 ~~LTOP System~~ COMS satisfies Criterion 2 of the NRC Policy Statement.

LCO

This LCO requires that the ~~LTOP System~~ COMS is OPERABLE. The ~~LTOP System~~ 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.

To limit the coolant input capability, the LCO requires ~~[one] [HPI] pump [and~~ one centrifugal charging pump~~]~~ capable of

(continued)

INSERT A

~~A survey has been performed of actual~~ Two vent flow paths have been identified in the Reactor Coolant System which could be opened up and realistically serve as pressure release (vent) paths. Two vent paths were identified. With one PORV removed, the open vent line could serve as one (vent) path. The pressurizer manway could serve as an alternative vent path with the manway cover removed. ~~The pressure drops in both flow paths were calculated.~~ ~~It was established that~~ Both flow paths are capable of discharging 475 gpm at low pressure in the Reactor Coolant System. Thus, either one of the two openings can be used for relieving the pressure to prevent violating the P/T limits.

BASES

LCO (continued)

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 (1) COMS ~~LTOP~~ MODE 4 small-break LOCA.

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 ~~LTOP~~ (1) 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 two valves and their control circuits.

(6) ~~[2. Two OPERABLE RHR suction relief valves; or]~~

(6) ~~[An~~ The RHR both suction relief valve is OPERABLE for COMS ~~LTOP~~ (1) when ~~its~~ RHR suction isolation valves ~~and its RHR suction valve~~ 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.~~]~~ (2)

2 ~~[3.~~ One OPERABLE PORV and ~~one~~ the OPERABLE RHR suction relief valve; or~~]~~ (2)

b. A depressurized RCS and an RCS vent.

An RCS vent is OPERABLE when ~~open with an area of at least [2.07] square inches.~~ capable of relieving atleast 475 GPM water flow. (3) (1) (1)

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

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is at or below ~~[275]~~ 310 (1) °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

(continued)

BASES

(1)

APPLICABILITY (continued)

Reference 1 P/T limits above ~~[275]~~ 310°F. When the reactor vessel head is off, overpressurization cannot occur.

LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," 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 ~~[275]~~ 310°F. (1)

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 performed only under these pressure and temperature conditions.

ACTIONS (A)

Required Action B.1 is modified by a Note that permits two pumps capable of RCS injection for a brief period to allow for Pump swaps (1)

INSERT (next page)

A.1 [and B.1] (5)

With two or more ~~HPI~~ charging 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.

~~A~~B.1, ~~B~~C.1, and ~~C~~C.2 (3) (A)

An accumulator unisolated 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 ~~B~~C.1 and Required Action ~~C~~C.2 provide two options, either of which must be performed in the next (1)

(continued)

BASES

ACTIONS 1 B ~~C~~.1, C ~~D~~.1, and C ~~D~~.2 (continued)

661 1

12 hours. By increasing the RCS temperature to more than 310 ~~[275]~~°F, an accumulator pressure of ~~[600]~~ psig cannot exceed the ~~LTOP~~ COMS limits if the accumulators are fully injected. Depressurizing the accumulators below the ~~LTOP~~ COMS limit from the PTLR also gives this protection. 1

1

A

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 ~~LTOP~~ COMS is not likely in the allowed times.

INSERT (from previous page).

D ~~E~~.1 1

In MODE 4 when any RCS cold leg temperature is at or below 310 ~~[275]~~°F, with one required RCS relief valve inoperable, ~~two~~ the RCS relief valve~~s~~ must be restored to OPERABLE status within a Completion Time of 7 days. Two RCS relief valves ~~[in any combination of the PORVS and the RHR suction relief valves]~~ are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component. A 1 6

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.

3 1 E ~~F~~.1

overpressurize

The consequences of operational events that will overpressure 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.

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.

(continued)

BASES

ACTIONS (continued)

F
~~E~~.1

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, B, C, D, E, or ~~F~~ is not met; or

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

The vent must be sized at least ~~2.07~~ 3.0 square inches to ensure that the flow capacity is greater than that required for the worst-case mass input transient reasonable during the applicable MODES. 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.~~3~~2

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, ~~a maximum of [one] [HPI] pump [and a maximum of~~ one centrifugal charging pump~~]~~ are verified deactivated with power removed and the accumulator discharge isolation valves are verified closed and locked out.

The Frequency of 15 COMS minutes before reducing RCS temperature to less than the ~~LTOP~~ arming temperature ensures that the maximum number of pumps permitted OPERABLE is not exceeded. 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)

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~SR 3.4.12.4~~

~~Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction valve and RHR suction isolation valves are open and by testing it in accordance with the Inservice Testing Program. (Refer to SR 3.4.12.7 for the RHR suction isolation valve Surveillance.) This Surveillance is only required to be performed if the RHR suction relief valve is being used to meet this LCO.~~

6

~~The RHR suction valve is verified to be opened every 12 hours. The Frequency is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RHR suction valve remains open.~~

~~The ASME Code, Section XI (Ref. 8), test per Inservice Testing Program verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.~~

SR 3.4.12.~~5~~3 1

capable of relieving at least 475 GPM water flow

The RCS vent of at least ~~[2.07]~~ 3.0 square inches is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a valve that cannot be locked.

b. Once every 31 days for a valve that is locked, sealed, or secured in position. A removed pressurizer safety valve fits this category. 1 PORV

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.~~6~~4 1

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. 2

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.~~6~~4 (continued)

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, such as valve position indication, available to the operator in the control room that verify the PORV block valve remains open.

The ASME Code, Section XI (Ref. 8), test per Inservice Testing Program verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

SR 3.4.12.~~7~~5

~~Each~~ The required RHR suction relief valve shall be both demonstrated OPERABLE by verifying ~~its RHR suction valve and~~ RHR suction isolation valves are open and by testing it in accordance with the Inservice Testing Program. ~~(Refer to SR 3.4.12.4 for the RHR suction valve Surveillance and for a description of the requirements of the Inservice Testing Program.)~~ This Surveillance is only performed if the RHR suction relief valve is being used to satisfy this LCO.

Every 31 days ~~the~~ both RHR suction isolation valves ~~is~~ are verified locked open, with power to the valve operators removed, to ensure that accidental closure will not occur. The "locked open" valves must be locally verified in ~~its~~ the open position with the manual actuator locked in its inactive position. 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.~~8~~6

Performance of an ACOT is required within 12 hours after decreasing RCS temperature to ≤ ~~[275]~~ 310 °F and every 31 days on each required PORV to verify and, as necessary, adjust its lift setpoint. The ACOT will verify the setpoint is within the PTLR allowed maximum limits in the PTLR. PORV actuation could depressurize the RCS and is not required.

(continued)

BASES (1)

SURVEILLANCE REQUIREMENTS

SR 3.4.12.~~8~~6 (continued)

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

(1) A Note 310 has been added indicating that this SR is required to be met 12 hours after decreasing RCS cold leg temperature to ≤ ~~[275]~~°F. The ACOT cannot be performed until in the ~~LTOP~~ COMS MODES when the PORV lift setpoint can be reduced to the ~~LTOP~~ COMS setting. The test must be performed within 12 hours after entering the ~~LTOP~~ COMS MODES. (1)

SR 3.4.12.~~9~~7 (1)

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. (2)

REFERENCES

(4)

Input Insert from Page B 3.4-70A

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. ASME, *Boiler and Pressure Vessel Code*, Section III.
4. FSAR, Chapter [15]
5. 10 CFR 50, Section 50.46.
6. 10 CFR 50, Appendix K.
7. Generic Letter 90-06.
8. ASME, *Boiler and Pressure Vessel Code*, Section XI.

# INSERT

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

2. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operation."

3. ASME Boiler and Pressure Vessel Code, Section III.

4. Watts Bar FSAR, Section 15.2, "Condition II - Faults of Moderate Frequency."

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

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

7. Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light Water Reactors,' pursuant to 10 CFR 50.44(f)."

8. ASME Boiler and Pressure Vessel Code, Section XI.

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 3.4.12

A. This change reflects comments made by the industry to the NRC at the RSTS P&R meeting held in Irvine, CA July 13 - July 20, 1992.

1. Change to reflect WBN specific parameter values or information/terminology.

2. Format change to delete brackets that identify plant specific information/values.

3. Change to correct errors in the STS.

4. Change to specify WBN specific reference information and format.

5. WBN utilizes one centrifugal charging pump to satisfy both normal charging and high pressure ECCS injection functions during MODE 4 when COMS is required.

6. WBN utilizes only one RHR suction relief valve for overpressure protection. This relief valve is located on a common suction header supplying both RHR pumps and on the RCS side of each RHR pump suction valve. Therefore, only the RHR suction isolation valves can isolate this single relief valve from the RCS during shutdown cooling.

7. Change "during" to "while" because "during" can be construed to mean "during the process of shutting down" and excludes the fact that startup can begin from the water solid state also. Water solid conditions only are allowed to occur while shutdown.

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE;

d. 1 gpm total primary-to-secondary LEAKAGE through all steam generators (SGs); and

1 e. [500] gallons per day primary-to-secondary LEAKAGE through any one SG.

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

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE. | A.1 Reduce LEAKAGE to within limits. | 4 hours |
| B. Required Action and associated Completion Time of Condition A not met.<br><br>OR<br><br>Pressure boundary LEAKAGE exists. | B.1 Be in MODE 3.<br><br>AND<br><br>B.2 Be in MODE 5. | 6 hours<br><br><br><br>36 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.13.1 | (A) ----------NOTE---------- Not required to be ~~met~~ performed in MODES 3 ~~and~~ or 4 until ~~after~~ 12 hours of steady-state operation. ---------- | -----NOTE------ (A) Only required to be ~~met~~ performed during steady-state operation ---------- |
| | Perform RCS water inventory balance. | 72 hours |

(A) SR 3.4.13.2 Relocate SR 3.4.4.2 here.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

10 CFR 50, Appendix A, GDC 30 (Ref. 1) requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss-of-coolant accident (LOCA).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

Except for primary-to-secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 1-gpm primary-to-secondary LEAKAGE as the initial condition.

Primary-to-secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve ~~secondary steam~~ secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid. 2 A

The FSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via safety valves and the majority is steamed to the condenser. The 1-gpm primary-to-secondary LEAKAGE is relatively inconsequential.

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes 1-gpm primary-to-secondary LEAKAGE in one generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100 or the staff-approved licensing basis (i.e., a small fraction of these limits).

The RCS operational LEAKAGE satisfies Criterion 2 of the NRC Policy Statement.

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets ~~are~~ is not pressure boundary LEAKAGE. 2 A

(continued)

BASES

LCO (continued)

b. Unidentified LEAKAGE

3

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

pocket

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of identified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary-to-Secondary LEAKAGE through All Steam Generators (SGs)

Total primary-to-secondary LEAKAGE amounting to 1-gpm through all SGs produces acceptable offsite doses in the SLB accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident. Primary-to-secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

e. Primary-to-Secondary LEAKAGE through Any One SG

1

The 500-gallons-per-day limit on one SG is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line rupture. If leaked through many cracks, the cracks are very small, and the above assumption is conservative.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized. (4)

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

A.1

Unidentified LEAKAGE, identified LEAKAGE, or primary-to-secondary LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

If any pressure boundary LEAKAGE exists, or if unidentified LEAKAGE, identified LEAKAGE, or primary-to-secondary LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets are not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

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. In MODE 5, the pressure stresses

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets are not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. Primary-to-secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary-steam and feedwater systems.

The RCS water inventory balance must be met with the reactor at steady-state operating conditions and near operating pressure. Therefore, this SR is not required to be performed in MODES 3 and 4 until 12 hours of steady-state operation near operating pressure have been established.

Steady-state operation is required to perform a proper inventory balance; calculations during maneuvering are not useful and a Note requires the Surveillance to be met when steady state is established. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCS seal injection and return flows.

2 P

3 pocket

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets are not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage

(continued)

BASES

SURVEILLANCE REQUIREMENTS

INSERT BASES 3.4.4.2 (A)

SR 3.4.13.1 (continued)

detection in the prevention of accidents. A Note under the Frequency column states that this SR is required to be met during steady-state operation.

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 30,~~
2. ~~Regulatory Guide 1.45, May 1973,~~
3. ~~FSAR, Section [15].~~

(3)

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criteria 30, "Quality of Reactor Coolant Boundary."
2. Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.
3. Watts Bar FSAR, Section 15.4, "Condition IV-Limiting Faults."

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 3.4.13

A. This change reflects comments made by the industry to the NRC at the RSTS P&R meeting held in Irvine, CA July 13 - July 20, 1992.

1. Format change to delete brackets that identify plant specific information/values.

2. Change to correct errors in the STS.

3. Change to specify WBN specific reference information and format.

4. Editorial change for clarity.

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

LCO 3.4.14 Leakage from each RCS PIV shall be within limit.

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

ACTIONS

3. Flowpaths may be unisolated intermittently under administrative controls.

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

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

---

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Leakage from one or more RCS PIVs not within limit. | ----------NOTE---------- Each valve used to satisfy Required Action A.1 ~~and Required Action A.2~~ (5) must have been verified to meet SR 3.4.14.1 and be in the reactor coolant pressure boundary ~~[for the high pressure portion of the system]~~ (5). ---------- | |
| | A.1 Isolate the high-pressure portion of the affected system from the low-pressure portion by use of one closed manual, deactivated automatic, or check valve. | 4 hours |
| | AND | |
| | | (continued) |

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. (continued) | A.2 Isolate the high-pressure portion of the affected system from the low-pressure portion by use of a second closed manual, deactivated automatic, or check valve. | 72 hours |
| | or | |
| | A.2 Restore RCS PIV to within limits. | 72 hours |
| B. Required Action and associated Completion Time for Condition A not met. | B.1 Be in MODE 3. | 6 hours |
| | AND | |
| | B.2 Be in MODE 5. | 36 hours |
| C. Residual Heat Removal (RHR) System auto-closure interlock function inoperable. | C.1 Isolate the affected penetration by use of one closed manual or deactivated automatic valve. | 4 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.14.1 | ----------NOTES----------<br>1. Not required to be performed in MODES 3 and 4.<br><br>2. Not required to be performed on the RCS PIVs located in the RHR flow path when in the shutdown cooling mode of operation.<br><br>3. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided.<br>---------- | |
| | Verify leakage from each RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure ≥ [2215] psig and ≤ [2255] psig. 2 | In accordance with the Inservice Testing Program, or [18] months 2<br>AND<br><br>Prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months<br><br>AND |

(continued)

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.14.1 (continued) | | Within 24 hours following valve actuation due to automatic or manual action or flow through the valve |
| SR 3.4.14.2 | ----------NOTE----------<br>Not required to be met when the RHR System auto-closure interlock is disabled in accordance with SR 3.4.12.7.<br>----------<br><br>Verify RHR System auto-closure interlock prevents the valves from being opened with a simulated or actual RCS pressure signal ≥ [425] psig. | [18] months |
| SR 3.4.14.3 | ----------NOTE----------<br>Not required to be met when the RHR System auto-closure interlock is disabled in accordance with SR 3.4.12.7.<br>----------<br><br>Verify RHR System auto-closure interlock causes the valves to close automatically with a simulated or actual RCS pressure signal ≥ [600] psig. | [18] months |

# B 3.4 REACTOR COOLANT SYSTEM (RCS)

## B 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3) define RCS PIVs as any two normally closed valves in series within the reactor coolant pressure boundary (RCPB), which separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV Leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

The PIV leakage limit applies to each individual valve. Leakage through both series PIVs in a line must be included as part of the identified LEAKAGE, governed by LCO 3.4.13, "RCS Operational LEAKAGE." This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance (SR 3.4.13.1). A known component of the identified LEAKAGE before operation begins is the least of the two individual leak rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other is leaktight.

3

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that ~~which~~ the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss-of-coolant accident (LOCA) outside of containment, an unanalyzed accident, which could degrade the ability for low pressure injection.

The basis for this LCO is the 1975 NRC "Reactor Safety Study" (Ref. 4) that identified potential intersystem LOCAs as a significant contributor to the risk of core melt. A subsequent study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs.

(continued)

BASES

BACKGROUND (continued)

PIVs are provided to isolate the RCS from the following typically connected systems:

a. Residual Heat Removal (RHR) System;

b. Safety Injection System; and

c. Chemical and Volume Control System. (1)

The PIVs are listed in the FSAR, Section ~~[ ]~~ 3.9 (Ref. 6).

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission-product barrier.

APPLICABLE SAFETY ANALYSES

Reference 4 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the RHR System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the RCPB, and the subsequent pressurization of the RHR System downstream of the PIVs from the RCS. Because the low pressure portion of the RHR System is typically designed for 600 psig, overpressurization failure of the RHR low pressure line would result in a LOCA outside containment and subsequent risk of core melt.

Reference 5 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA.

RCS PIV leakage satisfies Criterion 2 of the NRC Policy Statement.

LCO

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases

(continued)

BASES

LCO (continued)

significantly suggests that something is operationally wrong and corrective action must be taken.

The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm. The previous criterion of 1 gpm for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and resulted in higher personnel radiation exposures. A study concluded a leakage rate limit based on valve size was superior to a single allowable value.

Reference 7 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized.

In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.

ACTIONS

The Actions are modified by ~~two~~ three Notes. Note 1 provides clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.

INSERT 1 T-E

(continued)

BASES

LCO (continued)

significantly suggests that something is operationally wrong and corrective action must be taken.

The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm. The previous criterion of 1 gpm for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and resulted in higher personnel radiation exposures. A study concluded a leakage rate limit based on valve size was superior to a single allowable value.

Reference 7 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one-half power.

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized.

In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.

ACTIONS

The Actions are modified by ~~two~~ three Notes. Note 1 provides clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function. INSERT 1 (attached) (A)

(continued)

INSERT 1 FOR 3.4.14 BASES

Note 3 allows unisolating flow paths which have been isolated in accordance with the ACTIONS. This may be necessary to place systems in service to support necessary operations, such as unisolating [RHR] and placing ~~[shutdown cooling system]~~ in service to support operation in MODE 4. The administrative controls assure the system's flowpath is not unisolated unless the RCS pressure is below the design pressure of the system.

decay heat removal system

BASES

ACTIONS (continued)

A.1 and A.2

The flow path must be isolated, ~~by two valves~~. Required Actions A.1 and A.2 are modified by a Note that the valves used for isolation must meet the same leakage requirements as the PIVs and must be in the RCPB ~~[for the high pressure portion of the system]~~.

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 hours allows the actions and restricts the operation with leaking isolation valves.

Required Action A.2 specifies that the double isolation barrier of two valves be restored by closing some other valve qualified for isolation or restoring one leaking PIV. The 72 hour Completion Time after exceeding the limit considers the time required to complete the Action and the low probability of a second valve failing during this time period.

or

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. ~~(Reviewer Note: Two options are provided for Required Action A.2. The second option (72 hour restoration) is appropriate if isolation of a second valve would place the unit in an unanalyzed condition.)~~

B.1 and B.2

If leakage cannot be reduced, with the system isolated, ~~or the other Required Actions accomplished~~, 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

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

6

C.1

The inoperability of the RHR auto-closure interlock renders the RHR suction isolation valves incapable of: isolating in response to a high-pressure condition and preventing inadvertent opening of the valves at RCS pressures in excess of the RHR systems design pressure. If the RHR auto-closure interlock is inoperable, operation may continue as long as the affected RHR suction penetration is closed by at least one closed manual or deactivated automatic valve within 4 hours. This Action accomplishes the purpose of the auto-closure function.

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 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.

3 2

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 months] Frequency is required in 10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice ~~Testing~~ Inspection Program, is within the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. 9), and is based on the need to perform such surveillances under the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

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 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

6

SR 3.4.14.2 and SR 3.4.14.3

Verifying that the RHR auto-closure interlocks are OPERABLE ensures that RCS pressure will not pressurize the RHR system beyond 125% of its design pressure of [600] psig. The interlock setpoint that prevents the valves from being opened is set so the actual RCS pressure must be less than [425] psig to open the valves. This setpoint ensures the RHR design pressure will not be exceeded and the RHR relief valves will not lift. The [18] month Frequency is based on the need to perform the Surveillance under conditions that apply during a plant outage. The [18] month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

These SRs are modified by Notes allowing the RHR auto-closure function to be disabled when using the RHR System suction relief valves for cold overpressure protection in accordance with SR 3.4.12.7.

REFERENCES

4

INPUT INSERT FROM PAGE B 3.4-83A

1. ~~10 CFR 50, Section 50.2.~~
2. ~~10 CFR 50, Section 50.55a, Subsection (c).~~
3. ~~10 CFR 50, Appendix A, Section V, GDC 55.~~
4. ~~WASH-1400 (NUREG-75/014), Appendix V, U.S. Nuclear Regulatory Commission (NRC), October 1975.~~
5. ~~NUREG-0677, NRC, May 1980.~~
6. ~~[Document containing list of PIVs.]~~
7. ~~ASME, Boiler and Pressure Vessel Code, Section XI, Subsection IWV, Paragraph IWV-3423(c).~~
8. ~~10 CFR 50, Section 50.55a, Subsection (g).~~
9. ~~ASME, Boiler and Pressure Vessel Code, Section XI, Subsection IWV, Paragraph IWV-3422.~~

(4)

# INSERT


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).

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

9. ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWV, "Inservice Testing of Valves in Nuclear Power Plants," Paragraph IWV-3422.

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 3.4.14

A. This change reflects comments made by the industry to the NRC at the RSTS P&R meeting held in Irvine, CA July 13 - July 20, 1992.

1. Change to reflect WBN specific parameter values.

2. Format change to delete brackets that identify plant specific information/values.

3. Change to correct errors in the STS.

4. Change to specify WBN specific reference information and format.

5. WBN will utilize the "second" option for Required Action A.2.

6. WBN is in the process of, and intends to complete, deleting the RHR System auto-closure interlock function.

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.15 RCS Leakage Detection Instrumentation

LCO 3.4.15 The following RCS leakage detection instrumentation shall be OPERABLE:

1

a. One containment pocket sump (level ~~or discharge flow~~) monitor; AND

b. One lower containment atmosphere radioactivity monitor (gaseous ~~or~~ AND particulate). ~~[and~~

5 ~~c. One containment air cooler condensate flow rate monitor].~~

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

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| 5 A. Required containment pocket sump level monitor inoperable. | ----------NOTE---------- LCO 3.0.4 is not applicable. ---------- | |
| | A.1 Perform SR 3.4.13.1. | Once per 24 hours |
| | AND | |
| | A.2 Restore required containment pocket sump level monitor to OPERABLE status. | 5 30 days move up A |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| B. Required containment atmosphere radioactivity monitor inoperable. | 2 ----------NOTE---------- LCO 3.0.4 is not applicable. ---------- | |
| | B.1.1 Analyze grab samples of the containment atmosphere. | Once per 24 hours |
| | OR | |
| | B.1.2 Perform SR 3.4.13.1. | Once per 24 hours |
| 2 | AND | |
| | B.2.1 Restore required containment atmosphere radioactivity monitor to OPERABLE status. | 30 days 5 |
| | OR | |
| | B.2.2 Verify containment air cooler condensate flow rate monitor is OPERABLE. | 30 days |
| C. Required containment air cooler condensate flow rate monitor inoperable. | C.1 Perform SR 3.4.15.1. | Once per 8 hours 5 |
| | OR | |
| | C.2 Perform SR 3.4.13.1. | Once per 24 hours |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| D. Required containment atmosphere radioactivity monitor inoperable.<br><br>AND<br><br>Containment air cooler condensate flow rate monitor inoperable. | D.1 | Restore containment atmosphere radioactivity monitor to OPERABLE status. | 30 days |
| | OR | | |
| | D.2 | Restore containment air cooler condensate flow rate monitor to OPERABLE status. | 30 days |
| C ~~E~~. Required Action and associated Completion Time not met. | C ~~E~~.1 | Be in MODE 3. | 6 hours |
| | AND | | |
| | C ~~E~~.2 | Be in MODE 5. | 36 hours |
| D ~~F~~. All required monitors inoperable. | D ~~F~~.1 | Enter LCO 3.0.3. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.15.1 | Perform a CHANNEL CHECK of the required containment atmosphere radioactivity monitor. | 12 hours |
| SR 3.4.15.2 | Perform an ACOT of the required containment atmosphere radioactivity monitor. | 92 ~~31~~ days |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.15.3 | Perform a CHANNEL CALIBRATION of the required containment sump monitor. | [18] months |
| SR 3.4.15.4 | Perform a CHANNEL CALIBRATION of the required containment atmosphere radioactivity monitor. | [18] months |
| SR 3.4.15.5 | Perform a CHANNEL CALIBRATION of the required containment air cooler condensate flow rate monitor. | [18] months |

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

GDC 30 of Appendix A to 10 CFR 50 (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE.

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can be readily detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The containment sump used to collect unidentified LEAKAGE [is] ~~[(or) and air cooler condensate flow rate monitor] [are]~~ instrumented to alarm for increases of 0.5 to 1.0 gpm in the normal flow rates. This sensitivity is acceptable for detecting increases in unidentified LEAKAGE.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel-element-cladding contamination or cladding defects. Instrument sensitivities of $10^{-9}$ $\mu$Ci/cc radioactivity for particulate monitoring and of $10^{-6}$ $\mu$Ci/cc radioactivity for gaseous monitoring are practical for these leakage detection systems. Radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS LEAKAGE.

~~An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew-point temperature measurements can thus be used to monitor humidity levels of the containment atmosphere as an~~

(continued)

BASES

BACKGROUND (continued)

indicator of potential RCS LEAKAGE. A 1°F increase in dew-point is well within the sensitivity range of available instruments.

Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed increases in liquid flow into or from the containment sump. ~~[and condensate flow from air coolers]~~. Humidity level monitoring is considered most useful as an indirect alarm or indication to alert the operator to a potential problem. Humidity monitors are not required by this LCO.

Air temperature and pressure monitoring methods may also be used to infer unidentified LEAKAGE to the containment. Containment temperature and pressure fluctuate slightly during plant operation, but a rise above the normally indicated range of values may indicate RCS leakage into the containment. The relevance of temperature and pressure measurements are affected by containment-free volume and, for temperature, detector location. Alarm signals from these instruments can be valuable in recognizing rapid and sizable leakage to the containment. Temperature and pressure monitors are not required by this LCO.

APPLICABLE SAFETY ANALYSES

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. The system response times and sensitivities are described in the FSAR (Ref. 3). Multiple instrument locations are utilized, if needed, to ensure that the transport delay time of the leakage from its source to an instrument location yields an acceptable overall response time.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting, as well as monitoring, RCS LEAKAGE into the containment area are necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should a leakage occur detrimental to the safety of the unit and the public.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

RCS leakage detection instrumentation satisfies Criterion 1 of the NRC Policy Statement.

LCO

One method of protecting against large RCS leakage derives from the ability of instruments to rapidly detect extremely small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition, when RCS LEAKAGE indicates possible RCPB degradation.

The LCO is satisfied when monitors of diverse measurement means are available. Thus, the containment sump monitor, in combination with a gaseous ~~or~~ particulate radioactivity monitor ~~[and a containment air cooler condensate flow rate monitor]~~, provides an acceptable minimum.

APPLICABILITY

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE.

In MODE 5 or 6, the temperature is to be ≤ 200°F and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

ACTIONS

A.1 and A.2

With the required containment sump monitor inoperable, no ~~other~~ form of sampling can provide the equivalent information; however, the containment atmosphere radioactivity monitor will provide indications of changes in leakage. Together with the atmosphere monitor, the periodic surveillance for RCS water inventory balance, SR 3.4.13.1,

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage.

Restoration of the sump monitor to OPERABLE status in a Completion Time of 30 days is required to regain the function after the monitor's failure. This time is acceptable, considering the Frequency and adequacy of the RCS water inventory balance required by Required Action A.1.

Required Action A.1 is modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the containment sump monitor is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

B.1.1, B.1.2, B.2.1, and B.2.2

With both gaseous and particulate containment atmosphere radioactivity-monitoring instrumentation channels inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information.

With a sample obtained and analyzed or water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of at least one of the radioactivity monitors. ~~Alternatively, continued operation is allowed if the air cooler condensate flow rate monitoring system is OPERABLE, provided grab samples are taken every 24 hours.~~

The 24 hour interval provides periodic information that is adequate to detect leakage. The 30 day Completion Time recognizes at least one other form of leakage detection is available.

Required Action B.1 and Required Action B.2 are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the gaseous and particulate containment atmosphere

(continued)

BASES

ACTIONS

B.1.1, B.1.2, B.2.1, and B.2.2 (continued)

radioactivity monitor channel is inoperable. This allowance is provided because other instrumentation is available to monitor for RCS LEAKAGE.

C.1 and C.2

With the required containment air cooler condensate flow rate monitor inoperable, alternative action is again required. Either SR 3.4.15.1 must be performed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. Provided a CHANNEL CHECK is performed every 8 hours or a water inventory balance is performed every 24 hours, reactor operation may continue while awaiting restoration of the containment air cooler condensate flow rate monitor to OPERABLE status.

The 24 hour interval provides periodic information that is adequate to detect RCS LEAKAGE.

D.1 and D.2

With the required containment atmosphere radioactivity monitor and the containment air cooler condensate flow rate monitor inoperable, the only means of detecting leakage is the containment sump monitor. This Condition does not provide the required diverse means of leakage detection. The Required Action is to restore either of the inoperable monitors to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a reduced configuration for a lengthy time period.

~~E~~.1 and ~~E~~.2

If a Required Action of Condition A or B, [C], or [D] cannot be met, 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 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the

(continued)

BASES

ACTIONS

C C
~~E~~.1 and ~~E~~.2 (continued)

1 required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D
~~F~~.1

With all required monitors inoperable, no automatic means of monitoring leakage are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

SR 3.4.15.1 requires the performance of a CHANNEL CHECK of the containment atmosphere radioactivity monitor. The check gives reasonable confidence that the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off-normal conditions.

SR 3.4.15.2

3 SR 3.4.15.2 requires the performance of an ACOT ~~[a CHANNEL FUNCTIONAL TEST]~~ on the required containment atmosphere radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. The Frequency of ~~31~~ 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation. 7

1 AND

SR 3.4.15.3 OR SR 3.4.15.4, ~~[and SR 3.4.15.5]~~ 5

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]~~ 18 months is a typical refueling cycle and considers channel reliability. Again, operating experience has proven that this Frequency is acceptable. 2

(continued)

BASES (continued)

REFERENCES

1. ~~10 CFR 50, Appendix A, Section IV, GDC 30.~~
2. ~~Regulatory Guide 1.45, U.S. Nuclear Regulatory Commission.~~
3. ~~FSAR, Section [ ].~~

4

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.
3. Watts Bar FSAR, Section 5.2.7, "RCPB Leakage Detection Systems."

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 3.4.15

A. This change reflects comments made by the industry to the NRC at the RSTS P&R meeting held in Irvine, CA July 13 - July 20, 1992.

1. Change to reflect WBN specific parameter values.

2. Format change to delete brackets that identify plant specific information/values.

3. Change to correct errors in the STS.

4. Change to specify WBN specific reference information and format.

5. WBN utilizes only the containment pocket sump level monitor and containment radioactivity monitor as required instrumentation for RCS leak detection.

6. Humidity, air temperature, and pressure monitoring are not used to perform RCS leak detection nor are they required by this LCO. These discussions have been deleted as inappropriate material for this Bases.

7. The COT frequency has been changed to 92 days. This is consistent with the COT frequency in LCO 3.3.6 - 3.3.8 for other radiation monitoring channels.

8. Editorial change for clarity.

9. Change to delete information not applicable to Watts Bar.

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be limited to:

a. DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm; and

b. Gross specific activity ≤ 100/$\bar{E}$ μCi/gm. 3

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature ($T_{avg}$) ≥ 500°F.

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. DOSE EQUIVALENT I-131 > 1.0 μCi/gm. 3 | A.1 | Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.16-1. | Once per 4 hours |
| | AND | | |
| | A.2 | Restore DOSE EQUIVALENT I-131 to within limit. | 48 hours |
| B. Gross specific activity of the reactor coolant not within limit. | B.1 | Perform SR 3.4.16.2. | 4 hours |
| | AND | | |
| | B.2 | Be in MODE 3 with $T_{avg}$ < 500°F. | 6 hours |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| C. Required Action and associated Completion Time of Condition A not met.<br><br>OR<br><br>DOSE EQUIVALENT I-131 in the unacceptable region of Figure 3.4.16-1. | C.1 Be in MODE 3 with $T_{avg}$ < 500°F. | 6 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.16.1 | Verify reactor coolant gross specific activity ≤ 100/E μCi/gm. | 7 days |
| SR 3.4.16.2 | ----------NOTE----------<br>Only required to be performed in MODE 1.<br>----------<br><br>Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm. | 14 days<br><br>AND<br><br>Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1-hour period |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.16.3 | ----------NOTE----------<br>Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.<br>----------<br><br>Determine $\bar{E}$ from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. | 184 days |

~~THIS PAGE IS A PLACEHOLDER FOR FIGURE 3.4.16-1~~

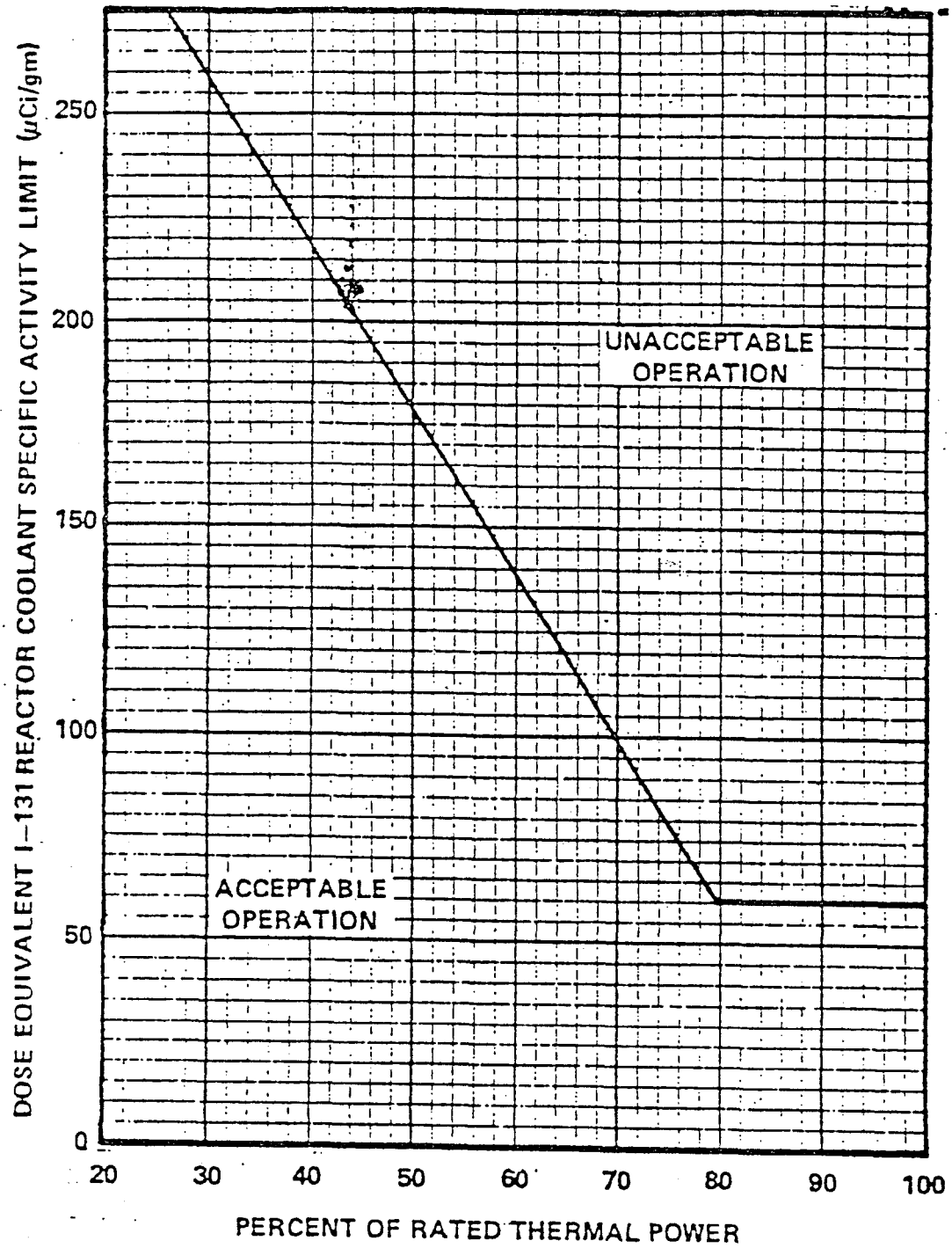


Figure 3.4.16-1 ~~(Page 1 of 1)~~

Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity Limit Versus Percent of RATED THERMAL POWER

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum dose to the whole body and the thyroid an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref. 1). The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline limits. The limits in the LCO are standardized, based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.

The parametric evaluations showed the potential offsite dose levels for a SGTR accident were an appropriately small fraction of the 10 CFR 100 dose guideline limits. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary will not exceed a small fraction of the 10 CFR 100 dose guideline limits following a SGTR accident. The SGTR safety analysis (Ref. 2) assumes the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm. The safety analysis assumes the specific activity of the secondary coolant at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.~~8~~ 14, "Secondary Specific Activity." 3

1

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The analysis for the SGTR accident establishes the acceptance limits for RCS specific activity. Reference to this analysis is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The analysis is for two cases of reactor coolant specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the I-131 activity in the reactor coolant by a factor of about 50 immediately after the accident. The second case assumes the initial reactor coolant iodine activity at 60.0 μCi/gm DOSE EQUIVALENT I-131 due to a preaccident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant assumes 1% failed fuel, which ~~closely equals~~ approximates the LCO limit of 100/E μCi/gm for gross specific activity.

The analysis also assumes a loss of offsite power at the same time as the SGTR event. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal.

The coincident loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power-operated relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends.

The safety analysis shows the radiological consequences of a SGTR accident are within a small fraction of the Reference 1 dose guideline limits. Operation with iodine-specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1, in the applicable specification, for more than 48 hours. The safety analysis has concurrent and preaccident iodine spiking levels up to 60.0 μCi/g DOSE EQUIVALENT I-131.

The remainder of the above-limit permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of a SGTR accident occurring during the established 48 hour time limit. The occurrence of a SGTR

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline limits.

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of the NRC Policy Statement.

LCO

3

The specific iodine activity is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the reactor coolant is limited to the number of μCi/gm equal to 100 divided by E (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary during the Design Basis Accident (DBA) will be a small fraction of the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary during the DBA will be a small fraction of the allowed whole body dose.

The SGTR accident analysis (Ref. 2) shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a SGTR, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of a SGTR to within the acceptable site boundary dose values.

For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the limits of Figure 3.4.16-1 are not exceeded. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is to continue to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours. The Completion Time of 48 hours is ~~required~~ allowed, ~~if~~ since the limit violation may have resulted from normal iodine spiking. (3)

(A) (3)

DOSE EQUIVALENT I-131 specific activity is considered out of limits if the equipment used to measure DOSE EQUIVALENT I-131 specific activity is determined to be inoperable at the time SR 3.4.16.2 is performed. Required Action A.1 and Required Action A.2 apply to restoring such equipment to OPERABLE status.

B.1 and B.2

With the gross specific activity in excess of the allowed limit, an analysis must be performed within 4 hours to determine DOSE EQUIVALENT I-131. The Completion Time of 4 hours is required to obtain and analyze a sample.

The change within 6 hours to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents venting the SG to the environment in an SGTR event. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

C.1 (3)

If a Required Action and the associated Completion Time of Condition A is (are) not met or if the DOSE EQUIVALENT I-131 is in the unacceptable region of Figure 3.4.16-1, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3

(continued)

BASES

ACTIONS

C.1 (continued)

below 500°F from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

2

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma-isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once every 7 days. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with $T_{avg}$ at least 500°F. The 7 day Frequency considers the unlikelihood of a gross fuel failure during the time.

SR 3.4.16.2

This Surveillance is performed in MODE 1 only to ensure iodine remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine-activity level, considering gross activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results.

SR 3.4.16.3

A radiochemical analysis for $\bar{E}$ determination is required every 184 days (6 months) with the plant operating in MODE 1 equilibrium conditions. The $\bar{E}$ determination directly relates to the LCO and is required to verify plant operation

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.16.3 (continued)

within the specified gross activity LCO limit. The analysis for E is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. The Frequency of 184 days recognizes E does not change rapidly.

This SR has been modified by a Note that indicates sampling is required to be performed within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures that the radioactive materials are at equilibrium so the analysis for E is representative and not skewed by a crud burst or other similar abnormal event.

REFERENCES

1. ~~10 CFR 100, Section 100.11, 1973.~~
2. ~~FSAR, Section [15.6.3].~~

4

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," 1973.
2. Watts Bar FSAR, Section ~~15.6.3~~ 15.4, "Condition IV - Limiting Faults."

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 3.4.16

1. Change to reflect WBN specific parameter values.

2. Format change to delete brackets that identify plant specific information/values.

3. Change to correct errors in the STS.

4. Change to specify WBN specific reference information and format.

5. Editorial change for clarity.

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.17 RCS Loop Isolation Valves

LCO 3.4.17 Each RCS hot and cold leg loop isolation valve shall be open with power removed from each isolation valve operator.

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

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each RCS loop isolation valve.
---

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. Power available to one or more loop isolation valve operators. | A.1 | Remove power from loop isolation valve operators. | 30 minutes |
| B. ---------NOTE--------- All Required Actions shall be completed whenever this Condition is entered. ---------- One or more RCS loop isolation valves closed. | B.1 | Maintain valve(s) closed. | Immediately |
| | AND | | |
| | B.2 | Be in MODE 4. | 6 hours |
| | AND | | |
| | B.3 | Be in MODE 5. | 36 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.17.1 | Verify each RCS loop isolation valve is open and power is removed from each loop isolation valve operator. | 31 days |

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.17 RCS Loop Isolation Valves

BASES

BACKGROUND

The RCS may be operated with loops isolated in order to perform maintenance. While operating with a loop isolated, there is potential for inadvertently opening the isolation valves in the isolated loop. In this event, the coolant in the isolated loop would suddenly begin to mix with the coolant in the operating loops. This situation has the potential of causing a positive reactivity addition with a corresponding reduction of SDM if:

a. The temperature in the isolated loop is lower than the temperature in the operating loops (cold water incident); or

b. The boron concentration in the isolated loop is lower than the boron concentration in the operating loops (boron dilution incident).

As discussed in the FSAR (Ref. 1), the startup of an isolated loop is performed in a controlled manner that virtually eliminates any sudden positive reactivity addition from cold water and/or boron dilution because:

a. LCO 3.4.18, "RCS Isolated Loop Startup," and plant operating procedures require that the boron concentration in the isolated loop be maintained higher than the boron concentration of the operating loops, thus eliminating the potential for introducing coolant from the isolated loop that could dilute the boron concentration in the operating loops;

b. The cold leg loop isolation valve cannot be opened unless the temperatures of both the hot and cold legs of the isolated loop are within 20°F of the temperatures of the hot and cold legs of the operating loops (compliance is ensured by operating procedures and automatic interlocks); and

c. Other automatic interlocks, all of which are part of the Reactor Protection System (RPS), prevent opening the hot leg loop isolation valve unless the cold leg loop isolation valve is fully closed.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

During startup of an isolated loop in accordance with LCO 3.4.18, the cold leg loop isolation valve interlocks and operating procedures prevent opening of the valve until the isolated loop and operating loop boron concentrations and temperatures are equalized. This ensures that any undesirable reactivity effect from the isolated loop does not occur.

The safety analyses assume a minimum SDM as an initial condition for Design Basis Accidents (DBAs) (Ref. 1). Violation of the LCO, combined with mixing of the isolated loop coolant into the operating loops, could result in the SDM being less than that assumed in the safety analyses.

The above analyses are for DBAs that establish the acceptance limits for the RCS loop isolation valves. Reference to the analyses for these DBAs is used to assess changes to the RCS loop isolation valves as they relate to the acceptance limits.

The boron concentration of an isolated loop may affect SDM and therefore RCS loop isolation valves satisfy Criterion 2 of the NRC Policy Statement.

LCO

This LCO ensures that a loop isolation valve that becomes closed in MODES 1 through 4 is fully isolated and the plant placed in MODE 5. Loop isolation valves are used for performing maintenance when the plant is in MODE 5 or 6, and startup of an isolated loop is covered by LCO 3.4.18.

APPLICABILITY

In MODES 1 through 4, this LCO is applicable when unisolating an isolated loop with a boron concentration less than that of the operating loops may cause an inadvertent criticality.

In MODES 5 and 6, the SDM of the operating loops is large enough to permit operation with isolated loops. In these MODES, controlled startup of isolated loops is possible without significant risk of inadvertent criticality.

(continued)

BASES (continued)

ACTIONS

The Actions have been provided with a Note to clarify that all RCS loop isolation valves for this LCO are treated as separate entities, each with separate Completion Times, i.e., the Completion Time is on a component basis.

A.1

If power is inadvertently restored to one or more loop isolation valve operators, the potential exists for accidental isolation of a loop with a subsequent inadvertent startup of the isolated loop. The loop isolation valves have motor operators. Therefore, these valves will maintain their last position when power is removed from the valve operator. With power applied to the valve operators, only the interlocks prevent the valve from being operated. Although operating procedures and interlocks make the occurrence of this event unlikely, the prudent action is to remove power from the loop isolation valve operators. The Completion Time of 30 minutes to remove power from the loop isolation valve operators is sufficient considering the complexity of the task.

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

Should a loop isolation valves be closed in MODES 1 through 4, the affected loop must be fully isolated immediately and the plant placed in MODE 5 to preclude inadvertent startup of the loop and the potential inadvertent criticality. The Completion Time of Required Action B.1 allows time for borating the operating loops to a shutdown boration level such that the plant can be brought to MODE 4 within 6 hours and 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 3.4.17.1

The Surveillance is performed at least once per 31 days to ensure that the RCS loop isolation valves are open, with power removed from the loop isolation valve operators. The primary function of this Surveillance is to ensure that

(continued)

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 3.4.17

1. Specification not applicable to WBN.

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.18 RCS Isolated Loop Startup

LCO 3.4.18 Each RCS isolated loop shall remain isolated with:

a. The hot and cold leg isolation valves closed if boron concentration of the isolated loop is less than boron concentration of the operating loops; and

b. The cold leg isolation valve closed if the cold leg temperature of the isolated loop is > [20]°F below the highest cold leg temperature of the operating loops:

APPLICABILITY: MODES 5 and 6.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Isolated loop hot or cold leg isolation valve open with LCO requirements not met. | A.1 --------NOTE--------- Only required if boron concentration requirement not met. ---------- Close hot and cold leg isolation valves. | Immediately |
| | OR | |
| | A.2 --------NOTE--------- Only required if temperature requirement not met. ---------- Close cold leg isolation valve. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.18.1 | Verify cold leg temperature of isolated loop is ≤ [20]°F below the highest cold leg temperature of the operating loops. | Within 30 minutes prior to opening the cold leg isolation valve in isolated loop |
| SR 3.4.18.2 | Verify boron concentration of isolated loop is greater than or equal to boron concentration of the operating loops. | Within 2 hours prior to opening the hot or cold leg isolation valve in isolated loop |

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.18 RCS Isolated Loop Startup

BASES

BACKGROUND

The RCS may be operated with loops isolated in MODES 5 and 6 in order to perform maintenance. While operating with a loop isolated, there is potential for inadvertently opening the isolation valves in the isolated loop. In this event, the coolant in the isolated loop would suddenly begin to mix with the coolant in the operating loops. This situation has the potential of causing a positive reactivity addition with a corresponding reduction of SDM if

a. The temperature in the isolated loop is lower than the temperature in the operating loops (cold water incident); or

b. The boron concentration in the isolated loop is lower than the boron concentration in the operating loops (boron dilution incident).

As discussed in the FSAR (Ref. 1), the startup of an isolated loop is done in a controlled manner that virtually eliminates any sudden reactivity addition from cold water or boron dilution because:

a. This LCO and plant operating procedures require that the boron concentration in the isolated loop be maintained higher than the boron concentration of the operating loops, thus eliminating the potential for introducing coolant from the isolated loop that could dilute the boron concentration in the operating loops.

b. The cold leg loop isolation valve cannot be opened unless the temperatures of both the hot leg and cold leg of the isolated loop are within 20°F of the operating loops. Compliance with the temperature requirement is ensured by operating procedures and automatic interlocks.

c. Other automatic interlocks prevent opening the hot leg loop isolation valve unless the cold leg loop isolation valve is fully closed. All of the interlocks are part of the Reactor Protection System.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

During startup of an isolated loop, the cold leg loop isolation valve interlocks and operating procedures prevent opening the valve until the isolated loop and operating loop boron concentrations and temperatures are equalized. This ensures that any undesirable reactivity effect from the isolated loop does not occur.

The safety analyses assume a minimum SDM as an initial condition for Design Basis Accidents (DBAs). Violation of this LCO could result in the SDM being reduced in the operating loops to less than that assumed in the safety analyses.

The boron concentration of an isolated loop may affect SDM and therefore RCS isolated loop startup satisfies Criterion 2 of the NRC Policy Statement.

LCO

Loop isolation valves are used for performing maintenance when the plant is in MODE 5 or 6. This LCO ensures that the loop isolation valves remain closed until the differentials of temperature and boron concentration between the operating loops and the isolated loops are within acceptable limits.

APPLICABILITY

In MODES 5 and 6, the SDM of the operating loops is large enough to permit operation with isolated loops. Controlled startup of isolated loops is possible without significant risk of inadvertent criticality. This LCO is applicable under these conditions.

ACTIONS

A.1 and A.2

Required Action A.1 and Required Action A.2 assume that the prerequisites of the LCO are not met and a loop isolation valve has been inadvertently opened. Therefore, the Actions require immediate closure of isolation valves to preclude a boron dilution event or a cold water event. However, each Required Action is preceded by a Note that states that Action is required only when a specific concentration or temperature requirement is not met.

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JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 3.4.18

1. Specification not applicable to WBN.

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.197 RCS Loops – Test Exceptions

BASES (2) (A)

BACKGROUND

The primary purpose of this test exception is to provide an exception to LCO 3.4.4, "RCS Loops – MODES 1 and 2," to permit reactor criticality under no flow conditions during certain PHYSICS TESTS (natural circulation demonstration, station blackout, and loss of offsite power) to be performed while at low THERMAL POWER levels. Section XI of 10 CFR Part 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the power plant as specified in GDC 1, Quality Standards and Records (Ref. 2).

The key objectives of a test program are to provide assurance that the facility has been adequately designed to validate the analytical models used in the design and analysis, to verify the assumptions used to predict plant response, to provide assurance that installation of equipment at the unit has been accomplished in accordance with the design, and to verify that the operating and emergency procedures are adequate. Testing is performed prior to initial criticality, during startup, and following low power operations.

The tests will include verifying the ability to establish and maintain natural circulation following a plant trip between 10% and 20% RTP, performing natural circulation cooldown on emergency power, and during the cooldown, showing that adequate boron mixture occurs and that pressure can be controlled using auxiliary spray and pressurizer heaters powered from the emergency power sources.

APPLICABLE SAFETY ANALYSES

The tests described above require operating the plant without forced convection flow and as such are not bounded by any safety analyses. However, operating experience has

(continued)

BASES (continued)

ACTIONS

A.1

When THERMAL POWER is ≥ the P-7 interlock setpoint 10%, the only acceptable action is to ensure the reactor trip breakers are opened immediately in accordance with Required Action A.1 to prevent operation of the fuel beyond its design limits. Opening the reactor trip breakers will shutdown the reactor and prevent operation of the fuel outside of its design limits.

SURVEILLANCE REQUIREMENTS

SR 3.4.1~~9~~7.1 (1)

Verification (2) that the power level is < the P-7 interlock setpoint ~~(~~10%~~)~~ will ensure that the fuel design criteria are not violated during the performance of the PHYSICS TESTS. The Frequency of once per hour is adequate to ensure that the power level does not exceed the limit. Plant operations are conducted slowly during the performance of PHYSICS TESTS and monitoring the power level once per hour is sufficient to ensure that the power level does not exceed the limit.

SR 3.4.1~~9~~7.2 (1)

The power range and intermediate range neutron detectors and the P-7 interlock setpoint must be verified to be OPERABLE and adjusted to the proper value. An ACOT is performed within 12 hours prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS. The time limit of 12 hours is sufficient to ensure that the instrumentation is OPERABLE shortly before initiating PHYSICS TESTS.

REFERENCES

1. ~~10 CFR 50, Appendix B, Section XI.~~
2. ~~10 CFR 50, Appendix A, GDC 1, 1988.~~

(3)

1. Title 10 Code of Federal Regulations, Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plant and Fuel Reprocessing Plants."
2. Title 10 Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," 1988.

## 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### 3.5.1 Accumulators

LCO 3.5.1 [Four] ECCS accumulators shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with pressurizer pressure > [1,000] psig.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One accumulator inoperable due to boron concentration not within limits. | A.1 Restore boron concentration to within limits. | 72 hours |
| B. One accumulator inoperable for reasons other than Condition A. | B.1 Restore accumulator to OPERABLE status. | 1 hour |
| C. Required Action and associated Completion Time of Condition A or B not met. | C.1 Be in MODE 3.<br>AND<br>C.2 Reduce pressurizer pressure to ≤ [1,000] psig. | 6 hours<br><br>12 hours |
| D. Two or more accumulators inoperable. | D.1 Enter LCO 3.0.3. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.1.1 | Verify each accumulator isolation valve is fully open. | 12 hours |
| SR 3.5.1.2 | Verify borated water volume in each accumulator is ≥ ~~[7,853~~ 7,717 gallons, ( )% and ≤ ~~8,171~~ 7,961 gallons ( )%. | 12 hours |
| SR 3.5.1.3 | Verify nitrogen cover pressure in each accumulator is ≥ ~~[385]~~ 602 psig and ≤ ~~[481]~~ 661 psig. | 12 hours |
| SR 3.5.1.4 | Verify boron concentration in each accumulator is ≥ [1,900] ppm and ≤ [2,100] ppm. | 31 days<br>AND<br>-----NOTE------<br>~~Required to be met~~ Only required to be performed for affected accumulators<br>----------<br>Once within 6 hours after each solution volume increase of ≥ ~~[ ]~~ 75 gallons ~~(1% of tank volume)~~ that is not the result of addition from the refueling water storage tank |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.1.5 | ----------NOTE---------- ~~Required to be performed~~ when pressurizer pressure ≥ [2,000] psig. ---------- Verify power is removed from each accumulator isolation valve operator. | 6 31 days |

# B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

## B 3.5.1 Accumulators

BASES

BACKGROUND

The functions of the ECCS accumulators are to supply water to the reactor vessel during the blowdown phase of a loss-of-coolant accident (LOCA), to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Reactor Coolant System (RCS) makeup for a small-break LOCA.

The blowdown phase of a large-break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission-product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

In the refill phase of a LOCA, which immediately follows the blowdown phase, reactor coolant inventory has vacated the core through steam flashing and ejection out through the break. The core is essentially in adiabatic heatup. The balance of accumulator inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing re-flood of the core with the addition of safety injection (SI) water.

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The accumulators are passive components, since no operator or control actions are required in order for them to perform their function. Internal accumulator tank pressure is sufficient to discharge the accumulator contents to the RCS, if RCS pressure decreases below the accumulator pressure.

Each accumulator is piped into an RCS cold leg via an accumulator line and is isolated from the RCS by a motor-operated isolation valve and two check valves in series. The motor-operated isolation valves are interlocked by P-11 with the pressurizer pressure measurement channels to ensure that the valves will automatically open as RCS pressure increases to above the permissive circuit P-11 setpoint.

(continued)

BASES

BACKGROUND (continued)

This interlock also prevents inadvertent closure of the valves during normal operation prior to an accident. The valves will automatically open, however, as a result of an SI signal. These features ensure that the valves meet the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standard 279-1971 (Ref. 1) for "operating bypasses" and that the accumulators will be available for injection without reliance on operator action.

The accumulator size, water volume, and nitrogen cover pressure are selected so that three of the four accumulators are sufficient to partially cover the core before significant clad melting or zirconium-water reaction can occur following a LOCA. The need to ensure that three accumulators are adequate for this function is consistent with the LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the LOCA.

APPLICABLE SAFETY ANALYSES

The accumulators are assumed OPERABLE in both the large- and small-break LOCA analyses at full power (Ref. 2). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow. In the early stages of a LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power is required by regulations and conservatively imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break.

The limiting large-break LOCA is a double-ended guillotine break at the discharge of the reactor coolant pump. During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

As a conservative estimate, no credit is taken for ECCS pump flow until an effective delay has elapsed. This delay accounts for the diesels starting and the pumps being loaded and delivering full flow. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. During this time, the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed during the blowdown stage of a large-break LOCA.

The worst-case small-break LOCA analyses also assume a time delay before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated solely by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators and centrifugal charging pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease until they are not required and the centrifugal charging pumps become solely responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. 3) will be met following a LOCA:

a. Maximum fuel element cladding temperature is $\leq$ 2200°F;

b. Maximum cladding oxidation is $\leq$ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium-water reaction is $\leq$ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and

d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown phase of a LOCA, they do not contribute to the long-term cooling requirements of 10 CFR 50.46.

For both the large- and small-break LOCA analyses, a nominal contained accumulator water volume is used. The contained

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. For small breaks, an increase in water volume is a peak clad temperature penalty. For large breaks, an increase in water volume can be either a peak clad temperature penalty or benefit, depending on downcomer filling and subsequent spill through the break during the core re-flooding portion of the transient. The analysis makes a conservative assumption with respect to ignoring or taking credit for line water volume from the accumulator to the check valve. The safety analysis assumes values of 7627 ~~[6468]~~ gallons and ~~[6879]~~ 8082 gallons. To allow for instrument inaccuracy, values of ~~[6520]~~ 7717 gallons and ~~[6820]~~ 7961 gallons are specified. (2)(3)

The LOCA analysis assumes a value of 585 psig. To account for the accumulator tank design pressure rating, and to allow for instrument accuracy, values of ≥602 psig and ≤661 psig are specified.

The minimum boron concentration setpoint is used in the post-LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post-LOCA environment. Of particular interest is the large-break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent reduction in the available containment sump concentration for post-LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH.

The large- and small-break LOCA analyses are performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit. (4)

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 2 and 4).

The accumulators satisfy Criterion 3 of the NRC Policy Statement.

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four

(6) The maximum nitrogen cover pressure limit prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity.

(continued)

BASES

LCO (continued)

accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spills through the break. If fewer than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 3) could be violated.

For an accumulator to be considered OPERABLE, the isolation valve must be fully open with power removed, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 3) limit of 2200°F.

In MODE 3, with RCS pressure ≤ 1000 psig, and in MODES 4, 5, and 6, the accumulator motor-operated isolation valves are closed to isolate the accumulators from the RCS. This allows RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

ACTIONS

A.1

If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially re-covered core during the early re-flooding phase of a

(continued)

BASES

ACTIONS

A.1 (continued)

large-break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during re-flood. Boiling of ECCS water in the core during re-flood concentrates boron in the saturated liquid that remains in the core. In addition, current analysis techniques demonstrate that the accumulators do not discharge following a large main steam line break for the majority of plants. Even if they do discharge, their impact is minor and not a design limiting event. Thus, 72 hours is allowed to return the boron concentration to within limits.

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 1 hour. In this Condition, the required contents of three accumulators cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 1-hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the potential for exposure of the plant to a LOCA under these conditions.

C.1 and C.2

If the accumulator cannot be returned to OPERABLE status within the associated Completion Time, the ~~plant~~ unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the ~~plant~~ unit must be brought to MODE 3 within 6 hours and pressurizer pressure reduced to ≤ 1000 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required ~~plant~~ unit conditions from full power conditions in an orderly manner and without challenging ~~plant~~ unit systems.

(continued)

BASES

ACTIONS (continued)

D.1

If more than one accumulator is inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Each accumulator valve should be verified to be fully open every 12 hours. This verification ensures that the accumulators are available for injection and ensures timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor-operated valve position should not change with power removed, a closed valve could result in not meeting accident analyses assumptions. This Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned isolation valve is unlikely.

SR 3.5.1.2 and SR 3.5.1.3

Every 12 hours, borated water volume and nitrogen cover pressure are verified for each accumulator. This Frequency is sufficient to ensure adequate injection during a LOCA. Because of the static design of the accumulator, a 12-hour Frequency usually allows the operator to identify changes before limits are reached. Operating experience has shown this Frequency to be appropriate for early detection and correction of off-normal trends.

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator every 31 days since the static design of the accumulators limits the ways in which the concentration can be changed. The 31-day Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage. Sampling within 6 hours after a 1% volume increase will identify whether inleakage has caused a reduction in boron concentration to below the required limit. It is not

6 the affected accumulator

77 gallon

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(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.4 (continued)

necessary to verify boron concentration if the added water inventory is from the refueling water storage tank (RWST), because the water contained in the RWST is within the accumulator boron concentration requirements. This is consistent with the recommendation of NUREG-1366 (Ref. 5).

SR 3.5.1.5

(6) When the pressurizer pressure is ≥ 2000 psig

Verification every 31 days that power is removed from each accumulator isolation valve operator ensures that an active failure could not result in the undetected closure of an accumulator motor-operated isolation valve. If this were to occur, only two accumulators would be available for injection given a single failure coincident with a LOCA. Since power is removed under administrative control, the 31-day Frequency will provide adequate assurance that power is removed.

(6) The allowance that the surveillance is only required when pressurizer pressure is ≥ 2000 psig

~~This SR is modified by a Note that~~ allows power to be supplied to the motor-operated isolation valves when pressurizer pressure is < 2000 psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during plant startups or shutdowns. Even with power supplied to the valves, inadvertent closure is prevented by the RCS pressure interlock associated with the valves.

Should closure of a valve occur in spite of the interlock, the SI signal provided to the valves would open a closed valve in the event of a LOCA.

REFERENCES

1. ~~IEEE Standard 279-1971.~~
2. ~~FSAR, Section [6].~~
3. ~~10 CFR 50.46.~~
4. ~~FSAR, Section [15].~~
5. ~~NUREG-1366, February 1990.~~

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1. IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."

2. WATTS BAR FSAR, Section 6.3, "Emergency Core Cooling System."

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

4. WATTS BAR FSAR, Section 15, "Accident Analysis."

5. NUREG-1366, Improvements to Technical Specifications Surveillance Requirements, (DRAFT).


5

JUSTIFICATIONS FOR DEVIATIONS FROM STS 3.5.1

1. Change to correct error(s) in the STS.

2. Format change to delete brackets that identify plant specific information/values.

3. Change to reflect Watts Bar specific parameter value(s).

4. Supplemental text to describe methodology utilized in determination of minimum/maximum parameter value(s) for Watts Bar.

5. Change to specify Watts Bar specific Reference information and format.

6. This change reflects comments made by the industry to the NRC at the RSTS Proof and Review meeting held in Irvine, CA July 13 - July 20, 1992.

## 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### 3.5.2 ECCS – Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

1 2

-----NOTES-----

1. In MODE 3, both safety injection (SI) pump flow paths may be isolated by closing the isolation valves for up to 2 hours to perform pressure isolation valve (PIV) testing per SR 3.4.14.1, "Reactor Coolant System (RCS) PIV Leakage" testing.

2. Operation in MODE 3 with ECCS pumps declared inoperable pursuant to LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," is allowed for up to 4 hours or until the temperature of one or more RCS cold legs exceeds [375]°F, whichever comes first.

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ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more trains inoperable.<br><br>AND<br><br>6 At least 100% of the ECCS ~~SI~~ flow equivalent to a single OPERABLE ECCS train available. | A.1 Restore train(s) to OPERABLE status. | 72 hours |
| B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3.<br><br>AND<br><br>B.2 Be in MODE 4. | 6 hours<br><br><br>12 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.2.1 | Verify the following valves are in the listed position with power to the valve operator removed.<br><br>Number / Position / Function<br>[ ] [ ] [ ]<br>[ ] [ ] [ ]<br>. . .<br>[ ] [ ] [ ] | 12 hours |
| SR 3.5.2.2 | Verify each ECCS manual, power-operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position. | 31 days |
| SR 3.5.2.3 | Verify ECCS piping is full of water. | 31 days |
| SR 3.5.2.4 | Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head. | In accordance with the Inservice Testing Program |
| SR 3.5.2.5 | Verify each ECCS automatic valve in the flow path actuates to ~~its~~ the correct position on an actual or simulated actuation signal. | [18] months |
| SR 3.5.2.6 | Verify each ECCS pump starts automatically on an actual or simulated actuation signal. | [18] months |

(continued)

| | | |
|---|---|---|
| FCV-63-1 | OPEN | RHR SUPPLY |
| FCV-63-22 | OPEN | SIS DISCHARGE |

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.2.7 | Verify, for each ECCS throttle valve listed below, each position stop is in the correct position. ~~Valve Number~~ ~~[ ]~~ ~~[ ]~~ ~~[ ]~~ | [18] months |
| SR 3.5.2.8 | Verify, by visual inspection, each ECCS train containment sump suction inlet is not restricted by debris and the suction inlet trash racks and screens show no evidence of structural distress or abnormal corrosion. | [18] months |

| CCP Discharge Throttle Valves | SI Cold Leg Throttle Valves | SI Hot Leg Throttle Valves |
|---|---|---|
| 63-582 | 63-550 | 63-542 |
| 63-583 | 63-552 | 63-544 |
| 63-584 | 63-554 | 63-546 |
| 63-585 | 63-556 | 63-548 |

# B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

## B 3.5.2 ECCS – Operating

BASES

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss-of-coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system;

b. Rod ejection accident;

c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater; and

d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

There are three phases of ECCS operation: injection, cold leg recirculation, and hot leg recirculation. In the injection phase, water is taken from the refueling water storage tank (RWST) and injected into the Reactor Coolant System (RCS) through the cold legs. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment sumps have enough water to supply the required net positive suction head to the ECCS pumps, suction is switched to the containment sump for cold leg recirculation. After approximately ~~24~~ 15 hours, the ECCS flow is shifted to the hot leg recirculation phase to provide a backflush, which would reduce the boiling in the top of the core and any resulting boron precipitation.

The ECCS consists of three separate subsystems: centrifugal charging (high head), safety injection (SI) (intermediate head), and residual heat removal (RHR) (low head). Each subsystem consists of two redundant, 100% capacity trains. The ECCS accumulators and the RWST are also part of the

(continued)

BASES

BACKGROUND (continued)

ECCS, but are not considered part of an ECCS flow path as described by this LCO.

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the RWST can be injected into the RCS following the accidents described in this LCO. The major components of each subsystem are the centrifugal charging pumps, the RHR pumps, heat exchangers, and the SI pumps. Each of the three subsystems consists of two 100% capacity trains that are interconnected and redundant such that either train is capable of supplying 100% of the flow required to mitigate the accident consequences. This interconnecting and redundant subsystem design provides the operators with the ability to utilize components from opposite trains to achieve the required 100% flow to the core.

During the injection phase of LOCA recovery, a suction header supplies water from the RWST to the ECCS pumps. Separate piping supplies each subsystem and each train within the subsystem. The discharge from the centrifugal charging pumps combines prior to entering the boron injection tank (BIT) (if the plant utilizes a BIT) and then divides again into four supply lines, each of which feeds the injection line to one RCS cold leg. The discharge from the SI and RHR pumps divides and feeds an injection line to each of the RCS cold legs. ~~Control valves~~ are set to balance the flow to the RCS. This balance ensures sufficient flow to the core to meet the analysis assumptions following a LOCA in one of the RCS cold legs.

1

Throttle valves and piping hydraulic design 4

and prevent pump runout 4

For LOCAs that are too small to depressurize the RCS below the shutoff head of the SI pumps, the centrifugal charging pumps supply water until the RCS pressure decreases below the SI pump shutoff head. During this period, the steam generators are used to provide part of the core cooling function.

During the recirculation phase of LOCA recovery, RHR pump suction is transferred to the containment sump. The RHR pumps then supply the other ECCS pumps. Initially, recirculation is through the same paths as the injection phase. Subsequently, recirculation ~~alternates~~ provides injection ~~between~~ to the hot and cold legs simultaneously 4

(continued)

BASES

BACKGROUND (continued)

The centrifugal charging subsystem of the ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as a main steam line break (MSLB). The limiting design conditions occur when the negative moderator temperature coefficient is highly negative, such as at the end of each cycle.

During low temperature conditions in the RCS, limitations are placed on the maximum number of ECCS pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "~~Low Temperature~~ Cold Overpressure ~~Protection (LTOP)~~ Mitigation System (COMS)," for the basis of these requirements. (4)

The ECCS subsystems are actuated upon receipt of an SI signal. The actuation of safeguard loads is accomplished in a programmed time sequence for a loss of offsite power (4). If offsite power is available, the safeguard loads start immediately. ~~in the programmed sequence~~ If offsite power is not available, the engineered safety feature buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total ES cladding thickness before oxidation; (3)

c. Maximum hydrogen generation from a zirconium-water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core is maintained in a coolable geometry; and

e. Adequate long-term core cooling capability is maintained.

The LCO also limits the potential for a post-trip return to power following an MSLB event and ensures that containment temperature limits are met.

Each ECCS subsystem is taken credit for in a large-break LOCA event at full power (Refs. 3 and 4). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The centrifugal charging pumps and SI pumps are credited in a small-break LOCA event. This event establishes the flow and discharge head at the design point for the centrifugal charging pumps. The SGTR and MSLB events also credit the centrifugal charging pumps. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

a. A large-break LOCA event, with loss of offsite power and a single failure disabling one RHR pump (both EDG trains are assumed to operate due to requirements for modeling full active containment heat removal system operation); and

b. A small-break LOCA event, with a loss of offsite power and a single failure disabling one ECCS train.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and re-floods the core.

The effects on containment mass and energy releases are accounted for in appropriate analyses (Refs. 3 and 4). The LCO ensures that an ECCS train will deliver sufficient water to match boiloff rates soon enough to minimize the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

consequences of the core being uncovered following a large LOCA. It also ensures that the centrifugal charging and SI pumps will deliver sufficient water and boron during a small LOCA to maintain core subcriticality. For smaller LOCAs, the centrifugal charging pump delivers sufficient fluid to maintain RCS inventory. For a small-break LOCA, the steam generators continue to serve as the heat sink, providing part of the required core cooling.

The ECCS trains satisfy Criterion 3 of the NRC Policy Statement.

LCO

In MODES 1, 2, and 3, two independent (and redundant) ECCS trains are required to ensure that sufficient ECCS flow is available, assuming a single failure affecting either train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

In MODES 1, 2, and 3, an ECCS train consists of a centrifugal charging subsystem, an SI subsystem, and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an SI signal and automatically transferring suction to the containment sump.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the four cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to supply its flow to the RCS hot and cold legs.

The flow path for each train must maintain its designed independence to ensure that no single failure can disable both ECCS trains.

APPLICABILITY

In MODES 1, 2, and 3, the ECCS OPERABILITY requirements for the limiting Design Basis Accident, a large-break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident

(continued)

BASES

APPLICABILITY (continued)

analysis does not provide for reduced cooling requirements in the lower MODES. The centrifugal charging pump performance is based on a small-break LOCA, which establishes the pump performance curve and has less dependence on power. The SI pump performance requirements are based on a small-break LOCA. MODE 2 and MODE 3 requirements are bounded by the MODE 1 analysis.

This LCO is only applicable in MODE 3 and above. Below MODE 3, the SI signal setpoint is manually bypassed by operator control, and system functional requirements are relaxed as described in LCO 3.5.3, "ECCS – Shutdown."

As indicated in Note 1, the flow path may be isolated for 2 hours in MODE 3, under controlled conditions, to perform pressure isolation valve testing per SR 3.4.14.1. The flow path is readily restorable from the control room.

1

As indicated in Note 2, Operation in MODE 3 with ECCS trains declared inoperable pursuant to LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," is necessary for plants with an LTOP arming temperature at or near the MODE 3 boundary temperature of 350°F. LCO 3.4.12 requires that certain pumps be rendered inoperable at and below the LTOP arming temperature. When this temperature is at or near the MODE 3 boundary temperature, time is needed to restore the inoperable pumps to OPERABLE status.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "RHR and Coolant Circulation – High Water Level," and LCO 3.9.6, "RHR and Coolant Circulation – Low Water Level."

ACTIONS

A.1

6

With one or more trains inoperable and at least 100% of the ECCS ~~SI~~ flow equivalent to a single OPERABLE ECCS train available, the inoperable components must be returned to OPERABLE status within 72 hours. The 72-hour Completion

(continued)

BASES

ACTIONS

A.1 (continued)

Time is based on an NRC reliability evaluation (Ref. 6) and is a reasonable time for repair of many ECCS components.

An ECCS train is inoperable if it is not capable of delivering design flow to the RCS. Individual components are inoperable if they are not capable of performing their design function or supporting systems are not available.

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. The intent of this condition is to maintain a combination of equipment, such that 100% of the SI flow equivalent to a single OPERABLE ECCS train remains available. This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable. ECCS (6)

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 6) has shown that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

Reference 5 describes situations in which one component, such as an RHR crossover valve, can disable both ECCS trains. With one or more component(s) inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered.

B.1 and B.2

If the inoperable trains cannot be returned to OPERABLE status within the associated Completion Time, the ~~plant~~ Unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the ~~plant~~ Unit must be brought to MODE 3 (3)

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

within 6 hours and MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required ~~plant~~ Unit conditions from full power conditions in an orderly manner and without challenging ~~plant~~ Unit systems. (3)

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Misalignment of these valves could render both ECCS trains inoperable. Securing these valves in position by removal of power or by key-locking the control in the correct position ensures that they cannot change position as a result of an active failure or be inadvertently misaligned. These valves are of the type, described in Reference 5, that can disable the function of both ECCS trains and invalidate the accident analyses. A 12-hour Frequency is considered reasonable in view of other administrative controls that will ensure a mispositioned valve is unlikely.

SR 3.5.2.2

Verifying the correct alignment for manual, power-operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a non-accident position provided the valve will automatically reposition within the proper stroke time. 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. The 31-day Frequency is appropriate because the valves are operated under administrative control, and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.3

With the exception of the operating centrifugal charging pump, the ECCS pumps are normally in a standby, non-operating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. This will also prevent water hammer, pump cavitation, and pumping of non-condensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SI signal or during shutdown cooling. The 31-day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the procedural controls governing system operation.

SR 3.5.2.4

Periodic Surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by Section XI of the American Society of Mechanical Engineers (ASME) Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program, which encompasses Section XI of the ASME Code. Section XI of the ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.5 and SR 3.5.2.6

These Surveillances the demonstrate that each automatic ECCS valve actuates to ~~its~~ required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. The 18-month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.5 and SR 3.5.2.6 (continued)

the Surveillances were performed with the reactor at power. The 18-month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of Engineered Safety Feature Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

SR 3.5.2.7

are secured in a throttled position

Realignment of valves in the flow path on an SI signal is necessary for proper ECCS performance. These valves ~~have stops to allow proper positioning~~ for restricted flow to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow. ~~This Surveillance is not required for plants with flow-limiting orifices.~~ The 18-month Frequency is based on the same reasons as those stated in SR 3.5.2.5 and SR 3.5.2.6.

SR 3.5.2.8

Periodic inspections of the containment sump suction inlet ensure that it is unrestricted and stays in proper operating condition. The 18-month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, on the need to have access to the location, and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency has been found to be sufficient to detect abnormal degradation and is confirmed by operating experience.

REFERENCES

1. ~~10 CFR 50, Appendix A, General Design Criterion 35.~~
2. ~~10 CFR 50.46.~~
3. ~~FSAR, Section [ ].~~
4. ~~FSAR, Section [15], "Accident Analysis."~~

SEE next page insert for Ref. 1-5.

(continued)

BASES

REFERENCES (continued)

5. ~~IE Information Notice No. 87-01.~~ (See below) 5

6. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 35, "Emergency Core Cooling System."

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

3. WATTS BAR FSAR, Section 6.3, "Emergency Core Cooling System."

4. WATTS BAR FSAR, Section 15, "Accident Analyses."

5. IE Information Notice No. 87-01, "RHR Valve Misalignment Causes Degradation of ECCS in PWRs," January 6, 1987.

JUSTIFICATIONS FOR DEVIATIONS FROM STS 3.5.2

1. Change to delete information that does not apply to Watts Bar.

2. Format change to delete brackets that identify plant specific information/values.

3. Change to correct error(s) in the STS.

4. Change to reflect Watts Bar specific information or values.

5. Change to specify Watts Bar specific Reference information and format.

6. This change reflects comments made by the industry to the NRC at the RSTS Proof and Review meeting held in Irvine, CA July 13 - July 20, 1992.

## 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### 3.5.3 ECCS – Shutdown

LCO 3.5.3 One ECCS train shall be OPERABLE.

APPLICABILITY: MODE 4.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Required ECCS residual heat removal (RHR) subsystem inoperable. | A.1 Initiate action to restore required ECCS RHR subsystem to OPERABLE status. | Immediately |
| B. Required ECCS ~~high head~~ subsystem inoperable. | B.1 Restore required ECCS ~~high head~~ subsystem to OPERABLE status. CENTRIFUGAL CHARGING | 1 hour |
| C. Required Action and associated Completion Time of Condition B not met. | C.1 ~~NOTE Only required if one RHR loop is OPERABLE.~~ Be in MODE 5. | 24 hours |

## SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| SR 3.5.3.1 Perform the following Surveillances for all equipment required to be OPERABLE: (4) [SR 3.5.2.1] ~~SR 3.5.2.2~~ [SR 3.5.2.3] SR 3.5.2.4 (1) ~~SR 3.5.2.5~~ ~~SR 3.5.2.6~~ (4) [SR 3.5.2.7] SR 3.5.2.8 | In accordance with applicable SRs |

NOTE
An RHR train may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned to the ECCS mode of operation. (4)

# B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

## B 3.5.3 ECCS – Shutdown

BASES

BACKGROUND

The Background section for Bases 3.5.2, "ECCS – Operating," is applicable to these Bases, with the following modifications.

In MODE 4, the required ECCS train consists of two separate subsystems: centrifugal charging (high head) and residual heat removal (RHR) (low head).

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the refueling water storage tank (RWST) can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2.

APPLICABLE SAFETY ANALYSES

The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section.

Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. It is understood in these reductions that certain automatic safety injection (SI) actuation is not available. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.

Only one train of ECCS is required for MODE 4. This requirement dictates that single failures are not considered during this MODE of operation. The ECCS trains satisfy Criterion 3 of the NRC Policy Statement.

LCO

In MODE 4, one of the two independent (and redundant) ECCS trains is required to be OPERABLE to ensure that sufficient ECCS flow is available to the core following a DBA.

In MODE 4, an ECCS train consists of a centrifugal charging subsystem and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow

(continued)

BASES

LCO (continued)

path capable of taking suction from the RWST and transferring suction to the containment sump.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the four cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to deliver its flow to the RCS hot and cold legs.

APPLICABILITY

In MODES 1, 2, and 3, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2, "ECCS – Operating."

In MODE 4 with RCS temperature below 350°F, one OPERABLE ECCS train is acceptable without single-failure consideration, on the basis of the stable reactivity of the reactor and the limited core cooling requirements.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "RHR and Coolant Circulation – High Water Level," and LCO 3.9.6, "RHR and Coolant Circulation – Low Water Level."

ACTIONS

A.1

With no ECCS RHR subsystem OPERABLE, the plant is not prepared to respond to a loss-of-coolant accident or to continue a cooldown using the RHR pumps and heat exchangers. The Completion Time ~~of 15 minutes~~ to initiate actions that would restore at least one ECCS RHR subsystem to OPERABLE status ensures that prompt action is taken to restore the required cooling capacity. Normally, in MODE 4, reactor decay heat is removed from the RCS by an RHR loop. If no RHR loop is OPERABLE for this function, reactor decay heat must be removed by some alternate method, such as use of the steam generators. The alternate means of heat removal must continue until the inoperable RHR loop components can be

3 Immediate

(continued)

BASES

ACTIONS

A.1 (continued)

restored to operation so that decay-heat removal is continuous.

With both RHR pumps and heat exchangers inoperable, it would be unwise to require the plant to go to MODE 5, where the only available heat removal system is the RHR. Therefore, the appropriate action is to initiate measures to restore one ECCS RHR subsystem and to continue the actions until the subsystem is restored to OPERABLE status.

2

B.1

With no ECCS centrifugal charging ~~high-head~~ subsystem OPERABLE, due to the inoperability of the centrifugal charging pump or flow path from the RWST, the plant is not prepared to provide high-pressure response to Design Basis Events requiring SI. The 1-hour Completion Time to restore at least one ECCS ~~high-head~~ subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity or to initiate actions to place the plant in MODE 5 where an ECCS train is not required.

C.1

When the Required Actions of Condition B cannot be completed within the required Completion Time, a controlled shutdown should be initiated. Twenty-four hours is a reasonable time, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems or operators.

INSERT

3

The Note associated with Required Action C.1 is intended to restrict entry into this Condition to those instances when at least one RHR loop is OPERABLE. The Note also is intended to convey the suspension of further action to reach MODE 5 if, while in Condition C, all RHR loops become inoperable. Should the plant be in Condition A, with no RHR loops OPERABLE, it is not advisable or practical to go to MODE 5. In this situation, the steam generators can be used to maintain MODE 4 until an RHR loop is restored to OPERABLE status. Should the plant be in Condition B only, with an

4

(continued)

INSERT

the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to MODE 5 within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions in an orderly manner and without challenging unit systems.

BASES

ACTIONS

(4)

C.1 (continued)

inoperable centrifugal charging loop, it is possible to reach MODE 5 by using an RHR loop.

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

The applicable Surveillance descriptions from Bases 3.5.2 apply.

REFERENCES

The applicable references from Bases 3.5.2 apply.

(4)

This SR is modified by a Note that allows an RHR train to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. This allows operation in the RHR mode during MODE 4, if necessary.

JUSTIFICATIONS FOR DEVIATIONS FROM STS 3.5.3

1. Format change to delete brackets that identify plant specific information/values.

2. Change to specify Watts Bar specific terminology for clarity.

3. Change to correct error(s) in the STS.

4. This change reflects comments made by the industry to the NRC at the RSTS Proof and Review meeting held in Irvine, CA July 13 - July 20, 1992.

## 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### 3.5.4 Refueling Water Storage Tank (RWST)

LCO 3.5.4 The RWST shall be OPERABLE.

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

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. RWST boron concentration not within limits.<br><br>OR<br><br>RWST borated water temperature not within limits. | A.1 Restore RWST to OPERABLE status. | 8 hours |
| B. RWST inoperable for reasons other than Condition A. | B.1 Restore RWST to OPERABLE status. | 1 hour |
| C. Required Action and associated Completion Time not met. | C.1 Be in MODE 3.<br><br>AND<br><br>C.2 Be in MODE 5. | 6 hours<br><br><br>36 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.4.1 | [Only ③ ----------NOTE---------- Required to be performed when ambient air temperature is < ~~[35]~~ 60 °F or > ~~[100]~~ 105 °F. ---------- ④ ] ① Verify RWST borated water temperature is ≥ ~~[35]~~ 60 °F ④ and ≤ ~~[100]~~ 105 °F. | 24 hours |
| SR 3.5.4.2 | Verify RWST borated water volume is ≥ ~~[466,200~~ 370,000 ④ gallons [( )%]. ① | 7 days |
| SR 3.5.4.3 | Verify RWST boron concentration is ≥ [2,000] ppm and ≤ ~~[2,200]~~ 2,100 ④ ppm. ① | 7 days |

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 Refueling Water Storage Tank (RWST)

BASES

BACKGROUND

The RWST supplies borated water to the Chemical and Volume Control System (CVCS) during abnormal operating conditions, to the refueling pool during refueling, and to the ECCS and the Containment Spray System during accident conditions.

The RWST supplies both trains of the ECCS and the Containment Spray System through ~~separate, redundant~~ A COMMON supply header~~s~~ during the injection phase of a loss-of-coolant accident (LOCA) recovery. ~~A~~ motor-operated isolation valves ARE ~~is~~ provided ~~in each header~~ to isolate the RWST from the ECCS once the system has been transferred to the recirculation mode. The recirculation mode is entered when pump suction is transferred to the containment sump following receipt of the RWST-Low ~~Low (Level 1)~~ signal. Use of a single RWST to supply both trains of the ECCS and Containment Spray System is acceptable since the RWST is a passive component, and passive failures are not required to be assumed to occur coincidentally with Design Basis Events UNTIL AFTER TRANSFER TO THE RECIRCULATION MODE.

(4) (4) (4)

Coincident with Containment Sump Level-High

The switchover from normal operation to the injection phase of ECCS operation requires changing centrifugal charging pump suction from the CVCS volume control tank (VCT) to the RWST through the use of isolation valves. Each set of isolation valves is interlocked so that the VCT isolation valves will begin to close once the RWST isolation valves are fully open. Since the VCT is under pressure, the preferred pump suction will be from the VCT until the tank is isolated. This will result in a delay in obtaining the RWST borated water. The effects of this delay are discussed in the Applicable Safety Analyses section of these Bases.

During normal operation in MODES 1, 2, and 3, the safety injection (SI) and residual heat removal (RHR) pumps are aligned to take suction from the RWST.

The ECCS and Containment Spray System pumps are provided with recirculation lines that ensure each pump can maintain minimum flow requirements when operating at or near shutoff head conditions.

(continued)

BASES

BACKGROUND (continued)

When the suction for the ECCS and Containment Spray System pumps is transferred to the containment sump, the RWST flow paths must be isolated to prevent a release of the containment sump contents to the RWST, which could result in a release of contaminants to the atmosphere and the eventual loss of suction head for the ECCS pumps.

This LCO ensures that:

a. The RWST contains sufficient borated water to support the ECCS during the injection phase;

b. Sufficient water volume exists in the containment sump to support continued operation of the ECCS and Containment Spray System pumps at the time of transfer to the recirculation mode of cooling; and

c. The reactor remains subcritical following a LOCA.

Insufficient water in the RWST could result in insufficient cooling capacity when the transfer to the recirculation mode occurs. Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following the LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the containment.

APPLICABLE SAFETY ANALYSES

During accident conditions, the RWST provides a source of borated water to the ECCS and Containment Spray System pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory, and is a source of negative reactivity for reactor shutdown (Ref. 1). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of Bases B 3.5.2, "ECCS-Operating"; B 3.5.3, "ECCS-Shutdown"; and B 3.6.6, "Containment Spray ~~and Cooling~~ System." These analyses are used to assess changes to the RWST in order to evaluate their effects in relation to the acceptance limits in the analyses.

4

The RWST must also meet volume, boron concentration, and temperature requirements for non-LOCA events. The volume is not an explicit assumption in non-LOCA events since the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

required volume is a small fraction of the available volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is different from the total volume contained since, due to the design of the tank, more water can be contained than can be delivered. The minimum boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability. ~~The importance of its value is small for units with a boron injection tank (BIT) with a high boron concentration. For units with no BIT or reduced BIT boron requirements, the minimum boron concentration limit is an important assumption in ensuring the required shutdown capability.~~ The maximum boron concentration is an explicit assumption in the inadvertent ECCS actuation analysis, although it is typically a non-limiting event and the results are very insensitive to boron concentrations. The maximum temperature is a conservative assumption that minimizes the additional cooling credited from the RWST during the heatup phase of the feedline break event analysis; the minimum is an assumption in both the MSLB and inadvertent ECCS actuation analyses, although the inadvertent ECCS actuation event is typically non-limiting.

The MSLB analysis has considered a delay associated with the interlock between the VCT and RWST isolation valves, and the results show that the departure from nucleate boiling design basis is met. The delay has been established as [27] seconds, with offsite power available, or [37] seconds without offsite power. This response time includes [2] seconds for electronics delay, a [15]-second stroke time for the RWST valves, and a [10]-second stroke time for the VCT valves. ~~Plants with a BIT need not be concerned with the delay since the BIT will supply highly borated water prior to RWST switchover, provided the BIT is between the pumps and the core.~~

For a large-break LOCA analysis, the minimum water volume limit of 370,000 ~~[466,200]~~ gallons and the lower boron concentration limit of [2000] ppm are used to compute the post-LOCA sump boron concentration necessary to assure subcriticality. The large-break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core.

The upper limit on boron concentration of 2100 ~~[2200]~~ ppm is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

from cold leg to hot leg injection is to avoid boron precipitation in the core following the accident.

In the ECCS analysis, the containment spray temperature is assumed to be equal to the RWST lower temperature limit of 60 ~~[35]~~°F. If the lower temperature limit is violated, the containment spray further reduces containment pressure, which decreases the rate at which steam can be vented out the break and increases peak clad temperature. The upper temperature limit of 105 ~~[100]~~°F is used in the small-break LOCA analysis and containment OPERABILITY analysis. Exceeding this temperature will result in a higher peak clad temperature, because there is less heat transfer from the core to the injected water for the small-break LOCA and higher containment pressures due to reduced containment spray cooling capacity. For the containment response following an MSLB, the lower limit on boron concentration and the upper limit on RWST water temperature are used to maximize the total energy release to containment.

1 4

The RWST satisfies Criterion 3 of the NRC Policy Statement.

LCO

The RWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment sump to support ECCS and Containment Spray System pump operation in the recirculation mode.

To be considered OPERABLE, the RWST must meet the water volume, boron concentration, and temperature limits established in the SRs.

APPLICABILITY

In MODES 1, 2, 3, and 4, RWST OPERABILITY requirements are dictated by ECCS and Containment Spray System OPERABILITY requirements. Since both the ECCS and the Containment Spray System must be OPERABLE in MODES 1, 2, 3, and 4, the RWST must also be OPERABLE to support their operation. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled." MODE 6 core cooling

(continued)

BASES

APPLICABILITY (continued)

requirements are addressed by LCO 3.9.5, "Residual Heat Removal and Coolant Circulation—High Water Level," and LCO 3.9.6, "Residual Heat Removal and Coolant Circulation—Low Water Level."

ACTIONS

A.1

With RWST boron concentration or borated water temperature not within limits, they must be returned to within limits within 8 hours. Under these Conditions neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE condition. The 8-hour limit to restore the RWST temperature or boron concentration to within limits was developed considering the time required to change either the boron concentration or temperature and the fact that the contents of the tank are still available for injection.

B.1

With the RWST inoperable for reasons other than Condition A, (e.g., water volume) it must be restored to OPERABLE status within 1 hour.

In this Condition, neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in which the RWST is not required. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this Condition simultaneously affecting redundant trains.

C.1 and C.2

3

If the RWST cannot be returned to unit OPERABLE status within the associated 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~~ unit conditions from full

3

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

(3) power conditions in an orderly manner and without challenging ~~plant~~ Unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

The RWST borated water temperature should be verified every 24 hours to be within the limits assumed in the accident analyses band. This Frequency is sufficient to identify a temperature change that would approach either limit, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that eliminates the requirement to perform this Surveillance when ambient air temperatures are within the operating limits of the RWST. With ambient air temperatures within the band, the RWST temperature should not exceed the limits.

SR 3.5.4.2

The RWST water volume should be verified every 7 days to be above the required minimum level in order to ensure that a sufficient initial supply is available for injection and to support continued ECCS and Containment Spray System pump operation on recirculation. Since the RWST volume is normally stable and is protected by an alarm, a 7-day Frequency is appropriate and has been shown to be acceptable through operating experience.

SR 3.5.4.3

The boron concentration of the RWST should be verified every 7 days to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. Since the RWST volume is normally stable, a 7-day sampling Frequency to verify boron

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.4.3 (continued)

concentration is appropriate and has been shown to be acceptable through operating experience.

REFERENCES

1. ~~FSAR, Section [6] and Section [15].~~

(6) WATTS BAR FSAR, Section 6.3, "Emergency Core Cooling System" and Section 15, "Accident Analyses."

JUSTIFICATIONS FOR DEVIATIONS FROM STS 3.5.4

1. Format change to delete brackets that identify plant specific information/values.

2. Change to specify Watts Bar specific terminology for clarity.

3. Change to correct error(s) in the STS.

4. Change to specify Watts Bar specific parameter values or information.

5. Change to delete information that does not apply to Watts Bar.

6. Change to specify Watts Bar specific Reference information and format.

7. The statement as written is confusing and appears contradictory. Change made to clarify the assumptions in the accident analysis based on discussions with Westinghouse.

## 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### 3.5.5 Seal Injection Flow

1

LCO 3.5.5 Reactor coolant pump seal injection flow shall be ≤ [40] gpm with [centrifugal charging pump discharge header] pressure ≥ ~~[2,480]~~ 2,430 psig and the ~~[charging flow]~~ PRESSURIZER LEVEL control valve full open.

1 2 2

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Seal injection flow not within limit. | A.1 Reduce flow to within limit. | 1 hour |
| | AND | |
| | A.~~2~~ 1 Adjust manual seal injection throttle valves to give a flow within limit with [centrifugal charging pump discharge header] pressure ≥ ~~[2,480]~~ 2,430 psig and the ~~[charging flow]~~ PRESSURIZER LEVEL control valve full open. | 4 hours |
| B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3. | 6 hours |
| | AND | |
| | B.2 Be in MODE 4. | 12 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.5.1 | Not (6) ----------NOTE---------- until<br>~~Required to be performed within~~ 4 hours after the Reactor Coolant System pressure stabilizes at ≥ [2,215 psig and ≤ 2,255 psig]. (1)<br>----------<br>Verify manual seal injection throttle valves are adjusted to give a flow within limit with [centrifugal charging pump discharge header] pressure ≥ ~~[2,480]~~ 2,430 psig and the ~~[charging flow]~~ PRESSURIZER LEVEL control valve full open. (2) (2) | 31 days |

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.5 Seal Injection Flow

BASES

BACKGROUND

3 ~~This LCO is applicable only to those units that utilize the centrifugal charging pumps for safety injection (SI).~~ The function of the seal injection throttle valves during an accident is similar to the function of the ECCS throttle valves in that each restricts flow from the centrifugal charging pump header to the Reactor Coolant System (RCS).

The restriction on reactor coolant pump (RCP) seal injection flow limits the amount of ECCS flow that would be diverted from the injection path following an accident. This limit is based on safety analysis assumptions that are required because RCP seal injection flow is not isolated during SI.

APPLICABLE SAFETY ANALYSES

All ECCS subsystems are taken credit for in the large-break loss-of-coolant accident (LOCA) at full power (Ref. 1). The LOCA analysis establishes the minimum flow for the ECCS pumps. The centrifugal charging pumps are also credited in the small-break LOCA analysis. This analysis establishes the flow and discharge head at the design point for the centrifugal charging pumps. The steam generator tube rupture and main steam line break event analyses also credit the centrifugal charging pumps, but are not limiting in their design. Reference to these analyses is made in assessing changes to the Seal Injection System for evaluation of their effects in relation to the acceptance limits in these analyses.

PRESSURIZER LEVEL 2

2430

This LCO ensures that seal injection flow of ≤ ~~[400]~~ 400 gpm, with centrifugal charging-pump discharge header pressure ≥ ~~[2480]~~ psig and ~~charging flow~~ control valve fully open, will be sufficient for RCP seal integrity but limited so that the ECCS trains will be capable of delivering sufficient water to match boiloff rates soon enough to minimize uncovering of the core following a large LOCA. It also ensures that the centrifugal charging pumps will deliver sufficient water for a small LOCA and sufficient boron to maintain the core subcritical. For smaller LOCAs, the charging pumps alone deliver sufficient fluid to overcome the loss and maintain RCS inventory. Seal

2

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

injection flow satisfies Criterion 2 of the NRC Policy Statement.

LCO

The intent of the LCO limit on seal injection flow is to make sure that flow through the RCP seal water injection line is low enough to ensure that sufficient centrifugal charging pump injection flow is directed to the RCS via the injection points (Ref. 2).

The LCO is not strictly a flow limit, but rather a flow limit based on a flow line resistance. In order to establish the proper flow line resistance, a pressure and flow must be known. The flow line resistance is determined by assuming that the RCS pressure is at normal operating pressure and that the centrifugal charging pump discharge pressure is greater than or equal to the value specified in this LCO. The centrifugal charging pump discharge header pressure remains essentially constant through all the applicable MODES of this LCO. A reduction in RCS pressure would result in more flow being diverted to the RCP seal injection line than at normal operating pressure. The valve settings established at the prescribed centrifugal charging pump discharge header pressure result in a conservative valve position should RCS pressure decrease. The additional modifier of this LCO, the control valve ~~(charging flow for four loop units and air-operated seal injection for three loop units)~~ being fully open, is required since the valve is designed to fail open for the accident condition. With the discharge pressure and control valve position as specified by the LCO, a flow limit is established. It is this flow limit that is used in the accident analyses.

The limit on seal injection flow, combined with the centrifugal charging pump discharge header pressure limit and an open wide condition of the ~~charging flow~~ control valve, must be met to render the ECCS OPERABLE. If these conditions are not met, the ECCS flow will not be as assumed in the accident analyses.

APPLICABILITY

In MODES 1, 2, and 3, the seal injection flow limit is dictated by ECCS flow requirements, which are specified for

(continued)

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 and A.2

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 or to place the plant in a MODE in which this system is not required. The 1-hour Completion Time to restore the seal injection flow is for prompt action that will reduce the flow to within its limit. The adjustment to the flow can be made by either the manual valves or the charging flow control valve. If the initial adjustment to meet the Completion Time of 1 hour for Required Action A.1 is made with the charging flow control valve, the operator has 4 hours from the time the flow is known to be above the limit to correctly position the manual valves to meet the Completion Time of Required Action A.2 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 below its limit or that the plant is promptly placed in a MODE in which seal injection flow is not critical. These times are conservative with respect to the Completion Times of other ECCS LCOs; they are based on operating experience, and are sufficient for taking corrective actions by operations personnel.

B.1 and B.2

When the Required Actions cannot be completed within the required Completion Time, a controlled shutdown must be initiated. The Completion Time of 6 hours for reaching MODE 3 from MODE 1 is a reasonable time for a controlled shutdown, based on operating experience and normal cooldown rates, and does not challenge plant safety systems or operators. Continuing the plant shutdown begun in Required

(continued)

INSERT

the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit 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 unit conditions in an orderly manner and without challenging unit systems.

BASES

ACTIONS

B.1 and B.2 (continued)

~~Action B.1, an additional 6 hours is a reasonable time, based on operating experience and normal cooldown rates, to reach MODE 4, where this LCO is no longer applicable.~~

SURVEILLANCE REQUIREMENTS

SR 3.5.5.1

Verification every 31 days that the manual seal injection throttle valves are adjusted to give a flow within the limit ensures that proper manual seal injection throttle valve position, and hence, proper seal injection flow, is maintained. The Frequency of 31 days is based on engineering judgment and is consistent with other ECCS valve Surveillance Frequencies. The Frequency has proven to be acceptable through operating experience.

6

As Noted, the Surveillance is not required to be performed until

A provision has been added to except SR 3.0.4 for entry into MODE 3. The Note permits entry into MODE 3, since the Surveillance cannot be performed in a lower MODE. The exception is permitted for up to 4 hours after the RCS pressure has stabilized within a ± 20 psig range of normal operating pressure. The RCS pressure requirement is specified since this configuration will produce the required pressure conditions necessary to assure that the manual valves are set correctly. The exception is limited to 4 hours to ensure that the Surveillance is timely.

REFERENCES

1. ~~FSAR, Section [6], and Section [15].~~
2. ~~10 CFR 50.46, 1974.~~

4

1. ~~WATTS BAR~~ SALEM FSAR, Section 6.3, "Emergency Core Cooling System", and Section 15, "Accident Analyses".
2. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Plants," 1974.

JUSTIFICATIONS FOR DEVIATIONS FROM STS 3.5.5

1. Format change to delete brackets that identify plant specific information/values.

2. Change to reflect Watts Bar specific information or parameter values.

3. Change to delete information that does not apply to Watts Bar.

4. Change to specify Watts Bar specific Reference information and format.

5. Change to correct error(s) in the STS.

6. This change reflects comments made by the industry to the NRC at the RSTS Proof and Review meeting held in Irvine, CA July 13 - July 20, 1992.

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## 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### 3.5.6 Boron Injection Tank (BIT)

LCO 3.5.6 The BIT shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. BIT inoperable. | A.1 | Restore BIT to OPERABLE status. | 1 hour |
| B. Required Action and associated Completion Time of Condition A not met. | B.1 | Be in MODE 3. | 6 hours |
| | AND | | |
| | B.2 | Borate to a SHUTDOWN MARGIN equivalent to [1%] $\Delta k/k$ at 200°F. | 6 hours |
| | AND | | |
| | B.3 | Restore BIT to OPERABLE status. | 7 days |
| C. Required Action and associated Completion Time of Condition B not met. | C.1 | Be in MODE 4. | 12 hours |

1

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.6.1 | Verify BIT borated water temperature is ≥ [145]°F. | 24 hours |
| [SR 3.5.6.2 | Verify BIT borated water volume is ≥ [ ] gallons.] | 7 days |
| SR 3.5.6.3 | Verify BIT boron concentration is ≥ [20,000] ppm and ≤ [22,500] ppm. | 7 days |

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.6 Boron Injection Tank (BIT)

BASES

BACKGROUND

The BIT is part of the Boron Injection System, which is the primary means of quickly introducing negative reactivity into the Reactor Coolant System (RCS) on a safety injection (SI) signal.

The main flow path through the Boron Injection System is from the discharge of the centrifugal charging pumps through lines equipped with a flow element and two valves in parallel that open on an SI signal. The valves can be operated from the main control board. The valves and flow elements have main control board indications. Downstream of these valves, the flow enters the BIT (Ref. 1).

The BIT is a stainless steel tank containing concentrated boric acid. Two trains of strip heaters are mounted on the tank to keep the temperature of the boric acid solution above the precipitation point. The strip heaters are controlled by temperature elements located near the bottom of the BIT. The temperature elements also activate High and Low alarms on the main control board. In addition to the strip heaters on the BIT, there is a recirculation system with a heat tracing system, including the piping section between the motor-operated isolation valves, which further ensures that the boric acid stays in solution. The BIT is also equipped with a High-Pressure alarm on the main control board. The entire contents of the BIT are injected when required; thus, the contained and deliverable volumes are the same.

During normal operation, one of the two BIT recirculation pumps takes suction from the boron injection surge tank (BIST) and discharges to the BIT. The solution then returns to the BIST. Normally, one pump is running and one is shut off. On receipt of an SI signal, the running pump shuts off and the air-operated valves close. Flow to the BIT is then supplied from the centrifugal charging pumps. The solution of the BIT is injected into the RCS through the RCS cold legs.

(continued)

1

BASES (continued)

APPLICABLE SAFETY ANALYSES

During a main steam line break (MSLB) or loss-of-coolant accident (LOCA), the BIT provides an immediate source of concentrated boric acid that quickly introduces negative reactivity into the RCS.

The contents of the BIT are not credited for core cooling or immediate boration in the LOCA analysis, but for post-LOCA recovery. The BIT maximum boron concentration of [22,500] ppm is used to determine the minimum time for hot leg recirculation switchover. The minimum boron concentration of [20,000] ppm is used to determine the minimum mixed mean sump boron concentration for post-LOCA shutdown requirements.

For the MSLB analysis, the BIT is the primary mechanism for injecting boron into the core to counteract any positive increases in reactivity caused by an RCS cooldown. The analysis uses the minimum boron concentration of the BIT, which also affects both the departure from nucleate boiling and containment design analyses. Reference to the LOCA and MSLB analyses is used to assess changes to the BIT to evaluate their effect on the acceptance limits contained in these analyses.

The minimum temperature limit of [145]°F for the BIT ensures that the solution does not reach the boric acid precipitation point. The temperature of the solution is monitored and alarmed on the main control board.

The BIT boron concentration limits are established to ensure that the core remains subcritical during post-LOCA recovery. The BIT will counteract any positive increases in reactivity caused by an RCS cooldown.

The BIT minimum water volume limit of [ ] gallons is used to ensure that the appropriate quantity of highly borated water with sufficient negative reactivity is injected into the RCS to shut down the core following an MSLB, to determine the hot leg recirculation switchover time, and to safeguard against boron precipitation.

The BIT satisfies Criteria 2 and 3 of the NRC Policy Statement.

(continued)

1

BASES (continued)

LCO

This LCO establishes the minimum requirements for contained volume, boron concentration, and temperature of the BIT inventory (Ref. 2). This ensures that an adequate supply of borated water is available in the event of a LOCA or MSLB to maintain the reactor subcritical following these accidents.

To be considered OPERABLE, the limits established in the SR for water volume, boron concentration, and temperature must be met.

If the equipment used to verify BIT parameters (temperature, volume, and boron concentration) is determined to be inoperable, then the BIT is also inoperable.

APPLICABILITY

In MODES 1, 2, and 3, the BIT OPERABILITY requirements are consistent with those of LCO 3.5.2, "ECCS – Operating."

In MODES 4, 5, and 6, the respective accidents are less severe, so the BIT is not required in these lower MODES.

ACTIONS

A.1

If the required volume is not present in the BIT, both the hot leg recirculation switchover time analysis and the boron precipitation analysis would not be met. Under these conditions, prompt action must be taken to restore the volume to above its required limit to declare the tank OPERABLE, or the plant must be placed in a MODE in which the BIT is not required.

The BIT boron concentration is considered in the hot leg recirculation switchover time analysis, the boron precipitation analysis, and the reactivity analysis for an MSLB. If the concentration were not within the required limits, these analyses could not be relied on. Under these conditions, prompt action must be taken to restore the concentration to within its required limits, or the plant must be placed in a MODE in which the BIT is not required.

The BIT temperature limit is established to ensure that the solution does not reach the boric acid crystallization point. If the temperature of the solution drops below the

(continued)

1

BASES

ACTIONS

A.1 (continued)

minimum, prompt action must be taken to raise the temperature and declare the tank OPERABLE, or the plant must be placed in a MODE in which the BIT is not required.

The 1-hour Completion Time to restore the BIT to OPERABLE status is consistent with other Completion Times established for loss of a safety function and ensures that the plant will not operate for long periods outside of the safety analyses.

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

When Required Action A.1 cannot be completed within the required Completion Time, a controlled shutdown should be initiated. Six hours is a reasonable time, based on operating experience, to reach MODE 3 from full power conditions and to be borated to the required SDM without challenging plant systems or operators. Borating to the required SDM assures that the plant is in a safe condition, without need for any additional boration.

After determining that the BIT is inoperable and the Required Actions of B.1 and B.2 have been completed, the tank must be returned to OPERABLE status within 7 days. These actions ensure that the plant will not be operated with an inoperable BIT for a lengthy period of time. It should be noted, however, that changes to applicable MODES cannot be made until the BIT is restored to OPERABLE status pursuant to the provisions of LCO 3.0.4.

C.1

Even though the RCS has been borated to a safe and stable condition as a result of Required Action B.2, either the BIT must be restored to OPERABLE status (Required Action C.1) or the plant must be placed in a condition in which the BIT is not required (MODE 4). The 12-hour Completion Time to reach MODE 4 is reasonable, based on operating experience and normal cooldown rates, and does not challenge plant safety systems or operators.

(continued)

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.5.6.1

Verification every 24 hours that the BIT water temperature is at or above the specified minimum temperature is frequent enough to identify a temperature change that would approach the acceptable limit. The solution temperature is also monitored by an alarm that provides further assurance of protection against low temperature. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.6.2

Verification every 7 days that the BIT contained volume is above the required limit is frequent enough to assure that this volume will be available for quick injection into the RCS. If the volume is too low, the BIT would not provide enough borated water to ensure subcriticality during recirculation or to shut down the core following an MSLB. Since the BIT volume is normally stable, a 7-day Frequency is appropriate and has been shown to be acceptable through operating experience.

SR 3.5.6.3

Verification every 7 days that the boron concentration of the BIT is within the required band ensures that the reactor remains subcritical following a LOCA; it limits return to power following an MSLB, and maintains the resulting sump pH in an acceptable range so that boron precipitation will not occur in the core. In addition, the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized.

The BIT is in a recirculation loop that provides continuous circulation of the boric acid solution through the BIT and the boric acid tank (BAT). There are a number of points along the recirculation loop where local samples can be taken. The actual location used to take a sample of the solution is specified in the plant Surveillance procedures. Sampling from the BAT to verify the concentration of the BIT is not recommended, since this sample may not be homogenous and the boron concentration of the two tanks may differ.

(continued)

1

BASES (continued)

| | |
|---|---|
| SURVEILLANCE REQUIREMENTS (continued) | The sample should be taken from the BIT or from a point in the flow path of the BIT recirculation loop. |
| REFERENCES | 1. FSAR, Section [6] and Section [15].<br>2. 10 CFR 50.46. |

JUSTIFICATIONS FOR DEVIATIONS FROM STS 3.5.6

1. Change to delete entire specification since it does not apply to Watts Bar.

NOTE: This copy of the Westinghouse Standard Technical Specifications (NUREG-1431, Draft) is based on a second round draft (May 1992) which addressed industry comments on the NUREG submitted in July 1991. The editorial review by the industry and the NRC (June 1992) resulted in additional changes which are denoted by a greek delta symbol. The Proof & Review copy of the RSTS (July 1992) was not available in time to for inclusion into the Watts Bar markup, however, there should be only minor editorial differences between the editorial review copy (June 1992) and the Proof & Review copy (July 1992) of the RSTS.

Watts Bar Nuclear

Justification Key
Chapter 3.6 - Containment Systems

Generic Justification Symbols applicable for the entire section:

△ ---- Large Triangle at the beginning of a Basis Section or Specification indicates that unless otherwise noted the revisions are a result of the Post-Column 6 MEREX meeting generic revisions.

△ ---- Small Triangle indicates that a particular mark-up is a result of the Post-Column 6 MEREX meeting generic revisions.

(R) ---- Watts Bar prefers to use an expanded reference section format.

(E) ---- Indicates an editorial change.

(P) ---- Indicates a Plant Specific Parameter or Value

(T) ---- Indicates Watts Bar Specific Terminology or information.

(W) ---- Change reflects comments made by the industry to the NRC at the RSTS Proof and Review meeting held in Irvine, CA during July 1992.

Specific Justifications

Section 3.6.1

(1) - Watts Bar shield building releases during accident conditions are through a filtration unit.

(2) - Change clarifies that the $P_a$ utilized at Watts Bar (15.0 psig) actually exceeds calculated peak containment internal pressure (11.21 psig) following a DBA.

Section 3.6.2

(1) - Watts Bar ice condenser containment design does not allow access to an airlock from the second airlock via containment.

(2) - Change clarifies that the $P_a$ utilized at Watts Bar (15.0 psig) actually exceeds calculated peak containment internal pressure (11.21 psig) following a DBA.

Section 3.6.3

1 - The Shutdown Purge System and Mini-Purge System functions are accomplished by the Watts Bar Reactor Building Purge Ventilation System.

2 - The 24 inch purge valves at Watts Bar are the block-type valves, consequently, a Surveillance is required to monitor the valves ability to be physically restricted to $\leq$ 50 degrees. Although, at the Post-column 6 meeting a surveillance was added for this function, the Watts Bar comments received indicated that this surveillance was to be slightly revised and left in its original location.

3 - Because the purge valve blockage verification was to be kept in SR 3.6.3.1, this surveillance would be redundant and consequently, is not required.

4 - Watts Bar utilizes three sizes of containment purge valves to which this SR applies.

5 - Change for editorial clarity.

6 - All of Watts Bar's purge valves are capable of closing during any DBA.

7 - Watts Bar utilizes an ice condenser containment.

Section 3.6.4 - No Additional Justifications Required

Section 3.6.5 - No Additional Justifications Required

Section 3.6.6

1 - Based on Data Base comment - GDC 50 was to be added to this section.

2 - Not Applicable based on Watts Bar Plant Specific configuration - Watts Bar does not have a Spray Additive System

3 - Not Applicable based on Watts Bar Plant Specific configuration.

4 - Watts Bar plant configuration requires that one train of the Containment Spray System as well as one RHR Spray Train be operable to provide heat removal capability during a Design Basis Accident.

5 - Per the Watts Bar Technical Specification Interpretation Manual Applicability Review of MERITS Implementation - this statement is to be added.

(6) - Change for editorial clarity.

Section 3.6.7

(1) - The alternate hydrogen control system at Watts Bar is the Hydrogen Mitigation System (HMS).

Section 3.6.8

(1) - Data Base Item from Watts Bar which was to be added with the accompanying reference.

Section 3.6.9

(1) - Additional verbiage added in the WBN specification found to be acceptable.

(2) - Based on Watts Bar Comments to the Phase I mark-up it was determined that verifying that the shield building annulus pressure is within limits was to be remain in the EGTS Section 3.6.9) -- therefore it was removed from Section 3.6.15.

(3) - Verbiage different from the standard which included a discussion of the annulus pressure and was found to be acceptable.

(4) - Different wording, same meaning - found to be acceptable.

(5) - Watts Bar does not have a Containment Spray System.

(6) - Change for editorial clarity.

(7) - The EGTS filter bypass damper is not a tech spec required feature at Watts BAR.

(8) - Since the shield building annulus is maintained at a vacuum during normal power and accident conditions, a time limit to draw the annulus down to vacuum conditions does not apply.

Section 3.6.10

(1) - The statements that have been removed are based on the Watts Bar Plant Specific configuration - no hydrogen skimmer system. The added statements are describing the ARS more completely.

(2) - Revised based on Watts Bar plant specific configuration.

(3) - Watts Bar additional verbiage found to be acceptable.

(4) - Watts Bar utilizes direct drive ARS fans with only one fan speed.

(5) - Watts Bar does not utilize containment vacuum relief valves. Alignment of the containment purge isolation valves fulfills this vacuum relief function.

(6) - Watts Bar safety analysis does not assume inadvertent simultaneous actuation of both the ARS and the Containment Spray System.

Section 3.6.11

(1) - Change for editorial clarity.

(2) - Modification of SR due to overhead obstructions above ice baskets.

Section 3.6.12

(1) - Based on Watts Bar's Phase I comments, it has been determined that verification via visual inspection of each top deck door for free movement of the top deck vent assembly will be performed. SR 3.6.12.7

(2) - Change for editorial clarity.

Section 3.6.13 -

(1) - Watts Bar does not perform an elongation test in its periodic testing of the divider barrier seal coupons.

(2) - Additional statement provides clarification information only.

Section 3.6.14 - No Additional Justifications Required

Section 3.6.15

(1) - Based on Watts Bar Comments to the Phase I mark-up it was determined that verifying that the shield building annulus pressure is within limits was to be remain in the EGTS Section 3.6.9) -- therefore it was removed from Section 3.6.15.

(2) - Watts Bar has only one door per access opening.

DRAFT

## 3.6 CONTAINMENT SYSTEMS

### 3.6.1 Containment ~~(Atmospheric, Subatmospheric, Ice Condenser, and Dual)~~ (E)

LCO 3.6.1 Containment shall be OPERABLE.

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

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| ~~A. Secondary containment bypass leakage not within limit.~~ | ~~A.1 Restore leakage within limit.~~ | ~~4 hours~~ |
| ~~B.~~ A. Containment inoperable ~~[for reasons other than Condition A.]~~ | ~~B.1~~ A. Restore containment to OPERABLE status. | 1 hour |
| ~~C.~~ B. Required Action and associated Completion Time not met. | ~~C.1~~ B. Be in MODE 3.<br>AND<br>~~C.2~~ B. Be in MODE 5. | 6 hours<br><br>36 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.6.1.1 | Perform required visual examinations and leakage rate testing except for containment air lock, ~~[and purge valve with resilient seal]~~ testing, in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, ~~[as contained in the Containment Leakage Rate Test Program]~~. The maximum allowable leakage rate, $L_a$, is *[.25]% of containment air weight per day at the calculated peak containment pressure, $P_a$. | -----NOTE------ SR 3.0.2 is not applicable ---------- In accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, [as contained in the Containment Leakage Rate Test Program] |
| SR 3.6.1.2 | Verify containment structural integrity in accordance with the Containment Tendon Surveillance Program. | In accordance with the Containment Tendon Surveillance Program |
| SR 3.6.1.3 | ----------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 [as contained in the Containment Leakage Rate Test Program]. ---------- Verify the combined leakage rate for all secondary containment bypass leakage is ≤ [ $L_a$] when pressurized to ≥ [ psig]. | [18] months |

~~shield building bypass leakage path~~ (W)

SPACE (P)

N/A