B 3.0 TECHNICAL REQUIREMENT (TR) AND TECHNICAL SURVEILLANCE REQUIREMENT (TSR) APPLICABILITY

BASES

TRs

TR 3.0.1 through TR 3.0.6 establish the general requirements applicable to all Requirements and apply at all times, unless otherwise stated.

TR 3.0.1

TR 3.0.1 establishes the Applicability statement within each individual Requirement as the requirements for when the TR is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Requirement).

TR 3.0.2

TR 3.0.2 establishes that upon discovery of a failure to meet a TR, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of a TR are not met. This Requirement establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Requirement; and

b. Completion of the Required Actions is not required when a TR is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the TR must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Requirement is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

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BASES

TRs

TR 3.0.2 (continued)

Completing the Required Actions is not required when a TR is met or is no longer applicable, unless otherwise stated in the individual Technical Requirements.

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual TR's ACTIONS specify the Required Actions where this is the case. An example of this is in TR 3.4.2, "Pressurizer Temperature Limits."

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Alternatives that would not result in redundant equipment being inoperable should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time other conditions exist which result in TR 3.0.3 being entered. Individual Requirements may specify a time limit for performing a TSR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Requirement becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Requirement becomes applicable, and the ACTIONS Condition(s) are entered.

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BASES

TRs (continued)

TR 3.0.3

TR 3.0.3 establishes the actions that must be implemented when a TR is not met and:

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

b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering TR 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that TR 3.0.3 be entered immediately.

This Requirement delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the TR and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering TR 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Requirement applies. The use and interpretation of specified times to complete the actions of TR 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

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BASES

TRs

TR 3.0.3 (continued)

A unit shutdown required in accordance with TR 3.0.3 may be terminated and TR 3.0.3 exited if any of the following occurs:

a. The TR is now met.

b. A Condition exists for which the Required Actions have now been performed.

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time TR 3.0.3 is exited.

The time limits of TR 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, TR 3.0.3 provides actions for Conditions not covered in other Requirements. The requirements of TR 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by TR 3.0.3. The requirements of TR 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Requirements sufficiently define the remedial measures to be taken.

Exceptions to TR 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with TR 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in TR 3.3.4, "Seismic Instrumentation". TR 3.3.4 has an Applicability of "At all times". Therefore, this TR can be applicable in any or all MODES. If the TR and the Required Actions of TR 3.3.4 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Actions are the appropriate Required Actions to complete in lieu of the actions of TR 3.0.3. These exceptions are addressed in the individual Requirements.

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BASES

TRs (continued)

TR 3.0.4

TR 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when a TR is not met. It allows placing the unit in a MODE or other specified condition stated in that Applicability (e.g., the Applicability desired to be entered) when unit conditions are such that the requirements of the TR would not be met, in accordance with TR 3.0.4.a, TR 3.0.4.b, or TR 3.0.4.c.

TR 3.0.4.a allows entry into a MODE or other specified condition in the Applicability with the TR not met when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions.

TR 3.0.4.b allows entry into a MODE or other specified condition in the Applicability with the TR not met after performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate.

The risk assessment may use quantitative, qualitative, or blended approaches, and the risk assessment will be conducted using the plant program, procedures, and criteria in place to implement 10 CFR 50.65(a)(4), which requires that risk impacts of maintenance activities be assessed and managed. The risk assessment, for the purposes of TR 3.0.4.b, must take into account all inoperable Technical Specification or TRM equipment regardless of whether the equipment is included in the normal 10 CFR 50.65(a)(4) risk assessment scope. The risk assessments will be conducted using the procedures and guidance endorsed by Regulatory Guide 1.182, “Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.” Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, “Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.” These documents address general guidance for conduct of the risk assessment, quantitative and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to plan and conduct other activities in a manner that controls overall risk, increased risk awareness by shift and

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BASES

TRs

TR 3.0.4 (continued)

management personnel, actions to reduce the duration of the condition, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed MODE change is acceptable. Consideration should also be given to the probability of completing restoration such that the requirements of the TR would be met prior to the expiration of ACTIONS Completion Times that would require exiting the Applicability.

TR 3.0.4.b may be used with single, or multiple systems and components unavailable. NUMARC 93-01 provides guidance relative to consideration of simultaneous unavailability of multiple systems and components.

The results of the risk assessment shall be considered in determining the acceptability of entering the MODE or other specified condition in the Applicability, and any corresponding risk management actions. The TR 3.0.4.b risk assessments do not have to be documented.

TRs allow continued operation with equipment unavailable in MODE 1 for the duration of the Completion Time. Since this is allowable, and since in general the risk impact in that particular MODE bounds the risk of transitioning into and through the applicable MODES or other specified conditions in the Applicability of the TR, the use of the TR 3.0.4.b allowance should be generally acceptable, as long as the risk is assessed and managed as stated above. However, a small subset of systems and components may be determined to be more important to risk and therefore, the use of the TR 3.0.4.b allowance will be prohibited. The TRs governing these system and components will contain Notes prohibiting the use of TR 3.0.4.b by stating that TR 3.0.4.b is not applicable.

TR 3.0.5

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

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

b. The OPERABILITY of other equipment.

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BASES

TRs

TR 3.0.5 (continued)

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

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

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

TR 3.0.6

TR 3.0.6 establishes an exception to TR 3.0.2 for support systems that have a TR specified in the Technical Requirements. This exception is provided because TR 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system TR be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system TR's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is a TR specified for it in the Technical Requirements, the supported system(s) is required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported system’s Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' TRs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

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BASES

TRs

TR 3.0.6 (continued)

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with TR 3.0.2.

Technical Specification 5.7.2.18, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into TR 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of TR 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the TR in which the loss of safety function exists are required to be entered.

BASES

TSRs

TSR 3.0.1 through TSR 3.0.4 establish the general requirements applicable to all Technical Surveillance Requirements and apply at all times, unless otherwise stated.

TSR 3.0.1

TSR 3.0.1 establishes the requirement that TSRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the TR apply, unless otherwise specified in the individual TSRs. This Requirement is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with TSR 3.0.2, constitutes a failure to meet a TR.

Systems and components are assumed to be OPERABLE when the associated TSRs have been met. Nothing in this Requirement, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the TSRs; or

b. The requirements of the Surveillance(s) are known not to be met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated TR are not applicable, unless otherwise specified.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with TSR 3.0.2, prior to returning equipment to OPERABLE status.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with TSR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be

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BASES

TSRs

TSR 3.0.1 (continued)

incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

TSR 3.0.2

TSR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval.

TSR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the TSRs. The exceptions to TSR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Requirements. An example of where TSR 3.0.2 does not apply is a Surveillance with a Frequency of "in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions." The requirements of regulations take precedence over the TRs. The TRs cannot in and of themselves extend a test interval specified in the regulations. Therefore, there is a Note in the Frequency stating, "TSR 3.0.2 is not applicable."

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

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BASES

TSRs

TSR 3.0.2 (continued)

The provisions of TSR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified, with the exception of surveillance required to be performed on a 31-day frequency. For surveillance performed in a 31-day frequency, the normal surveillance interval may be extended in accordance with Technical Requirement 3.0.2 cyclically as required to remain synchronized to the 13-week maintenance work schedules. This practice is acceptable based on the results of an evaluation of 31-day frequency surveillance test histories that demonstrate that no adverse failure rate changes have occurred nor would be expected to develop as a result of cyclical use of surveillance interval extensions and the fact that the total number of 31-day frequency surveillances performed in any one-year period remains unchanged.

TSR 3.0.3

TSR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with TSR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

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

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TSR 3.0.3 (continued)

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, TSR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

TSR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

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BASES

TSRs

TSR 3.0.3 (continued)

Failure to comply with specified Frequencies for TSRs is expected to be an infrequent occurrence. Use of the delay period established by TSR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable TR Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable TR Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Requirement, or within the Completion Time of the ACTIONS, restores compliance with TSR 3.0.1.

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BASES

TSRs (continued)

TSR 3.0.4

TSR 3.0.4 establishes the requirement that all applicable TSRs must be met before entry into a MODE or other specified condition in the Applicability.

This TSR ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit. The provisions of the TR should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

A provision is included to allow entry into a MODE or other specified condition in the Applicability when a TR is not met due to Surveillance not being met in accordance with TR 3.0.4.

However, in certain circumstances, failing to meet a TSR will not result in TSR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated TSR(s) are not required to be performed, per TSR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, TSR 3.0.4 does not apply to the associated TSR(s) since the requirement for the TSR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in a TSR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the TR is not met in this instance, TR 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes. TSR 3.0.4 does not restrict changing MODES or other specified conditions of the Applicability when a TSR has not been performed within the specified Frequency, provided the requirement to declare the TR not met has been delayed in accordance with TSR 3.0.3.

The provisions of TSR 3.0.4 shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of TSR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

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BASES

TSRs

TSR 3.0.4 (continued)

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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 Boration Systems Flow Paths, Shutdown

BASES

BACKGROUND

The boration injection system is a subsystem of the Chemical and Volume Control System (CVCS). The CVCS regulates the concentration of chemical neutron absorber (boron) in the reactor coolant to control reactivity changes. The boration system ensures that negative reactivity control is available during each mode of facility operation. The amount of boric acid stored in the borated water sources always exceeds the amount required to borate the Reactor Coolant System (RCS) to cold shutdown concentration assuming that the control assembly with the highest reactivity worth is stuck in its fully withdrawn position. This amount of boric acid also exceeds the amount required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay.

The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators. The boration system Technical Requirements place limitations on the contained water volume, boron concentration, and temperature of both the Refueling Water Storage Tank (RWST) and Boric Acid Storage System. For MODES 4, 5 and 6, the boron capability is necessary to provide a sufficient SDM to compensate for xenon decay and cooldown from 350°F to 140°F. For MODES 1, 2, and 3, the boron capability is necessary to provide a sufficient SDM to compensate for xenon decay and cooldown to 200°F.

During reactor operation, changes are made in the reactor coolant boron concentration for the following conditions:

1. Reactor Startup - boron concentration must be decreased from shutdown concentration to achieve criticality.

2. Load Follow - boron concentration must be either increased or decreased to compensate for the xenon transient following a change in load.

3. Fuel Burnup - boron concentration must be decreased to compensate for fuel burnup and the buildup of fission products in the fuel.

4. Cold Shutdown - boron concentration must be increased to the cold shutdown concentration.

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BASES

BACKGROUND (continued)

Boric acid is stored in three boric acid tanks. Two boric acid transfer pumps are provided for each unit with one pump normally aligned with one boric acid tank and continuously running at low speed to provide recirculation for the boric acid system and the boric acid tank. On a demand signal by the reactor makeup control system, the boric acid transfer pumps are shifted to high speed and the pump aligned to the makeup system delivers boric acid to the suction header of the charging pumps (Ref. 1).

APPLICABLE SAFETY ANALYSES

The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the CVCS, which causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system and/or stop the primary water pumps. This action is required before the shutdown margin is lost. Operation of the boration subsystem is not assumed to mitigate this event (Ref. 2 & 4). OPERABILITY of the charging pumps, the RWST, and the appropriate flow paths is required as part of the Emergency Core Cooling System (ECCS). The Technical Specifications for the ECCS address the requirements of these components.

TR

TR 3.1.1 requires at least one boron injection flow path to be OPERABLE and capable of being powered from an OPERABLE emergency power source during MODES 4, 5, and 6 in order to provide a path to accomplish (1) normal makeup, (2) chemical shim reactivity control, and (3) miscellaneous fill and transfer operations. This requirement may be achieved by meeting one of the following two conditions:

a. A flow path from an OPERABLE boric acid storage tank, through the boric acid transfer pump, through a charging pump to the RCS, or

b. A flow path from an OPERABLE RWST through a charging pump to the RCS.

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BASES (continued)

APPLICABILITY

The OPERABILITY of one boron injection flow path ensures that this system is available for reactivity control while in MODES 4, 5, and 6. The APPLICABILITY statement is modified by the following Note to ensure the restrictions imposed by Technical Specification LCO 3.0.4.b are considered:

> For Mode 4, Technical Specification LCO 3.0.4.b is not applicable to ECCS high head (centrifugal charging) subsystem.

Boron injection flow paths for MODES 1, 2, and 3 are covered in Technical Requirement 3.1.2, "Boration Systems Flow Paths, Operating".

ACTIONS

A.1 and A.2

With the Boration Systems flow path OPERABILITY requirements not met, or the Boration Systems flow path not capable of being powered by an OPERABLE emergency power source, the plant must be placed in a condition where negative reactivity addition is not required. This is accomplished by suspending all CORE ALTERATIONS and positive reactivity additions immediately. One boron injection flow path is required to meet the TR and to ensure that negative reactivity control is available during MODES 4, 5, and 6. Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition.

The immediate Completion Time is consistent with the required times for actions to be performed without delay and in a controlled manner.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.1.1.1

This surveillance verifies the temperature of the areas containing portions of the flow path from the boric acid tanks is ≥~~>~~ 63°F (value does ~~not~~ account for instrument error) (Ref. 3). This ensures that the high concentration of boric acid in the storage tanks is not allowed to precipitate due to cooling.

The Surveillance is modified by a note stating that the surveillance is required only if a flow path from the boric acid storage tanks is required OPERABLE. The 12 hour Frequency is based on engineering experience and is reasonable considering the time required for performing the surveillance and the probability for changes in the area temperatures.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.1.1.2

Verifying the correct alignment for manual, power operated, and automatic valves in the Boration System flow path provides assurance that the proper flow paths exist for Boration System operation. This TSR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This TSR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This TSR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

REFERENCES

1. Watts Bar FSAR, Section 9.3.4, “Chemical and Volume Control System.”

2. WCAP-11618, “MERITS Program-Phase II, Task 5, Criteria Application,” including Addendum 1 dated April, 1989, as clarified by Reference 4.

3. CEN-603, “Boric Acid Concentration Reduction Effort, Technical Bases and Operational Analysis for Watts Bar Nuclear Plant, Unit 1, Revision 00, April 1993.”

4. TVA letter, “Watts Bar Nuclear Plant (WBN) Unit 1 - Technical Requirements Manual (TRM),” dated August 27, 1992 (ADAMS Accession No. ML073230174) including Enclosure 1, “Watts Bar Technical Requirements Manual,” (ADAMS Accession No. ML073620391).

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Boration Systems Flow Paths, Operating

BASES

BACKGROUND	A description of the Boration Systems Flow Paths is provided in the Bases for Technical Requirement 3.1.1, "Boration Systems Flow Paths, Shutdown."
APPLICABLE SAFETY ANALYSES	The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or Transient. In the case of a malfunction of the Chemical and Volume Control System, which causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system and/or stop the primary water pumps. This action is required before the shutdown margin is lost. Operation of the boration subsystem is not assumed to mitigate this event (Refs. 1 & 3). OPERABILITY of the charging pumps, the Refueling Water Storage Tank (RWST), and the appropriate flow paths is required as part of the Emergency Core Cooling System (ECCS). The Technical Specifications for the ECCS address the requirements of these components.
TR	TR 3.1.2 requires at least two boron injection flow paths to be OPERABLE during MODES 1, 2, and 3, in order to provide two redundant paths to accomplish (1) normal makeup, (2) chemical shim reactivity control, and (3) miscellaneous fill and transfer operations. This requirement may be achieved by having two of the following three flow paths OPERABLE: a. One flow path from the boric acid storage tanks, through a boric acid transfer pump, through a charging pump to the Reactor Coolant System (RCS). b. Two flow paths from the RWST, through a charging pump to the RCS.

(continued)

BASES (continued)

APPLICABILITY

The OPERABILITY of two boron injection flow paths ensures that this system is available for reactivity control while in MODES 1, 2, and 3. Two flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable.

Boron injection flow paths for MODES 4, 5, and 6 are covered in Technical Requirement 3.1.1, "Boration Systems Flow Paths, Shutdown."

ACTIONS

A.1

If one of the required boron injection flow paths is inoperable, action must be taken to restore the required flow path to OPERABLE status. The 72 hour Completion Time was developed taking into account the redundant capabilities afforded by the OPERABLE flow path and reasonable time for repairs. The Completion Time is consistent with the time allowed to restore an ECCS train to OPERABLE status (See Technical Specification 3.5.2, "ECCS - Operating.").

A.2.1, A.2.2, and A.2.3

An alternative to Required Action A.1 is to place the plant in MODE 3 and borate to a SDM equivalent to $\geq$ 1% Δk/k at 200°F within 78 hours, and restore the required flow path to OPERABLE status within 246 hours. This precludes the need for a flow path for load follow and fuel burnup compensation, allowing the additional 7 days to restore two flow paths to OPERABLE status. An additional 6 hours (78 hours total) are allowed to reach MODE 3 from full power in an orderly manner and without challenging plant systems. The allowed Completion Time to reach MODE 3 is reasonable, based on operating experience.

B.1

If the required flow path cannot be restored to OPERABLE status or the Required Actions of Condition A are not met within the associated Completion Times, the unit must be placed in a MODE in which the TR does not apply. This is done by placing the unit in at least MODE 4 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, to reach required plant conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.1.2.1

This surveillance verifies the temperature of the required flow path from the boric acid tanks to be at least 63°F (value does ~~not~~ account for instrument error). This ensures that the high concentration of boric acid in the storage tanks is not allowed to precipitate due to cooling (Ref. 2).

The surveillance is modified by a note stating that the surveillance is required only if the flow path from the boric acid storage tanks is used as one of the two required flow paths. The 12 hour Frequency is based on engineering experience and is reasonable considering the time required for performing the surveillance and the probability for changes in the area temperatures.

TSR 3.1.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the Boration System flow path provides assurance that the proper flow paths exist for Boration System operation. This TSR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This TSR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This TSR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

TSR 3.1.2.3

This surveillance demonstrates that each automatic valve in the flow path actuates to its required position on an actual or simulated actuation signal. The 18 month Frequency was developed considering it is prudent that this surveillance only be performed during a plant outage. This is due to the plant conditions needed to perform the TSR and the potential for unplanned plant transients if the TSR is performed with the reactor at power.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.1.2.4

Verification that the flow path from the boric acid tanks delivers at least 35 gpm (value does not account for instrument error) to the RCS demonstrates that gross degradation of the boric acid transfer pumps, crystallization of boric acid in the system, and other hydraulic component problems have not occurred (Ref. 2).

The 18 month Frequency was developed considering it is prudent that this surveillance only be performed during a plant outage. This is due to the plant conditions needed to perform the TSR and the potential for unplanned plant transients if the TSR is performed with the reactor at power.

REFERENCES

1. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989, as clarified by Reference 3.

2. CEN-603, “Boric Acid Concentration Reduction Effort, Technical Bases and Operational Analysis For Watts Bar Nuclear Plant, Unit 1,” Revision 00, April 1993

3. TVA letter, “Watts Bar Nuclear Plant (WBN) Unit 1 - Technical Requirements Manual (TRM),” dated April 27, 1992 (ADAMS Accession No. ML073230174) including Enclosure 1, “Watts Bar Technical Requirements Manual,” (ADAMS Accession No. ML073620391).

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Charging Pump, Shutdown

BASES

BACKGROUND	A description of the Boration Systems Flow Paths, which include charging pumps, is provided in the Bases for Technical Requirement 3.1.1, "Boration Systems Flow Paths, Shutdown."
APPLICABLE SAFETY ANALYSES	The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or Transient. In the case of a malfunction of the Chemical and Volume Control System, which causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system and/or stop the primary water pumps. This action is required before the SDM is lost. Operation of the boration subsystem is not assumed to mitigate this event (Ref. 1 & 3). OPERABILITY of the charging pumps, the refueling water storage tank, and the appropriate flow paths is required as part of the Emergency Core Cooling System (ECCS). The Technical Specifications for the ECCS address the requirements of these components. Technical Specification 3.4.12, "Cold Overpressure Mitigation System," places restrictions on maximum number of charging pumps allowed OPERABLE for overpressure concerns.
TR	TR 3.1.3 requires one charging pump in the required boron injection flow path to be OPERABLE and capable of being powered from an OPERABLE emergency power source during MODES 4, 5, and 6 in order to provide the driving force to accomplish (1) normal makeup, (2) chemical shim reactivity control, and (3) miscellaneous fill and transfer operations.

(continued)

BASES (continued)

APPLICABILITY

The OPERABILITY of one charging pump in the required boron injection flow path ensures that this system is available for reactivity control while in MODES 4, 5, and 6. The APPLICABILITY statement is modified by the following Note to ensure the restrictions imposed by Technical Specification LCO 3.0.4.b are considered:

> For Mode 4, Technical Specification LCO 3.0.4.b is not applicable to ECCS high head (centrifugal charging) subsystem.

Charging pump OPERABILITY requirements for MODES 1, 2, and 3 are covered in Technical Requirement 3.1.4, "Charging Pumps, Operating."

ACTIONS

A.1 and A.2

With the required charging pump inoperable or not capable of being powered by an OPERABLE emergency power source, the plant must be placed in a condition where negative reactivity addition is not required. This is accomplished by suspending all CORE ALTERATIONS and positive reactivity additions immediately. One OPERABLE charging pump in the required boron injection flow path is required to meet the TR and to ensure that negative reactivity control is available during Modes 4, 5, and 6. Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.1.3.1

Periodic surveillance testing of charging pumps (Ref. 2) to detect gross degradation caused by impeller structural damage or other hydraulic component problems is performed in accordance with the American Society of Mechanical Engineers (ASME) OM 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 the ASME OM Code. The ASME OM Code provides the activities and Frequencies necessary to satisfy the requirements.

(continued)

BASES (continued)

REFERENCES

1. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989, as clarified by Reference 3.

2. Technical Specification Surveillance Requirement (SR) 3.5.2.4.

3. TVA letter, “Watts Bar Nuclear Plant (WBN) Unit 1 - Technical Requirements Manual (TRM),” dated August 27, 1992 (ADAMS Accession No. ML073230174) including Enclosure 1, “Watts Bar Technical Requirements Manual,” (ADAMS Accession No. ML073620391).

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Charging Pumps, Operating

BASES

BACKGROUND	A description of the Boration Systems Flow Paths is provided in the Bases for Technical Requirement 3.1.1, "Boration Systems Flow Paths, Shutdown."
APPLICABLE SAFETY ANALYSES	The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the Chemical and Volume Control System (CVCS), which causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system and/or stop the primary water pumps. This action is required before the shutdown margin is lost. Operation of the boration subsystem is not assumed to mitigate this event (Ref. 1 & 2). OPERABILITY of the charging pumps, the refueling water storage tank, and the appropriate flow paths is required as part of the Emergency Core Cooling System (ECCS). The Technical Specifications for the ECCS address the requirements of these components.
TR	TR 3.1.4 requires at least two charging pumps to be OPERABLE during MODES 1, 2, and 3 in order to assure redundant pumps to the two redundant flow paths to accomplish (1) normal makeup, (2) chemical shim reactivity control, and (3) miscellaneous fill and transfer operations.
APPLICABILITY	The OPERABILITY of two charging pumps ensures that the CVCS system is available for reactivity control while in MODES 1, 2, and 3. Two charging pumps are required to ensure single functional capability in the event an assumed failure renders one of the pumps inoperable. Charging pump OPERABILITY requirements for MODES 4, 5, and 6 are covered in Technical Requirement 3.1.3, "Charging Pumps, Shutdown."

(continued)

BASES (continued)

ACTIONS

A.1

If one of the required charging pumps is inoperable, action must be taken to restore a required charging pump to OPERABLE status. The 72-hour Completion Time was developed taking into account the redundant capabilities afforded by the OPERABLE charging pump and reasonable time for repairs. The Completion Time is consistent with the time allowed to restore an ECCS train or to restore a boron injection flow path to OPERABLE status (See Technical Specification 3.5.2, "ECCS - Operating" and Technical Requirement 3.1.2, "Boration Systems Flow Paths, Operating.").

A.2.1, A.2.2, and A.2.3

An alternative to Required Action A.1 is to place the plant in MODE 3 and borate to a SDM equivalent to $\geq$ 1% $\Delta k/k$ at 200°F within 78 hours, and restore the required charging pump to OPERABLE status within 246 hours. This precludes the need for a flow path/charging pump for load follow and fuel burnup compensation, allowing the additional 7 days to restore two charging pumps to OPERABLE status. An additional 6 hours (78 hours total) are allowed to reach MODE 3 from full power in an orderly manner and without challenging plant systems. The allowed Completion Time to reach MODE 3 is reasonable, based on operating experience.

B.1

If two charging pumps cannot be restored to OPERABLE status or the Required Actions of Condition A are not met within the associated Completion Times, the plant must be placed in a MODE in which the TR does not apply. This is done by placing the plant in at least MODE 4 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.1.4.1

Periodic surveillance testing of charging pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is performed in accordance with the American Society of Mechanical Engineers (ASME) OM 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 the ASME OM Code. The ASME OM Code provides the activities and Frequencies necessary to satisfy the requirements.

REFERENCES

1. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989, as clarified by Reference 2.

2. TVA letter, "Watts Bar Nuclear Plant (WBN) Unit 1 - Technical Requirements Manual (TRM)," dated August 27, 1992 (ADAMS Accession No. ML073230174) including Enclosure 1, "Watts Bar Technical Requirements Manual," (ADAMS Accession No. ML073620391).

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Borated Water Sources, Shutdown

BASES

BACKGROUND	A description of the Boration System Flow Paths, which include borated water sources is provided in the Bases for Technical Requirement 3.1.1, "Boration System Flow Paths, Shutdown."
APPLICABLE SAFETY ANALYSES	The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the Chemical and Volume Control System which causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system and/or stop the primary water pumps. This action is required before the SDM is lost. Operation of the boration subsystem is not assumed to mitigate this event (Ref. 1 & 4). OPERABILITY of the charging pumps, the Refueling Water Storage Tank (RWST), and the appropriate flow paths is required as part of the Emergency Core Cooling System (ECCS). The Technical Specifications for the ECCS address the requirements of these components.
TR	TR 3.1.5 requires at least one borated water source to be OPERABLE during MODES 4, 5, and 6 to accomplish (1) normal makeup, (2) chemical shim reactivity control, and (3) miscellaneous fill and transfer operations. This requirement may be achieved by one of the following being OPERABLE as required by TR 3.1.1: a. A Boric Acid Storage System (BASS); or b. The RWST.
APPLICABILITY	The OPERABILITY of one borated water source in the required boron injection flow path ensures that this system is available for reactivity control while in MODES 4, 5, and 6. Borated water source OPERABILITY requirements for MODES 1, 2, and 3 are covered in Technical Requirement 3.1.6, "Borated Water Sources, Operating."

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

If the required borated water source is inoperable, the plant must be placed in a condition where negative reactivity addition is not required. This is accomplished by suspending all CORE ALTERATIONS and positive reactivity additions immediately. One borated water source is required to meet the TR and to ensure that negative reactivity control is available during MODES 4, 5, and 6. Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition.

The immediate Completion Time is consistent with the required times for actions requiring prompt attention.

TECHNICAL SURVEILLANCE REQUIREMENTS

The Notes in the Technical Surveillance Requirements state that TSR 3.1.5.1, TSR 3.1.5.2, and TSR 3.1.5.3 are only required to be performed if the RWST is the required borated water source, and TSR 3.1.5.4, TSR 3.1.5.5, and TSR 3.1.5.6 are only required to be performed if the BASS is the required borated water source.

TSR 3.1.5.1

This surveillance requires verification every 24 hours that the RWST temperature is greater than or equal to 60°F (value does not account for instrument error). The Frequency of 24 hours for performance of the surveillance is frequent enough to identify a temperature change that would approach the 60°F temperature limit and has been shown to be acceptable through operating experience. The TSR is modified by a Note which eliminates the requirement to perform this surveillance when ambient air temperature is greater than or equal to 60°F. With ambient air temperature greater than 60°F, the RWST solution temperature should not decrease below this limit, therefore, monitoring is not required.

TSR 3.1.5.2

This surveillance requires verification every 7 days that the boron concentration of the RWST is ≥ 3,100 ppm. This boron concentration is sufficient to provide an adequate SDM and also ensure a pH value between 7.5 and 10.0. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. Since the RWST volume is normally stable, a 7-day Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.1.5.3

This surveillance requires verification every 7 days that the RWST borated water volume is ≥ 62,900 gallons (value does not account for instrument error). This borated water volume is sufficient to provide an adequate SDM and also ensure a pH value between 7.5 and 10.0. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. Since the RWST volume is normally stable, a 7-day Frequency to verify borated water volume is appropriate and has been shown to be acceptable through operating experience. The 62,900 gallon volume requirement includes ~~11,400~~10,900 gallons for shutdown margin and adjustments for minimum safety limit level in the RWST.

TSR 3.1.5.4

This surveillance requires verification every 24 hours that the Boric Acid Tank (BAT) solution temperature is ≥ 63°F (value does ~~not~~ account for instrument error). This ensures that the concentration of boric acid in the BAT is not allowed to precipitate due to cooling. The frequency of 24 hours for performance of the surveillance is frequent enough to identify a temperature change that would approach the 63°F temperature limit.

TSR 3.1.5.5

This surveillance requires verification every 7 days that the boron concentration of the BAT is between 6,120 ppm and 6,990 ppm. This boron concentration is sufficient to provide an adequate SDM and also ensure a pH value between 7.5 and 10.0. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. Since the BAT volume is normally stable, a 7-day Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

TSR 3.1.5.6

This surveillance requires verification every 7 days that the BAT borated water volume is ≥ ~~5,300~~5,100 gallons (value does account for instrument error). This borated water volume is sufficient to provide an adequate SDM and also ensure a pH value between 7.5 and 10.0. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. Since the BAT volume is normally stable, a 7-day Frequency to verify borated water volume is appropriate and has been shown to be acceptable through operating experience.

(continued)

BASES

REFERENCES

1. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989, as clarified by Reference 4.

2. TVA Calculation, EPM-PDM-071197, Revision 6, “Boric Acid Concentration Analysis for BAT and RWST.”

3. CEN-603, "Boric Acid Concentration Reduction Effort, Technical Bases and Operational Analysis for Watts Bar, Unit 1," Revision 00, April 1993.

4. TVA letter, “Watts Bar Nuclear Plant (WBN) Unit 1 - Technical Requirements Manual (TRM), dated August 27, 1992 (ADAMS Accession No. ML073230174) including Enclosure 1, “Watts Bar Technical Requirements Manual,” (ADAMS Accession No. ML073620391).

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Borated Water Sources, Operating

BASES

BACKGROUND	A description of the Boration System Flow Paths, which include borated water sources is provided in the Bases for Technical Requirement 3.1.1, "Boration System Flow Paths, Shutdown."
APPLICABLE SAFETY ANALYSES	The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the Chemical and Volume Control System, which causes a boron dilution event, the ~~automatic~~ response~~, or that~~ required by the operator, is to close the appropriate valves in the reactor makeup system. This action is required before the SDM is lost. Operation of the boration subsystem is not assumed to mitigate this event (Ref. 1). OPERABILITY of the charging pumps, the RWST, and the appropriate flow paths is required as part of the Emergency Core Cooling System (ECCS). The Technical Specifications for the ECCS address the requirements of these components.
TR	TR 3.1.6 requires a Boric Acid Storage System (BASS) and the Refueling Water Storage Tank (RWST) to be OPERABLE as required by TR 3.1.2. This is a requirement during MODES 1, 2, and 3 to accomplish (1) normal makeup, (2) chemical shim reactivity control, and (3) miscellaneous fill and transfer operations.
APPLICABILITY	The OPERABILITY of borated water sources (as required by TR 3.1.2) in the required boron injection flow paths ensures that this system is available for reactivity control while in MODES 1, 2, and 3. Borated water source OPERABILITY requirements for MODES 4, 5 and 6 are covered in Technical Requirement 3.1.5, "Borated Water Sources, Shutdown."

(continued)

BASES (continued)

ACTIONS

A.1, A.2.1, A.2.2, and A.2.3

With the required BASS inoperable, action must be taken to restore the BASS to OPERABLE status within 72 hours. The Completion Time of 72 hours to perform Required Action A.1 is reasonable based upon the typical time necessary to effect repairs and the redundant capabilities afforded by the OPERABLE borated water source.

If the BASS cannot be restored to OPERABLE status the plant must be placed in a MODE in which the requirement does not apply. This is done by placing the plant in at least MODE 3 and by borating to a SDM equivalent to at least 1% $\Delta k/k$ at 200°F in 6 additional hours (78 hours total time). It is also required that the BASS be restored to OPERABLE status in an additional 7 days (246 hours total time).

The 6 additional hours to perform Required Actions A.2.1 and A.2.2 are reasonable and based on operating experience to reach MODE 3 and the required SDM from full power operation in an orderly manner and without challenging plant systems. The 7 day Completion Time per Required Action A.2.3 is based on the low probability of an event occurring during this time period, and the consideration that the remaining borated water sources can provide the required capability.

B.1

If the Required Actions and associated Completion Times of Condition A are not met, the plant must be placed in a MODE in which the TR does not apply. This is done by placing the plant in MODE 4 within 6 hours. The allowed Completion Time is reasonable and based on operating experience to reach required plant conditions in an orderly manner and without challenging plant systems.

C.1

With the RWST boron concentration or borated water temperature not within limits, action must be taken within 8 hours to restore the RWST to OPERABLE status. This 8 hour limit was developed considering the time required to change either the boron concentration or water temperature. The Completion Time is consistent with Technical Specification 3.5.4, “Refueling Water Storage Tank.”

(continued)

BASES

ACTIONS (continued)

D.1

With the RWST inoperable for reasons other than Condition C (e.g., water volume), it must be restored to OPERABLE status within 1 hour. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting two of the boration system flow paths. The Completion Time is consistent with Technical Specification 3.5.4, "Refueling Water Storage Tank."

E.1 and E.2

If the Required Actions and associated Completion Times of Condition C or D are not met, the plant must be placed in a MODE in which the TR does not apply. This is done by placing the plant in MODE 3 within 6 hours and in MODE 4 within 12 hours. The allowed Completion Time is reasonable and based on operating experience to reach required plant conditions in an orderly manner and without challenging plant systems.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.1.6.1

The limits assumed in the accident analysis band for the RWST borated water temperature are ≥ 60°F and ≤ 105°F (values do not account for instrument error). This surveillance requires verification of the water temperature limits every 24 hours. This is frequent enough to identify a temperature change that would approach either temperature limit and has been shown to be acceptable through operating experience.

The TSR is modified by a Note which 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 solution temperature should not exceed the limits.

TSR 3.1.6.2

This surveillance requires verification every 7 days that the boron concentration of the RWST is within the required band. This ensures 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 Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.1.6.3

This surveillance requires verification every 7 days that the RWST borated water volume is within the required limit of ≥ 370,000 gallons (value does not account for instrument error). This will 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, a 7 day Frequency to verify borated water volume is appropriate and has been shown to be acceptable through operating experience.

TSR 3.1.6.4

This surveillance requires verification every 24 hours that the Boric Acid Tank (BAT) solution temperature is ≥ 63°F (value does ~~not~~ account for instrument error). This ensures that the concentration of boric acid in the BAT is not allowed to precipitate due to cooling (Ref. 2 & 3). The Frequency of 24 hours for performance of the surveillance is frequent enough to identify a temperature change that would approach the 63°F temperature limit and has been shown to be acceptable through operating experience.

This surveillance has been modified by a NOTE stating that the surveillance is only required if the BAT is used as one of the required borated water sources for TR 3.1.2.

TSR 3.1.6.5

This surveillance requires verification every 7 days that the boron concentration of the BAT is in accordance with Figure~~s~~ 3.1.6~~A, 3.1.6B, and 3.1.6.C~~ of TR 3.1.6. This boron concentration is sufficient to provide an adequate SDM and also ensure a pH value between 7.5 and 10.0 (Ref. 2 & 3). This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. Since the BAT volume is normally stable, a 7 day sampling Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

This surveillance has been modified by a NOTE stating that the surveillance is only required if the BAT is used as one of the required borated water sources for TR 3.1.2.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.1.6.6

This surveillance requires verification every 7 days that the BAT borated water volume is in accordance with Figure~~s~~ 3.1.6~~A, 3.1.6B, and 3.1.6C~~ (the values listed on the figure do not account for instrument error). This borated water volume at the boron concentration specified in TSR 3.1.6.5 is sufficient to provide an adequate SDM. Since the BAT volume is normally stable, a 7 day Frequency to verify borated water volume is appropriate and has been shown to be acceptable through operating experience.

This surveillance has been modified by a NOTE stating that the surveillance is only required if the BAT is used as one of the required borated water sources for TR 3.1.2.

The maximum expected boration capability requirement occurs near EOL from full power peak xenon conditions and requires borated water from a boric acid tank in accordance with Figure~~s~~ 3.1.6~~A, 3.1.6B, and 3.1.6C~~, and additional makeup from either (1) the common boric acid tank and/or batching tank, or (2) a maximum of ~~23~~22,000 gallons of 3,100 ppm borated water from the refueling water storage tank. With the refueling water storage tank as the only borated water source, a maximum of ~~62~~61,000 gallons of 3,100 ppm borated water is required.

REFERENCES

1. WCAP-11618, “MERITS Program-Phase II, Task 5, Criteria Application,” including Addendum 1 dated April, 1989, as clarified by Reference 4.

2. TVA Calculation, EPM-PDM-071197, Revision 6, “Boric Acid Concentration Analysis For BAT and RWST.”

3. CEN-603, "Boric Acid Concentration Reduction Effort, Technical Bases and Operational Analysis for Watts Bar, Unit 1," Revision 00, April 1993.

4. TVA letter, “Watts Bar Nuclear Plant (WBN) Unit 1 - Technical Requirements Manual (TRM),” dated August 27, 1992 (ADAMS Accession No. ML073230174) including Enclosure 1, “Watts Bar Technical Requirements Manual,” (ADAMS Accession No. ML073620391).

B 3.3 INSTRUMENTATION

B 3.3.2 Engineered Safety Features 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. A detailed Background for ESFAS is given in Reference 1. This TR covers only RESPONSE TIME testing.

The ESFAS RESPONSE TIME is defined as the interval required for the ESF sequence to be initiated subsequent to the time that the appropriate variables exceed the setpoints. This definition is augmented in the Standard Technical Specifications to include automatic system lineups and diesel generator starting and sequence loading delays. The ESF sequence is initiated by the output of the ESFAS, which is by the operation of the dry contacts of the slave relays (600 series relays) in the output cabinets of the Solid State Protection System (SSPS). The RESPONSE TIMES listed in Table 3.3.2-1 of this TR include the interval of time which will elapse between the time the parameter as sensed by the sensor exceeds the safety setpoint and the time the SSPS slave relay dry contacts are operated. The values listed are maximum allowable values consistent with the safety analyses and this Technical Requirement and are systematically verified during plant preoperational startup tests. 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 sequential tests such that the entire response time is measured. The overall ESFAS RESPONSE TIMES are listed in this TR.

The ESFAS is always capable of having response time tests performed using the same methods as those tests performed during the preoperational test program or following significant component changes (Ref. 2).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The required channels of ESFAS Instrumentation provide plant protection in the event of any of the analyzed accidents. The accident analyses described in Reference 3 take credit for operation of ESF systems during DBAs. The safety analyses applicable to each ESFAS function are discussed in the bases for the Technical Specifications, B 3.3.2 (Ref. 1), B 3.3.5 (Ref. 4) and B 3.3.6 (Ref. 5).

TR

OPERABILITY requirements for ESFAS Instrumentation are specified in Technical Specifications, LCOs 3.3.2, 3.3.5 and 3.3.6. TR 3.3.2 requires the ESFAS Instrumentation of Table 3.3.2-1 of the TR to be OPERABLE with RESPONSE TIMES as shown in the table. RESPONSE TIMES must be within the specified limits for the affected instruments to be considered OPERABLE.

APPLICABILITY

Applicable MODES for the specific ESFAS Instrumentation are delineated in Table 3.3.2-1 of Reference 1; in the Applicability of Reference 4; and in Table 3.3.6-1 of Reference 5. The ~~bases for Applicability of each function is~~bases for Applicability of each function are included in References 1, 4, ~~and~~ 5, and 6.

ACTIONS

A.1

The required Actions for inoperable instruments are found in Reference 1. With one or more RESPONSE TIMES outside the specified limits, the affected instrument(s) must be considered inoperable and the appropriate Action referenced in Table 3.3.2-1 of Reference 1; the Actions of Reference 4; or the appropriate Action of Table 3.3.6-1, must be taken. The bases for these actions ~~is~~ are found in References 1, 4, ~~and~~ 5, and 6.

(continued)

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.3.2.1

TSR 3.3.2.1 demonstrates that the ESFAS RESPONSE TIME of each ESFAS function is within the limits listed in Table 3.3.2-1 of the TR. This ensures that the time delays assumed in the safety analyses are not exceeded. Response time tests are conducted on an 18 month STAGGERED TEST BASIS. The 18 month Frequency was developed considering it was prudent that these Surveillances only be performed during a plant outage. This was due to the plant conditions needed to perform the Surveillance and the potential for unplanned plant transients if the Surveillance is performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed on the 18 month Frequency.

Table 3.3.2-1 of this TR specifies the RESPONSE TIMES for the ESFAS Instrumentation.

REFERENCES

1. Watts Bar Technical Specifications (Unit 2), Section 3.3.2, “Engineered Safety Features Actuation System Instrumentation,” and Bases for 3.3.2.

2. Watts Bar FSAR, Section 7.3.1.2.6, “Minimum Performance Requirements.”

3. Watts Bar FSAR, Section 15.0 “Accident Analyses.”

4. Watts Bar Technical Specifications (Unit 2), Section 3.3.5, “LOP Diesel Generator Start Instrumentation,” and Bases for 3.3.5.

5. Watts Bar Technical Specifications (Unit 2), Section 3.3.6, “Containment Vent Isolation Instrumentation,” and Bases for 3.3.6.

6. Westinghouse Letter, WAT-D-11264, “Tennessee Valley Authority Watts Bar Nuclear Plant Containment Spray Delay Time Increase,” dated August 13, 2004.

B 3.3 INSTRUMENTATION

B 3.3.~~3~~ 9 Power Distribution Monitoring System (PDMS)

BASES

BACKGROUND

The Power Distribution Monitoring System (PDMS) generates a continuous measurement of the incore power distribution using the methodology documented in Reference 1. The PDMS employs an advanced three-dimensional nodal code to calculate the incore power distribution. The reference incore power distribution is continuously normalized to the incore flux measurements from the self-powered detector elements. On a nominal once-per-minute basis, the incore power distribution is updated with self powered detector measurements and other plant instrumentation.

The PDMS incore power distribution measurement can be used to determine the most limiting core peaking factors, $F^N_{\Delta H}$, the Nuclear Enthalpy Rise Hot Channel Factor (Technical Specification 3.2.2) and $F_Q(Z)$, the Heat Flux Hot Channel Factor (Technical Specification 3.2.1). The incore power distribution measurement can also be used in the calibration of the excore neutron flux detection system (Technical Specification 3.3.1), monitoring the QUADRANT POWER TILT RATIO (QPTR) (Technical Specification 3.2.4), and verifying the position of a rod with inoperable position indicators (Technical Specification 3.1.8).

The PDMS requires information on current plant and core conditions in order to determine the core power distribution using the core peaking factor measurement and measurement uncertainty methodology described in Reference 1. The OPERABILITY of the PDMS with the specified minimum complement of instrumentation channel inputs ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The PDMS requires input for average reactor vessel inlet temperature, reactor power level, control bank positions, and signals from the Self-Powered Detector (SPD) elements.

The OPERABLE PDMS is to be used for calibration of the Excore Neutron Flux Detection System and~~, monitoring the QUADRANT POWER TILT RATIO, or~~ measurement of $F_Q(Z)$ or $F^N_{\Delta H}$. Similarly, the PDMS may be used for verifying the position of a rod with inoperable position indicators, or monitoring the QUADRANT POWER TILT RATIO.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The PDMS is used for periodic measurement of the core power distribution to confirm operation within design limits and periodic calibration of the excore detectors. This system does not initiate any automatic protection action. The PDMS is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient (References 2, 3, and ~~3~~4).

TR

TR 3.3.~~3~~ 9 requires the PDMS to be OPERABLE with the specified number of instrument channel inputs from the plant computer for each function listed in Table 3.3.~~3~~9-1. The PDMS is OPERABLE when the required channel inputs are available, the calibration data set is valid, and reactor power is ≥~~>~~ 25% RTP.

This TR ensures the OPERABILITY of the PDMS when required to monitor the power distribution within the core. The PDMS is used for periodic surveillance of the incore power distribution and calibration of the excore detectors. The surveillance of incore power distribution verifies that the peaking factors are within their design envelope (References 3 and 4). The peaking factor limits include measurement uncertainty which bounds the actual measurement uncertainty of an OPERABLE PDMS (Reference 1).

Maintaining the minimum number of instrumentation channel inputs ensures the uncertainty is bounded by the uncertainty methodology. Similarly, when THERMAL POWER is less than 25% RTP, then the accuracy of the adjustment provided by the Self-Powered Detector (SPD) elements ~~Core Exit Thermo-Couples (CETCs)~~ to the measured PDMS power distribution may not be bounded by the uncertainties documented in Reference 1.

APPLICABILITY

The PDMS must be OPERABLE when it is used for calibration of the Excore Neutron Flux Detection System, monitoring the QPTR, measurement of $F^N_{\Delta H}$ and $F_Q(Z)$, or verifying the position of a rod with inoperable position indicators.

(continued)

BASES (continued)

ACTIONS

A.1

The Required Action A.1 has been modified by a Note stating that the provisions of TR 3.0.3 do not apply.

With THERMAL Power less than 25% RTP or with one or more required channel inputs inoperable or unavailable to the PDMS, the PDMS must not be used to obtain an incore power distribution measurement. Therefore, the Required Action A.1 prohibits the use of the inoperable system for the applicable monitoring or calibration functions.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.3.~~3~~9.1

Performance of the CHANNEL CHECK ensures that a gross instrumentation failure has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels.

A CHANNEL CHECK will detect gross channel failure, thus it is a 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 isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

The Frequency of 24 hours is sufficient considering the PDMS provides automatic validation of the channel inputs and either discards the inoperable channel input or declares itself inoperable, but at the same time ensures that the required channel inputs to the PDMS are manually verified to be valid within a reasonable time frame prior to using the PDMS to obtain an incore power distribution measurement.
~~to the average RCS Loop ΔT power input.~~

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.3.~~3~~9.2

Verification by administrative means of the surveillance requirements required elsewhere ensures the instrumentation channels satisfy nominal accuracy and reliability for power operation. Many of these surveillance requirements are CHANNEL CALIBRATIONS.

CHANNEL CALIBRATIONS are typically 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 a measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATION must be performed consistent with the assumptions of the 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.

The Frequency of 18 months is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of sensor/transmitter drift in the setpoint methodology.

Eight~~Four~~ notes modify the ~~surveillance~~ instrumentation requirements specified in Table 3.3.9-1.
~~Table 3.3.3-1.~~

Note 1 allows up to three parameters to be used for reactor power input into PDMS, but BEACON™ will only accept two options at any one time.

Note 2 allows the control bank position input to come from either the Demand Position Indication or the average of the individual Rod Position Indications.

Note 3 ~~clarified~~ clarifies that the CHANNEL CALIBRATION requirements ~~do not apply to the secondary calorimetric power or average power range neutron flux power inputs~~specified in this table apply only to the RCS Loop ΔT power input.

Note 4 ~~clarified~~ clarifies that the calorimetric heat balance adjustment is not applicable to the average RCS Loop ΔT power input.

Note 5 allows a reduced number of SPDs subsequent to the initial incore power distribution measurement in each fuel cycle.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.3.9.2 (continued)

Note 6 clarifies that the TSR 3.3.1.10 requirements specified in this table apply only to the Narrow Range RTDs.

Note 7 clarifies that the TSR 3.3.9.2 requirements specified in this table apply only to the Wide Range RTDs.

Note 8 allows the RCS Cold Leg Temperature to come from either the Narrow or Wide Range RTDs.

~~TSR 3.3.3.3~~

A functional check of the PDMS software system~~The PDMS must be calibrated using an incore power distribution measurement data set obtained above 25% RTP to ensure the accuracy of the calibration data set which is derived from the SPD and other input channels. The initial calibration in each fuel cycle must utilize incore flux measurements from at least 218 of the SPD elements meeting the distribution requirements provided in Table 3.3.3-1. The incore flux measurements in combination with at least the minimum channel inputs from Table 3.3.3-1 are used to generate the calibration data set, including nodal calibration factors. Subsequent PDMS calibrations require only ≥ 145 of the SPD elements meeting the distribution requirements listed in Table 3.3.3-1.~~

REFERENCES

1. WCAP-12472-P-A, "BEACON™ Core Monitoring and Operations Support System," August 1994 (Addendum 1, January 2000; Addendum 2, April 2002; and Addendum 4, 2012).

2. 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors."

3. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989, as clarifed by Reference 4.

4. TVA letter, "Watts Bar Nuclear Plant (WBN) Unit 1 - Technical Requirements Manual (TRM)," dated August 27, 1992 (ADAMS Accession No. ML073230174) including Enclosure 1, "Watts Bar Technical Requirements Manual," (ADAMS Accession No. ML073620391).

B 3.3 INSTRUMENTATION

B 3.3.4 Seismic Instrumentation

BASES

BACKGROUND

The seismic instrumentation is common to both units and is made up of several instruments such as accelerometers, an accelerograph, recorders, etc. These instruments are placed in several appropriate locations throughout the plant in order to provide data on the seismic input to containment, data on the frequency, amplitude and phase relationship of the seismic response of the containment structure, and data on the seismic input to other Seismic Category I structures (Ref. 1).

The seismic instrumentation is used to promptly determine the nature and severity of a seismic event and to predict the impact (i.e., potential for damage) on nuclear power plant features which are important to safety. This is required to permit comparison of the measured response to that used in the design basis for the unit to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Reference 1.

The original seismic instrumentation was replaced with state of the art digital instrumentation in order to permit application of EPRI OBE exceedance criteria delineated in References 4 and 5. Use of these criteria is permitted by Reference 6 provided that upgraded instrumentation is used. The replacement instrumentation is capable of recording a seismic event and performing appropriate analyses of the recorded data to provide an immediate basis for determining whether an OBE exceedance has occurred. Reference 6 directs that this information must be evaluated within 4 hours after an event and a walkdown of critical plant features must be accomplished within 8 hours after an event in order to make a determination as to whether a plant shutdown is warranted.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and to determine the impact on those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the unit to determine if plant equipment inspection is required pursuant to Appendix A of 10 CFR Part 100 prior to restart. Seismic risks which appear as dominant sequences in PRAs occur for very severe earthquakes with magnitudes which are a factor of two or three above the Safe Shutdown Earthquake and Design Basis Earthquake. The Seismic Instrumentation System was not designed to function or to provide comparative information for such severe earthquakes. This instrumentation is more pertinent to determining the need to shut down following a seismic event and the ability to restart the plant after seismic events which are not risk contributors, and is therefore not of prime importance in risk dominant sequences (Refs. 2 & 8).

TR

TR 3.3.4 requires that the seismic monitoring instrumentation which is shown in Table 3.3.4-1 shall be OPERABLE. This requirement ensures that an assessment can be made of the effects on the plant of earthquakes which may occur that exceed the design basis spectra for the Operating Basis Earthquake (Ref. 3).

APPLICABILITY

Since the possibility of earthquakes is not MODE dependent, OPERABILITY of the seismic instrumentation is required at all times. The Applicability has been modified by a Note stating that the provisions of TR 3.0.3 do not apply.

ACTIONS

A.1

The determination as to whether an OBE exceedance has occurred is made by comparing the calculated spectra for the event with the applicable design basis spectra for that building and location. Reference 6 requires that this determination be made considering the data from instruments located on the Containment foundation. Therefore, the exceedance determination for WBN will be made using event data from 0-XT-52-75A in the Containment annulus. Data from this instrument is recorded at panel 0-R-113, which also contains the computer used to calculate the spectral content and the alarm panel used to annunciate in the control room. These devices are the key components used to detect

(continued)

BASES

ACTIONS

A.1 (continued)

the event and make a shutdown determination. With one or more of these required seismic monitoring instruments inoperable for more than 30 days, the inoperability of the instruments must be documented in accordance with the Corrective Action Program.

With one or more of the remaining seismic instruments inoperable for more than 60 days, the inoperability of the instruments must be documented in accordance with the Corrective Action Program. A longer period of inoperability is allowed for these instruments since they are used only for evaluating plant condition following an event and not for input to the shutdown decision.

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

When one or more seismic monitoring instruments actuate during a seismic event with greater than or equal to 0.01g ground acceleration, all of the Required Actions under Condition B must be completed. The data retrieved from the actuated instruments must be analyzed to determine the magnitude of the vibratory ground motion. The replacement digital instrumentation provides the capability to analyze the event data onsite and generate event spectra to be used in determining whether an OBE exceedance has occurred. If an OBE exceedance has occurred, Reference 6 directs that this evaluation should occur within 4 hours after the event. Reference 6 also requires performance of a limited scope walkdown per Reference 7 to determine the extent of actual damage within 8 hours following the event. The information provided by this walkdown and the spectral analysis are to be used in making a determination as to whether to proceed with plant shutdown. In addition, the seismic event must be documented in accordance with the Corrective Action Program.

B.4 and B.5

Each actuated monitoring instrument must be restored to OPERABLE status within 24 hours. Within 10 days of the actuation, a CHANNEL CALIBRATION must be performed on each actuated monitoring instrument. The Completion Time of 10 days to perform Required Action B.2 is reasonable and is based on engineering judgment.

(continued)

BASES

ACTIONS (continued)

B.6

Subsequent analysis must then be performed using data from the remaining seismic monitoring instruments to evaluate the plant response in comparison with previously generated design basis spectra at the locations of those instruments. The Completion Time of 14 days to perform Required Action B.6 is reasonable and based upon the typical time necessary to analyze data.

TECHNICAL SURVEILLANCE REQUIREMENTS

The TSRs for each seismic monitoring Function are identified by the SRs column of Table 3.3.4-1.

A Note has been added to the TSRs to clarify that Table 3.3.4-1 determines which SRs apply to which seismic monitoring instruments.

TSR 3.3.4.1

Performance of a CHANNEL CHECK on the seismic instrumentation once every 31 days ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a check of external system status indications that the seismic monitoring equipment is in a state of readiness to properly function should an earthquake occur. A CHANNEL CHECK will detect gross system failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL OPERATIONAL TEST.

The Surveillance Frequency of 31 days is based on operating experience related to instrumentation systems, which demonstrates that gross instrumentation system failure in any 31 day interval is a rare event. The CHANNEL CHECK supplements the loss of power annunciation for the equipment in the auxiliary instrument room. The equipment in the auxiliary control room does not have a loss of power alarm but only provides supplemental data.

TSR 3.3.4.2

A CHANNEL OPERATIONAL TEST is to be performed on each required channel to ensure the entire channel will perform the intended function. A CHANNEL OPERATIONAL TEST is the comparison of the response of the instrumentation, including all components of the instrument, to a known signal. Although the seismic trigger is functionally checked, its setpoint is not verified. The 184 day Surveillance Frequency is based upon the known reliability of the monitoring instrumentation and has been shown to be acceptable through operating experience.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.3.4.3

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor by comparing the response of the instrument to a known input on the sensor. This test verifies the capability of the seismic instrumentation to correctly determine the magnitude of a seismic event and evaluate the response of those features important to safety. The 18 month Surveillance Frequency is based upon operating experience and consistency with the typical industry refueling cycle.

REFERENCES

1. Regulatory Guide 1.12, "Instrumentation for Earthquakes," Revision 1, April 1974.

2. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989, as clarified by Reference 8.

3. Watts Bar FSAR, Section 3.7.4, "Seismic Instrumentation Program."

4. EPRI NO-5930, July 1988, "A Criterion for Determining Exceedance of the Operating Basis Earthquake."

5. EPRI TR-104239, June 1994, "Seismic Instrumentation in Nuclear Power Plants for Response to OBE Exceedance: Guideline for Implementation."

6. Regulatory Guide 1.166, "Pre-Earthquate Planning And Immediate Nuclear Power Plant Operator Post-Earthquake Actions," Revision 0, March 1997.

7. EPRI NP-6695, December 1989, "Guidelines for Nuclear Plant Response to an Earthquake."

8. TVA letter, "Watts Bar Nuclear Plant (WBN) Unit 1 - Technical Requirements Manual (TRM)," dated August 27, 1992 (ADAMS Accession No. ML073230174) including Enclosure 1, "Watts Bar Technical Requirements Manual," (ADAMS Accession No. ML073620391).

B 3.3 INSTRUMENTATION

B 3.3.5 ~~Turbine Overspeed Protection~~Deleted

~~BASES~~

~~BACKGROUND~~

~~Three types of overspeed protection mechanisms are provided to isolate main steam to the turbo-generator when the rated operating speed of 1800 rpm is exceeded. During normal speed-load control, the Analog Electro Hydraulic (AEH) Overspeed Protection Control (OPC) which is set at 1854 rpm (103 percent of rated speed) will rapidly close the governor and interceptor valves in case of an overspeed condition. Rotational speed is then maintained below this runback setpoint by moving the interceptor valves between the closed and open position until the reheater steam (steam between the high pressure turbine exhaust and the low pressure turbines) is dissipated. If the AEH control system is in the automatic mode, the governor valves will take over speed control and will maintain reference speed. However, if the AEH control system is in the manual mode (normally only at low power levels during startup), the turbine generator will coast down to turning gear operation, if no operator action is taken.~~

~~If for some reason the AEH OPC control system does not function and the turbine speed increases to 1980 rpm (110 percent of rated speed), the mechanical overspeed mechanism will trip close all steam valves (throttle, governor, reheat, stop, and interceptor valves and prevent the turbine speed from exceeding 120 percent of rated speed. The unit will then coast down to turning gear operation.~~

~~In addition to these two control systems, an independent electrical overspeed trip is provided in the Analog Electro Hydraulic (AEH) Control System. If the turbine generator speed increases to 1998 rpm (111 percent of rated speed), all steam valves (as listed in the previous paragraph) will be tripped closed. This trip will be actuated by a contact output from the AEH controller which energizes a trip solenoid in the autostop oil fluid lines. Again, during the overspeed condition, turbine speed will remain below 120 percent of rated speed. The unit will then coast down to turning gear operation. (Ref. 1)~~

(continued)

~~BASES (continued)~~

~~APPLICABLE SAFETY ANALYSES~~	~~The Turbine Overspeed Protection System trips the turbine to prevent the generation of potentially damaging missiles from the turbine, in the event of a loss of the Turbine Speed Control System, or a transient. However, the turbine overspeed event is not a DBA (Ref. 2). Turbine Overspeed Protection is not assumed to function in the safety analyses.~~
~~TR~~	~~This requirement is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could present a personnel and equipment hazard.~~
~~APPLICABILITY~~	~~At least one Turbine Overspeed Protection System must be OPERABLE whenever the potential for turbine overspeed exists. Since steam may be admitted to the turbine in MODES 1, 2, or 3, the requirement is applicable in these MODES.~~ ~~The Applicability has been modified by a Note stating that it is not applicable to MODES 2 and 3 when all main steam isolation valves are closed and all other steam flow paths to the turbine are isolated. Under these conditions, the potential for turbine overspeed does not exist.~~
~~ACTIONS~~	~~A.1.1 and A.2.1~~ ~~If one high pressure turbine steam inlet valve is inoperable, action must be taken to verify the two high pressure turbine steam inlet valves on the same steam chest which are opposite the inoperable valve are OPERABLE. The verification of operability (by testing) is needed to assure that these valves will close when the turbine is tripped. The 6 hour Completion Time was developed taking into account the time to reduce power and conduct the functional testing of the turbine steam inlet valves.~~

(continued)

~~BASES~~

~~ACTIONS (continued)~~

~~A.1.2~~

~~If one high pressure turbine steam inlet valve is inoperable, action must be taken to restore the inoperable valve to OPERABLE status. The 72 hour Completion Time was developed taking into account the redundant capabilities afforded by the OPERABLE valves in the same steam chest and reasonable time for repairs. OPERABILITY is established by either having a fully functional valve or manually closing the otherwise inoperable valve. Closing the inoperable valve ensures total steam isolation to the high pressure turbine in the event of an overspeed condition, even with a single failure of another valve. Since the turbine must not have flow into non-adjacent zones (i.e., governor valves 1 and 3 or governor valves 2 and 4) due to possible turbine damage, isolation of a steam chest is precluded.~~

~~A.2.2~~

~~This alternative ensures total steam isolation to the high pressure turbine in the event that the inoperable valve cannot be closed. If it is determined through testing that another valve is also inoperable, then sufficient time would be available for an orderly turbine shutdown. This action is intended to prevent a condition involving a turbine trip coincident with multiple inoperable high pressure turbine steam inlet valves resulting in steam flow to the turbine. In this condition, insufficient time would be available for the operator to isolate the high speed turbine to avoid turbine overspeed. Since the turbine must not have flow through non-adjacent zones (i.e., governor valves 1 and 3 or governor valves 2 and 4) due to possible turbine damage, isolation of a steam chest is precluded. An additional 6 hours (total of 78 hours) are allowed for a power reduction, if necessary, in an orderly manner and without challenging plant systems.~~

~~A.3~~

~~Another alternative is to isolate the turbine from the steam supply using the Main Steam Isolation Valves (MSIV). This alternative assumes there may be partial leakage through the high pressure turbine steam inlet valves, or the steam path into the turbine cannot be isolated by the turbine inlet valves, thereby requiring closure of the MSIVs. This places the turbine in a condition where overspeed protection is not required. Again, an additional 6 hours (total of 78 hours) are allowed for a power reduction, if necessary, in an orderly manner and without challenging plant systems.~~

(continued)

~~BASES~~

~~ACTIONS (continued)~~

~~B.1~~

~~If one reheat stop valve or one reheat intercept valve in one or more low-pressure turbine steam lines is inoperable, action must be taken to restore the inoperable valve(s) to OPERABLE status. The 72 hour Completion Time was developed taking into account the redundant capabilities afforded by the OPERABLE valve in the same steam line(s) and reasonable time for repairs.~~

~~B.2~~

~~A first alternative to Required Action B.1 is to close at least one valve in the affected steam line(s). This places the low pressure steam line(s) with the inoperable valve(s) in a no flow condition. This ensures total steam isolation to the low pressure turbine(s) in the event of an overspeed condition, even with a single failure of another reheat stop valve or reheat intercept valve. An additional 6 hours (total of 78 hours) are allowed for a power reduction, if necessary, in an orderly manner and without challenging plant systems.~~

~~B.3~~

~~A second alternative is to isolate the turbine from the steam supply. This places the turbine in a condition where overspeed protection is not required. Again, an additional 6 hours (total of 78 hours) are allowed for a power reduction, if necessary, in an orderly manner and without challenging plant systems.~~

~~C.1~~

~~If the Turbine Overspeed Protection System is inoperable for causes other than Condition A or Condition B, the turbine must be placed in a condition where overspeed protection is not required. This is accomplished by isolating the turbine from the steam supply system. A Completion Time of 6 hours is allowed to shutdown the turbine in an orderly manner and without challenging plant systems.~~

~~TECHNICAL SURVEILLANCE REQUIREMENTS~~

~~TSR 3.3.5.1~~

~~The Turbine Overspeed Protection System testing requirements and frequencies are provided in Reference 3.~~

(continued)

~~BASES (continued)~~

~~REFERENCES~~

~~1. Watts Bar FSAR, Section 10.2, "Turbine Generator."~~

~~2. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.~~

~~3. Turbine Integrity Program With Turbine Overspeed Protection (TIPTOP).~~

B 3.3 INSTRUMENTATION

B 3.3.6 Loose-Part Detection System

BASES

BACKGROUND

The Loose-Part Detection System consists of 12 sensors, a system cabinet, alarm units, and an audio monitor. The sensors are located in the six natural collection regions. These regions consist of the top and bottom plenums of the reactor vessel and the primary coolant inlet plenum to each steam generator. The entire system is described in Reference 1.

The Loose-Part Detection System provides the capability to detect acoustic disturbances indicative of loose parts within the Reactor Coolant System (RCS) pressure boundary. This system is provided to avoid or mitigate damage to RCS components that could occur from these loose parts. The Loose-Part Detection System Technical Requirement is consistent with the recommendations of Reference 2.

APPLICABLE SAFETY ANALYSES

The presence of a loose part in the RCS can be indicative of degraded reactor safety resulting from failure or weakening of a safety-related component. A loose part, whether it ~~be~~is from a failed or weakened component, or from an item inadvertently left in the primary system during construction, refueling, or maintenance, can contribute to component damage and material wear by frequent impacting with other parts in the system. Also, a loose part increases the potential for control-rod jamming and for accumulation of increased levels of radioactive crud in the primary system (Ref. 2).

The Loose-Part Detection System provides the capability to detect loose parts in the RCS which could cause damage to some component in the RCS. Loose parts are not assumed to initiate any DBA, and the detection of a loose part is not required for mitigation of any DBA (Refs. 3 & 4).

TR

TR 3.3.6 requires the Loose-Part Detection System to be OPERABLE. This is necessary to ensure that sufficient capability is available to detect loose metallic parts in the RCS and avoid or mitigate damage to the RCS components. This requirement is provided in Reference 2.

(continued)

BASES (continued)

APPLICABILITY

TR 3.3.6 is required to be met in MODES 1 and 2 as stated in Reference 2. These MODES of applicability are provided in Reference 2.

The Applicability has been modified by a Note stating that the provisions of TR 3.0.3 do not apply.

ACTIONS

A.1

With one or more Loose-Part Detection System channels inoperable for more than 30 days, document the inoperability of the channel in accordance with the Corrective Action Program.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.3.6.1

Performance of a CHANNEL CHECK for the Loose-Part Detection System once every 24 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 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 plant 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 Surveillance and the Surveillance Frequency are provided in Reference 2.

TSR 3.3.6.2

A CHANNEL OPERATIONAL TEST is to be performed every 31 days on each required channel to ensure the entire channel will perform the intended function. This test verifies the capability of the Loose-Part Detection System to detect impact signals which would indicate a loose part in the RCS. The Surveillance and the Surveillance Frequency are provided in Reference 2.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.3.6.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The 18 month Surveillance Frequency is based upon operating experience and is consistent with the typical industry refueling cycle. The Surveillance and the Surveillance Frequency are provided in Reference 2. Reference 1 describes the use of a computer-based analytical system to verify proper channel calibration. This is an acceptable option to using a mechanical impact device for sensors located in plant areas where plant personnel radiation exposure is considered by Plant Management to be excessive.

REFERENCES

1. Watts Bar FSAR, Section ~~4.4.6~~7.6.7, "~~Digital Metal Impact Monitoring System (DMIMS-DX™)~~Loose Part Monitoring System (LPMS) System Description."

2. Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors."

3. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989, as clarified by Reference 4.

4. TVA letter, "Watts Bar Nuclear Plant (WBN) Unit 1 - Technical Requirements Manual (TRM)," dated August 27, 1992 (ADAMS Accession No. ML073230174) including Enclosure 1, "Watts Bar Technical Requirements Manual," (ADAMS Accession No. ML073620391).

B 3.3 INSTRUMENTATION

B 3.3.8 Hydrogen Monitor

BASES

BACKGROUND

A Hydrogen Monitor is provided to detect high hydrogen concentration conditions that represent a potential for containment breach from a hydrogen explosion. This variable is also useful in verifying the adequacy of postaccident mitigating actions. Hydrogen concentration may also be used to determine whether or not the Hydrogen Ignitors should be started or other action taken. Containment hydrogen instrumentation has a monitoring range of 0-10% (by volume) hydrogen concentration.

By rule change in 2003, 10 CFR 50.44 (Ref. 1) no longer defined a design basis Loss of Coolant Accident (LOCA) hydrogen release, and eliminated the requirements for hydrogen control systems to mitigate such a release. The installation of Hydrogen Recombiners and/or vent and purge systems required by 50.44(b)(3) prior to revision in 2003 was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design basis LOCA. The Commission found that this hydrogen release was not risk significant because the design-basis LOCA hydrogen release did not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage. In addition, these systems were ineffective at mitigating hydrogen releases from risk significant accident sequences that could threaten Reactor Building integrity. The Improved Standard Technical Specifications (ISTS) at the time stated that Hydrogen Recombiners met 10 CFR 50.36(c)(2)(ii) Criterion 3 (accident mitigation). As stated in the rule change, since Hydrogen Recombiners are no longer required to respond to a LOCA, the Hydrogen Recombiners no longer meet Criterion 3 or any of the other criteria for retention in the Technical Specifications (TSs). Therefore, the new rule states that the requirements related to Hydrogen Recombiners currently in the ISTS no longer meet the criteria of 10 CFR 50.36(c)(2)(ii) for retention in the TSs and may be eliminated.

With the elimination of the design-basis LOCA hydrogen release, the Hydrogen Monitors are also no longer required to mitigate design basis accidents and, therefore, the Hydrogen Monitors do not meet the definition of a safety-related component as defined in 10 CFR 50.2 (Ref. 2). Regulatory Guide (RG) 1.97 (Ref. 3) Category 1 instrumentation is intended for key variables that most directly indicate the accomplishment of a safety function for design basis accident events. The Hydrogen Monitors no longer meet the definition of Category 1 in RG 1.97. As part of the rulemaking to revise 50.44 the Commission found that Category 3,

(continued)

Bases

BACKGROUND (continued)	as defined in RG 1.97, is an appropriate categorization for the Hydrogen Monitors because the monitors are required to diagnose beyond design basis accidents. Hydrogen monitoring is not the primary means of indicating a significant abnormal degradation of the reactor coolant pressure boundary and has been found to not be risk-significant. Therefore, the rule making stated that hydrogen monitoring equipment requirements no longer meet the criteria of 50.36(c)(2)(ii) for retention in TSs and, therefore, may be removed from the TSs.
APPLICABLE SAFETY ANALYSES	As stated in the BACKGROUND section, the Hydrogen Monitor is no longer required for mitigation of design basis accidents. Based on this, the Hydrogen Monitor does not meet the definition of a safety-related component. However, the elimination of Hydrogen Recombiners in accordance with Technical Specification Task Force (TSTF)-447, Rev. 1 (Ref. 4), was contingent on each licensee maintaining the capability to monitor hydrogen concentrations in the Reactor Building during beyond design basis accidents. This TR maintains that commitment (Ref. 5).
TR	The Hydrogen Monitor is required to be OPERABLE to ensure that the necessary equipment will be available to monitor the hydrogen concentration within containment during significant beyond design-basis accident conditions (Ref. 5).
APPLICABILITY	The Hydrogen Monitor is required to be OPERABLE 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 hydrogen monitoring is low; therefore, the monitor is not required to be OPERABLE in these MODES.
ACTIONS	A.1 Seven days to restore the hydrogen monitor capability is reasonable given the requirement to be available for use in diagnostics during a beyond design basis event.

(continued)

Bases

ACTIONS (continued)

B.1

Condition B applies when the Required Action and associated Completion Time for Condition A are not met. This Required Action specifies that the failure to comply with the 7 day Completion Time must be documented in the Corrective Action Program so that the impact of the inoperable equipment with regard to continued plant operation may be evaluated. Consideration should be given to alternate means, such as core damage assessments performed under the Severe Accident Management Guidelines during the extended period the Hydrogen Monitor is inoperable. During normal operations the probability of occurrence of a beyond design basis accident is low and therefore, continued plant operation should not be significantly impacted. However, the actions required to restore the inoperable monitor should be pursued in a manner that is commensurate with the component's importance to safety.

TECHNICAL SURVEILLANCE REQUIREMENTS

SR 3.3.~~3~~8.1

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 with the necessary range and accuracy. The Frequency is based on operating experience and consistency with the typical industry refueling cycle.

(continued)

Bases

REFERENCES

1. 10 CFR 50.44, “Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors,” October 16, 2003.

2. 10 CFR 50.2, “Definition of Safety Related Structures, Systems, and Components.”

3. Regulatory Guide 1.97, Revision 2, December 1980, “Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.”

4. TSTF-447, Revision 1, “Elimination of Hydrogen Recombiners and Change to Hydrogen and Oxygen Monitors.”

5. Commitment (NCO080031001) made in TVA’s letter dated September 4, 2008, to maintain a hydrogen monitoring system capable of diagnosing beyond design basis accidents.

6. Regulatory Guide 1.7, Revision 3, “Control of Combustible Gas Concentrations in Containment.”

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 Safety Valves, Shutdown

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Trip 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 (Ref. 1).

Because the safety valves are totally enclosed and self-actuating, they are considered independent components. The relief capacity for each valve, 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.

Overpressure protection is required in MODES 1, 2, 3, 4, 5, and 6 (with the reactor vessel head on); however, in MODE 4, MODE 5 and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of Technical Specification 3.4.12, “Cold Overpressure Mitigation System (COMS).”

The upper and lower pressure limits are based on the ±± 3% tolerance. 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 above 350°F. OPERABILITY of the safety valves ensures that 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. 2) could include damage to RCS components, increased LEAKAGE, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The pressurizer safety valves protect the RCS from being pressurized above the RCS pressure Safety Limit. The pressurizer safety valves provide overpressurization protection during both power operation and hot standby. However, the pressurizer safety valves are not assumed to function to mitigate a DBA or transient in MODES 4 and 5 (Refs. 3 & 5).

TR

This requirement is provided to ensure continuity in the restructuring of Standard Technical Specifications. Reactor Coolant System overpressure protection is provided in MODES 4 and 5 by the Cold Overpressure Mitigation System (COMS) covered by Technical Specification LCO 3.4.12.

Reference 4 specifies requirements which, when met, may preclude the need for this TR.

A Note modifies this TR to indicate that the lift setting of the pressurizer Code safety valves can be outside the required lift setting when in MODE 4 for the purpose of setting at hot ambient conditions. Safety valves can lift at a slightly different pressure as the valve temperature varies. Therefore, setting the safety valve for nominal operating conditions in MODE 1 may result in a lift pressure drifting outside the required tolerance limits as the plant is shutdown to MODE 5. This exception is allowed for entry and operation into and exit from MODES 4 and 5 provided a preliminary cold setting was made prior to heatup.

APPLICABILITY

The OPERABILITY of one pressurizer Code safety valve ensures that overpressure protection is provided in MODES 4 and 5. OPERABILITY of Code safety valves is not required in MODE 6. Code safety valve OPERABILITY requirements for MODES 1, 2, and 3 are covered in Technical Specification 3.4.10, “Pressurizer Safety Valves.”

ACTIONS

A.1

With no pressurizer Code safety valves OPERABLE, the plant must be placed in a condition which minimizes the risk of a pressure spike large enough to actuate a safety valve. This is done by suspending all operations involving positive reactivity changes. The immediate Completion Time for performance of Required Action A.1 shall not preclude completion of actions to establish a safe condition.

(continued)

BASES

ACTIONS (continued)

A.2

In addition to Action A.1, an OPERABLE Residual Heat Removal loop shall be placed in operation in the shutdown cooling mode. This provides overpressure protection through the Residual Heat Removal suction and discharge relief valves. The immediate Completion Time requires an operator to initiate actions to place the loop in shutdown cooling. Once actions are initiated, they must be continued until the loop is in the shutdown cooling mode.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.4.1.1

TSR 3.4.1.1 requires verification that the pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program.

REFERENCES

1. Watts Bar FSAR, Section 5.5.13, “Safety and Relief Valves.”

2. ASME Boiler and Pressure Vessel Code, Section III, NB 7000.

3. WCAP-11618, “MERITS Program-Phase II, Task 5, Criteria Application,” including Addendum 1 dated April, 1989, as clarified by Reference 5.

4. 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 50.54(f).

5. TVA letter, “Watts Bar Nuclear Plant (WBN) Unit 1 - Technical Requirements Manual (TRM),” dated August 27, 1992 (ADAMS Accession No. ML073230174) including Enclosure 1, “Watts Bar Technical Requirements Manual,” (ADAMS Accession No. ML073620391).

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 Pressurizer Temperature Limits

BASES

BACKGROUND

The pressurizer is an ASME Section III, vertical vessel with hemispherical top and bottom heads constructed of carbon steel. The vessel is clad with austenitic stainless steel on all surfaces exposed to the reactor coolant. A stainless steel liner or tube may be used in lieu of cladding in some nozzles. The surge line nozzle and removable electric heaters are installed in the bottom head. Spray line nozzles, relief and safety valves are located in the top head of the vessel. A small continuous spray is provided through a manual bypass valve around the power-operated spray valves. The temperature, and hence the pressure are controlled by varying the power input to selected heater elements. The pressurizer is designed to withstand the effects of cyclic loads due to pressure and temperature changes. These loads are introduced by startup and shutdown operations, power transients and reactor trips. During startup and shutdown, the rate of temperature change is controlled by the operator. Heatup rate is controlled by the input to the heater elements, and cooldown is controlled by spray. When the pressurizer is filled with water, i.e., during initial system heatup, and near the end of the second phase of plant cooldown, Reactor Coolant System (RCS) pressure is maintained by the letdown flow rate via the Residual Heat Removal System.

These Bases address the control of the rate of change of temperature and the effect of the thermal cycling on critical areas of the pressure boundary of the pressurizer. The Reactor Coolant Pressure Boundary, which includes the pressurizer, is defined in 10 CFR 50, section 50.2 (Ref. 1). General rules for design and fabrication are provided in 10 CFR 50, section 50.55a (Ref. 2). These design and fabrication rules are based on the ASME Boiler and Pressure Vessel Code.

APPLICABLE SAFETY ANALYSES

The limits on the rate of change of temperature for the heatup and cooldown of the pressurizer are not derived from Design Basis Accident analyses (Refs. 3 & 5). The limits are prescribed during normal operation to limit the cyclic, thermal loading on critical areas in the pressure boundary. The limits on the rate of change of temperature have been established, using approved methodology, to preclude operation in an unanalyzed condition.

(continued)

BASES (continued)

TR

TR 3.4.2 specifies the acceptable rates of heatup and cooldown of the pressurizer. These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to cyclic induced failure in the pressure boundary of the pressurizer.

APPLICABILITY

The limits on the rate of change of temperature provide a definition of acceptable operation to limit cyclic temperature loading to analyzed conditions. Although these limits were developed to provide rules for operation during heatup and cooldown (MODES 3, 4, and 5), they are applicable at all times.

ACTIONS

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

If the rate of change of temperature is outside the limits, the rate of temperature change must be restored to within limits in 30 minutes. 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 corrective actions can be accomplished in this time in a controlled manner. In addition to restoring operation to within limits, an evaluation is required within 72 hours to determine if operation may continue. This may require event-specific stress analyses or inspections. A favorable evaluation must be completed before continuing operation. The 72-hour Completion Time is consistent with that allowed in Technical Specification 3.4.3, “RCS Pressure and Temperature Limits.”

A Note is provided to clarify that all Actions must be completed whenever this Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration to within limits is insufficient without the evaluation of the structural integrity of the pressure boundary of the pressurizer.

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 and the pressure reduced. This will allow a more careful examination of the event. The 6-hour Completion Time is reasonable, considering operating experience, to reach MODE 3 from full power. The additional 30 hours to reduce the pressure to < 500 psig in an orderly manner also considers operating experience. This reduction in pressure is possible without challenging the plant systems or violating any operating limits.

(continued)

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.4.2.1

TSR 3.4.2.1 verifies that the rate of heatup and the rate of cooldown are within limits. "Step wise" cooling must be avoided as discussed in Reference 4. The 30-minute Frequency is considered reasonable in view of the instrumentation available in the control room to monitor the status of the RCS. The Surveillance has been modified by a Note which requires the Surveillance to be performed only during heatup and cooldown of the system.

REFERENCES

1. 10 CFR 50.2, "Definitions."
2. 10 CFR 50.55a, "Codes and Standards."
3. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989, as clarified by Reference 5.
4. Westinghouse letter WAT-D-8376, "Reactor Coolant System Accelerated Cooldown," dated November 5, 1990.
5. TVA letter, "Watts Bar Nuclear Plant (WBN) Unit 1 - Technical Requirements Manual (TRM)," dated August 27, 1992 (ADAMS Accession No. ML073230174) including Enclosure 1, "Watts Bar Technical Requirements Manual," (ADAMS Accession No. ML073620391).

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 ~~RCS Vents~~Reactor Vessel Head Vent System

BASES

BACKGROUND

The Reactor Vessel Head Vent System (RVHVS) is installed on the reactor vessel head. The RVHVS consists of a safety-grade venting flow path with redundancy around process solenoid valves. Two one inch solenoid-operated globe valves are mounted in series in each redundant portion of the flow path. The piping between these valves is provided with a temperature monitor. Any leakage through the upstream valve will be detected as an increase in temperature. The two redundant upstream valves are open/close isolation valves and are powered by opposite vital power buses. The two redundant downstream valves are throttle valves that are used to regulate the release rate of the non-condensible gases and steam. The two throttle valves are also powered by opposite vital power buses. All four valves are remote, manual-operated from the control room. The valves are normally closed, de-energized and designed to fail closed in accordance with Regulatory Guide 1.48. The system provides venting during plant startup/shutdown or for post-accident. The system is designed to operate in the containment atmosphere during and after a design basis event. However, the system is not utilized during emergency operation until an inadequate water level in the reactor vessel has been determined. During an incident with hydrogen generation and release, a venting period of approximately ten minutes is acceptable without violating the combustible concentration of hydrogen in the containment.

The capability and the function of the system are consistent with the requirements of Item II.B.1 of NUREG-0737, “Clarification of TMI Action Plan Requirements” (Ref. 1). Direct operator action is required to actuate the system. System actuation is only required when the accumulation of non-condensible gases could impair forced or natural circulation and, hence, cooling of the core.

APPLICABLE SAFETY ANALYSES

The RVHVS is designed to ensure that non-condensible gases do not accumulate under the reactor vessel head and thereby impair the cooling of the core. However, in designing the accident sequences for theoretical hazard evaluation, the RVHVS is not assumed to be a system that directly serves to prevent or mitigate a DBA or transient (Refs. 2 & 3).

(continued)

BASES (continued)

TR

TR 3.4.3 requires that the two redundant vent paths are OPERABLE. One condition for OPERABILITY is that the upstream manual isolation valve is locked open. However, in case of one inoperable path, one condition for continued operation (while restorative actions take place) is that the inoperable path is maintained closed with power removed from both valve actuators. With two paths inoperable, no requirement exists with respect to isolation during the much shorter time of restorative actions.

APPLICABILITY

The TR is basically protecting against uncovering the core and reduces the possibility for impairment of natural or forced circulation through the core. This is mainly a concern during the production of power and early in the decay heat removal phase. Accordingly, Applicability is consistent with operation in MODES 1, 2, 3 and 4. In higher-numbered MODES, the heat flux in the core is low and protection by this TR is not required.

ACTIONS

A.1 and A.2

With one vent path inoperable, it is necessary to immediately start actions to close and fully isolate ~~see to that~~ the inoperable path ~~is closed and fully isolated~~ from the Reactor Coolant System. The inoperable path must be restored to OPERABLE condition in 30 days. It should be noted that during this period of time one path is fully OPERABLE. If the need for venting should occur during this time period, the OPERABLE path will provide 100% of the required venting capacity. Based on this, 30 days is an acceptable time period for restoring the inoperable path.

B.1

With two paths inoperable, it is required to restore one path in 72 hours. The 72 hours is based on operating experience and is a reasonable time period for identifying and correcting problems which could be associated with an inoperable path.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

If the Required Action and associated Completion Time of Condition A or B ~~are~~is not met, the plant must be placed in a condition in which the TR does not apply. This is accomplished by placing the unit in MODE 3 within 6 hours and MODE 5 in an additional 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required conditions from full power conditions in an orderly manner and without challenging plant systems.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.4.3.1, TSR 3.4.3.2 and TSR 3.4.3.3

Every 18 months it is necessary to verify that each of the two vent paths is OPERABLE. This verification consists of checking that the manual isolation valve is locked in the open position~~the upstream and downstream isolation valves and ensuring that the valves are locked in the open position~~. Further, the two open/close isolation valves and the two throttle control valves are operated from the control room, in accordance with the Inservice Testing Program through one complete cycle of full travel. Lastly, the test includes a verification of flow through the two vent paths.

REFERENCES

1. NUREG-0737, “Clarification of TMI Action Plan Requirements.”

2. WCAP-11618, “MERITS Program-Phase II, Task 5, Criteria Application,” including Addendum 1 dated April, 1989, as clarified by Reference 3.

3. TVA letter, “Watts Bar Nuclear Plant (WBN) Unit 1 - Technical Requirements Manual (TRM),” dated August 27, 1992 (ADAMS Accession No. ML073230174) including Enclosure 1, “Watts Bar Technical Requirements Manual,” (ADAMS Accession No. ML073620391).

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 Chemistry

BASES

BACKGROUND

The Reactor Coolant System (RCS) water chemistry is selected to minimize corrosion. A periodic analysis of the coolant chemical composition is performed to verify that the reactor coolant quality meets the specifications (Ref. 1). This Technical Requirement places limits on the dissolved oxygen, chloride and fluoride content of the RCS to minimize corrosion.

Limiting dissolved oxygen content of the RCS limits the amount of general corrosion and reduces the possibility of stress corrosion. General corrosion is a contributing factor in Reactor Coolant Activity (Refs. 2 & 3) and must be controlled for ALARA (as low as reasonably achievable) considerations as well as structural integrity considerations.

Both chlorides and fluorides have been shown to cause stress corrosion if present in the RCS in sufficiently high concentrations at high pressure and temperature conditions. Stress corrosion can lead to either localized leakage or catastrophic failure of the RCS. The associated effects of exceeding the dissolved oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the RCS.

APPLICABLE SAFETY ANALYSES

Minimizing corrosion of the RCS reduces the potential for RCS leakage and for failure due to stress corrosion, thus ultimately ensuring the structural integrity of the RCS (Ref. 3). It is not, however, a consideration in the analyses of Design Basis Accidents.

TR

TR 3.4.4 establishes the limits on concentration of dissolved oxygen, chloride and fluoride in the RCS. These limits ensure that dissolved oxygen, chloride and fluoride concentrations are maintained at levels low enough to prevent unacceptable degradation of the RCS pressure boundary.

(continued)

BASES (continued)

APPLICABILITY

Concentrations of dissolved oxygen, chloride and fluoride in the RCS must be maintained within limits at all times. Applicability is modified by a Note stating with $T_{avg} \leq 250°F$, the dissolved oxygen limit is not applicable.

ACTIONS

A.1

If one or more chemistry parameters are not within Steady State Limits in MODES 1, 2, 3, and 4, the parameter(s) must be restored to its Steady State Limit within 24 hours. This allows time to take corrective actions to restore the contaminant concentrations to within the Steady State Limits.

B.1 and B.2

With one or more chemistry parameters not within Transient Limits in MODES 1, 2, 3, and 4, or if the Required Action of Condition A is not met in the associated Completion Time, the plant must be placed in a condition where the limit is not applicable or where corrosion rates are reduced. This is accomplished by placing the plant in MODE 3 within 6 hours and MODE 5 within 36 hours. In MODE 5, the dissolved oxygen limit is not applicable and stress corrosion rates are reduced. The 6 hours allotted to reach MODE 3 is a reasonable time, based on operating experience, to shutdown the plant from full power in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows 36 hours for restoration of the parameters and to reach MODE 5.

If high chloride or fluoride concentrations are the reason for entering MODE 5, and the condition is not corrected before entering MODE 5, Required Actions C.1, C.2 and C.3 must be performed.

C.1

If RCS chloride or fluoride concentration are not within Steady State Limits for more than 24 hours in any condition other than MODES 1, 2, 3 or 4, or if RCS chloride or fluoride concentration are not within Transient Limits for any amount of time in any condition other than MODES 1, 2, 3 or 4, action must be immediately initiated to reduce pressurizer pressure to $\leq$ 500 psig unless it is already below 500 psig. The immediate Completion Time is consistent with the required times for actions requiring prompt attention.

A Note is added to Condition C stating that all Required Actions must be completed whenever this Condition is entered.

(continued)

BASES

ACTIONS (continued)

C.2 and C.3

In addition to Required Action C.1, an engineering evaluation must be performed to determine the effects of the out-of-limit condition on the structural integrity of the RCS. It must also be determined that the RCS remains acceptable for continued operation. These actions must be taken prior to increasing pressurizer pressure above 500 psig or prior to entry to MODE 4. These evaluations are necessary because of the time/temperature/concentration dependency of the effects of exceeding the limits. Corrosion evaluations for conditions outside the limits are made on a case by case basis.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.4.4.1, 3.4.4.2 and 3.4.4.3

Demonstrating that the chemistry parameters are within their Steady State Limits at a Frequency of 72 hours provides adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action. TSR 3.4.4.1 is modified by a Note stating that it is not required with $T_{avg} \leq 250°F$. With $T_{avg} \leq 250°F$, the dissolved oxygen limit is not applicable.

REFERENCES

1. Watts Bar FSAR, Section 5.2, “Integrity of Reactor Coolant Pressure Boundary.”

2. Watts Bar FSAR, Section 11.1, “Source Terms.”

3. WCAP-11618, “MERITS Program-Phase II, Task 5, Criteria Application,” including Addendum 1 dated April, 1989, as clarified by Reference 4.

4. TVA letter, “Watts Bar Nuclear Plant (WBN) Unit 1 - Technical Requirements Manual (TRM),” dated August 27, 1992 (ADAMS Accession No. ML073230174) including Enclosure 1, “Watts Bar Technical Requirements Manual,” (ADAMS Accession No. ML073620391).

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 Piping System Structural Integrity

BASES

BACKGROUND

Inservice inspection of ASME Code Class 1, 2, and 3 components and pressure testing of ASME Code Class 1, 2, and 3 pumps and valves are performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code (Ref. 1) and applicable Addenda, as required by 10 CFR 50.55a(g) (Ref. 2). Exception to these requirements apply where relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i) and (a)(3). In general, the surveillance intervals specified in Section XI of the ASME Code apply. However, the Inservice Inspection Program includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Code. This clarification is provided to ensure consistency in surveillance intervals throughout the Technical Specifications. Each reactor coolant pump flywheel is, in addition, inspected as recommended in Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975 (Ref. 3).

Additionally, programmatic information on Inservice Inspection is provided in Technical Specifications, Chapter 5.0, Administrative Controls, Section 5.7.2.11, Inservice Inspection Program.

APPLICABLE SAFETY ANALYSES

Certain components which are designed and manufactured to the requirements of specific sections of the ASME Boiler and Pressure Vessel Code are part of the primary success path and function to mitigate DBAs and transients. However, the operability of these components is addressed in the relevant specifications that cover individual components. In addition, this particular Requirement covers only structural integrity inspection/testing requirements for these components, which is not a consideration in designing the accident sequences for theoretical hazard evaluation (Refs. 4 & 5).

(continued)

BASES (continued)

TR

TR 3.4.5 requires that the structural integrity of the ASME Code Class 1, 2, and 3 components in all systems be maintained in accordance with TSR 3.4.5.1 and TSR 3.4.5.2. In those areas where conflict may exist between the Technical Specifications and the ASME Boiler and Pressure Vessel Code, the Technical Specifications take precedence.

APPLICABILITY

The structural integrity of the ASME Code Class 1 components is required in all MODES, when the temperature is above the minimum temperature required by NDT considerations. For ASME Code Class 2 components, the structural integrity is required when the temperature is above 200°F. For ASME Code Class 3 components, the structural integrity is required at all times when the particular component is in service.

ACTIONS

A.1 and A.2

Required Actions A.1 and A.2 apply to ASME Code Class 1 components. Required Action A.1 stipulates that structural integrity should be restored before the temperature of the component is increased more than 50°F above the minimum temperature required by NDT considerations. Alternatively, the component could be isolated before the temperature reaches 50°F above the minimum temperature required by NDT considerations.

B.1 and B.2

Required Actions B.1 and B.2 apply to ASME Code Class 2 components. Required Action B.1 stipulates that structural integrity should be restored before the temperature of the component is increased more than 200°F. Alternatively, the component could be isolated before the temperature reaches 200°F.

C.1, C.2.1, and C.2.2

Required Actions C.1, C.2.1, and C.2.2 apply to ASME Code Class 3 components. Required Action C.1 requires that the applicable Conditions and Required Actions for the affected components be entered immediately. Additionally, the structural integrity of all components must be satisfied or the particular component which does not satisfy the required structural integrity must be isolated from the system within the Completion Time specified in the affected components LCO or TR.

(continued)

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.4.5.1

This surveillance stipulates inspection of the coolant pump flywheel in accordance with Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1. This inspection verifies the structural integrity of the flywheel.

TSR 3.4.5.2

TSR 3.4.5.2 requires the verification of structural integrity of ASME Code Class 1, 2, and 3 components are in accordance with the Inservice Inspection Program.

REFERENCES

1. ASME Boiler and Pressure Vessel Code, "Section XI: Rules for Inservice Inspection of Nuclear Power Plant Components."~~Section XI.~~

2. 10 CFR 50.55a, "Codes and Standards."

3. Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," Revision 1, 1975.

4. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989, as clarified by Reference 5.

5. TVA letter, "Watts Bar Nuclear Plant (WBN) Unit 1 - Technical Requirements Manual (TRM)," dated August 27, 1992 (ADAMS Accession No. ML073230174) including Enclosure 1, "Watts Bar Technical Requirements Manual," (ADAMS Accession No. ML073620391).

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Ice Bed Temperature Monitoring System

BASES

BACKGROUND

The Ice Bed Temperature Monitoring System consists of Resistance Temperature Detectors (RTDs) which are located in various parts of the ice condenser. They serve to verify the attainment of a uniform equilibrium temperature in the ice bed and to detect general gradual temperature rise in the cooling system if breakdown occurs.

Forty-seven RTDs are mounted on ice bed probes which are located throughout the ice bed (i.e., ice bed and floor/air). These 47 RTDs and an additional 38 RTDs which serve to monitor various ice condenser temperatures (i.e., floor cooling piping, wall panels, and wear slab) tie into 2~~a~~ separate temperature recorders (indicator)~~scanner unit~~, located in the Incore Instrumentation Room. The recorders ~~scanner~~ multiplex~~es~~ the ice condenser RTD~~'s~~ signals via an Ethernet connection to ~~a Westronics recorder in~~ the Integrated Computer System (ICS) for Main Control Room indication on the Satellite Display System (SDS). There is one alarm common to the group of 47 RTDs indicating the ice bed temperature is abnormal which is caused by either a high-high, high, or low temperature. There are also six temperature switches located at various points in the ice bed ~~to serve as backup indication should the scanner unit or recorder fail to operate~~that provide an alarm in the MCR ~~. These inputs provide an alarm on the control room annunciator panel~~ should the ice bed temperature exceed preset value (Ref. 1). In addition, the 47 RTDs can be read from the local ice condenser temperature recorder (indicator) located in the Incore Instrumentation Room~~monitoring panel~~.

APPLICABLE SAFETY ANALYSES

The ice condenser is a passive device requiring only maintenance of the ice inventory in the ice bed. As such there are no actuation circuits or equipment which ~~are~~is required for the ice condenser to operate in the event of a Loss of Coolant Accident (LOCA). The Ice Bed Temperature Monitoring System serves only to monitor the ice bed temperature. Since the ice bed has a very large thermal capacity, postulated off-normal conditions can be successfully tolerated for a week to two weeks. Therefore, the Ice Bed Temperature Monitoring System provides an early warning of any incipient ice condenser temperature anomalies. The Ice Bed Temperature Monitoring System is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. Based on the PRA Summary Report (Ref. 2), the Ice Bed Temperature Monitoring System

(continued)

has not been identified as a significant risk contributor.

(continued)

BASES (continued)

TR

TR 3.6.1 states that the Ice Bed Temperature Monitoring System shall be OPERABLE with at least two OPERABLE RTD channels in the ice bed at each of three basic elevations: 10' 6", 30' 9", and 55' above the floor of the ice condenser, for each one-third of the ice condenser.

The OPERABILITY of the Ice Bed Temperature Monitoring System ensures that the capability is available for monitoring the ice bed temperature. The ice bed temperature may be determined at the ~~local~~ ice condenser temperature ~~monitoring~~ recorder (indicator) located in the Incore Instrumentation Room~~panel~~ as well as in the Main Control Room (i.e., Satellite Display System (SDS)) and the Monitoring System would still be considered OPERABLE. In the event the Monitoring System is inoperable, the Required Actions provide assurance that the ice bed heat removal capacity will be retained within the specified time limits.

APPLICABILITY

The Ice Bed Temperature Monitoring System is required to be OPERABLE in MODES 1, 2, 3, and 4. This corresponds to the Applicability requirements for the ice bed in Technical Specification LCO 3.6.11, "Ice Bed."

ACTIONS

A.1

With the ice bed temperature not available in the Main Control Room (i.e., Satellite Display System (SDS)), the ice bed temperature must be determined at the temperature recorder (indicator) located in the Incore Instrumentation Room ~~local ice condenser temperature monitoring panel (local panel)~~ every 12 hours. Since the ice bed has a very large thermal capacity, postulated off-normal conditions can be successfully tolerated for one or two weeks. Therefore, a 12 hour surveillance of the ice bed temperature will give sufficient warning of any incipient ice condenser temperature anomalies.

B.1.1, B.1.2, and B.1.3

With the Ice Bed Temperature Monitoring System inoperable and being unable to determine the ice bed temperature at the temperature recorder (indicator)~~local panel~~, Required Actions B.1.1, B.1.2, and B.1.3 require verification that: the ice compartment lower inlet doors, intermediate deck doors, and top deck doors are closed; the last recorded mean ice bed temperature was less than or equal to 20°F (value does not account for instrument error) and steady; and the Ice Condenser Cooling System is OPERABLE.

(continued)

BASES

ACTIONS

B.1.1, B.1.2, and B.1.3 (continued)

The Completion Time of 1 hour and every 12 hours thereafter to perform Required Actions B.1.1 and B.1.3 and 1 hour to perform Required Action B.1.2 is reasonable and based upon the typical time necessary to perform the Required Actions. These Required Actions, along with the high thermal capacity of the ice bed, ensure that the ice bed will remain below critical temperatures while the Monitoring System is inoperable.

B.2.1 and B.2.2

With the Ice Bed Temperature Monitoring System inoperable and being unable to determine the ice bed temperature at the temperature recorder (indicator)~~local panel~~, either the Monitoring System or the temperature recorder (indicator) ~~local monitoring panel~~ must be restored to OPERABLE status within 30 days. A Completion Time of 30 days is given, provided that Required Actions B.1.1, B.1.2, and B.1.3 are met. These Required Actions, along with the high thermal capacity of the ice bed, ensure that the ice bed will remain below critical temperatures during the 30 day Completion Time. Also, the six alarmed temperature switches (which provide an alarm at 25°F) will continue to monitor the ice bed temperature. If the Ice Condenser Cooling System becomes inoperable before the Ice Bed Temperature Monitoring System is OPERABLE, then Required Action C must be performed.

C.1.1 and C.1.2

With the Ice Bed Temperature Monitoring System inoperable and being unable to determine the ice bed temperature at the temperature recorder (indicator) ~~local panel~~ and with the Ice Condenser Cooling System not satisfying the minimum components OPERABILITY requirements of Required Action B.1.3, Required Actions C.1.1 and C.1.2 require verification that: the ice compartment lower inlet doors, intermediate deck doors, and top deck doors are closed; and that the last recorded mean ice bed temperature was less than or equal to 15°F (value does not account for instrument error) and steady. The Completion Time of 1 hour and every 12 hours thereafter to perform Required Action C.1.1 and 1 hour to perform Required Action C.1.2 is reasonable and based upon the typical time necessary to perform the Required Actions. These Required Actions, along with the high thermal capacity of the ice bed, ensure that the ice bed will remain below critical temperatures while the Ice Bed Temperature Monitoring System, temperature recorder (indicator), and Ice Condenser Cooling System are inoperable.

(continued)

BASES

ACTIONS (continued)

C.2.1, C.2.2, and C.2.3

With the Ice Bed Temperature Monitoring System inoperable and being unable to determine the ice bed temperature at the temperature recorder (indicator) ~~local panel~~ and with the Ice Condenser Cooling System not satisfying the minimum components OPERABILITY requirements of Required Action B.1.3, the Ice Condenser Cooling System, Ice Bed Temperature Monitoring System or the temperature recorder (indicator) ~~local temperature monitoring panel~~ must be restored to OPERABLE status. A Completion Time of 6 days is given, provided that Required Actions C.1.1 and C.1.2 are met. These Required Actions, along with the high thermal capacity of the ice bed, ensure that the ice bed will remain below critical temperatures during the 6 day Completion Time. Also, the six alarmed temperature switches (which provide an alarm at 25°F) will continue to monitor the ice bed temperature.

D.1 and D.2

If the Required Action and associated Completion Time of Condition A, B or C is not met, the integrity of the ice bed may be threatened. Therefore, the plant must be placed in a MODE in which the TR does not apply. This is done by placing the plant in MODE 3 in 6 hours and MODE 5 in 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.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.6.1.1

Performance of a CHANNEL CHECK on the Ice Bed Temperature Monitoring System 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 instrument channels could be an indication of excessive instrument drift in one of the channels or of even something more serious. The Surveillance Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Thus, TSR 3.6.1.1 ensures that loss of function will be identified within 12 hours.

(continued)

BASES (continued)

REFERENCES

1. Watts Bar FSAR, Section 6.7.15, “Ice Condenser Instrumentation.”

2. WCAP-11618, “MERITS Program-Phase II, Task 5, Criteria Application,” including Addendum 1 dated April, 1989, as clarified by Reference 3.

3. TVA letter, “Watts Bar Nuclear Plant (WBN) Unit 1 - Technical Requirements Manual (TRM),” dated August 27, 1992 (ADAMS Accession No. ML073230174) including Enclosure 1, “Watts Bar Technical Requirements Manual,” (ADAMS Accession No. ML073620391).

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Inlet Door Position Monitoring System

BASES

BACKGROUND

Ninety-six limit switches monitor the position of the lower inlet doors. Two switches are mounted on the door frame for each door panel. The position and movement of the switches are such that the doors must be effectively sealed before the switches are actuated. A single annunciator window in the control room gives a common alarm signal when any door is open. Open/shut indication is also provided at the lower inlet door position display panel located in the Main Control Room. For door monitoring purposes, the ice condenser is divided into six zones, each containing four inlet door assemblies, or a total of eight door panels. The limit switches on the doors in any single zone are wired to a single light on the inlet door position display panel such that a closed light indicates that all the doors in that zone are shut and an open light indicates that one or more doors in that zone are open (Ref. 1). The display panel is considered the Inlet Door Position Monitoring System.

Monitoring of inlet door position is necessary because the inlet doors form the barrier to air flow through the inlet ports of the ice condenser for normal unit operation. Failure of the Inlet Door Position Monitoring System requires an alternate OPERABLE monitoring system to be used to ensure that the ice condenser is not degraded.

APPLICABLE SAFETY ANALYSES

Proper operation of the inlet doors is necessary to mitigate the consequences of a loss of coolant accident or a main steam line break inside containment. The Inlet Door Position Monitoring System, however, is not required for proper operation of the inlet doors, nor is it considered OPERABLE as an initial condition for a DBA. Hence, the Inlet Door Position Monitoring System is not a consideration in the analyses of DBAs. Based on the PRA Summary Report in References 2 and 3, the Inlet Door Position Monitoring System has not been identified as a significant risk contributor.

(continued)

BASES (continued)

TR

The Inlet Door Position Monitoring System provides the only direct means of determining that the inlet doors are shut. Since an open door would allow heat input that could cause sublimation and mass transfer of ice in the ice condenser compartment, the Inlet Door Position Monitoring System must be OPERABLE whenever the ice bed is required to be OPERABLE. This ensures early detection of an inadvertently opened or failed door, allowing prompt action before ice bed degradation can occur.

APPLICABILITY

The Inlet Door Position Monitoring System is required to be OPERABLE in MODES 1, 2, 3 and 4. This corresponds to the Applicability requirements for the ice bed.

ACTIONS

A.1 and A.2

If the Inlet Door Position Monitoring System is inoperable in MODE 1, an alternate OPERABLE monitoring system must be used to ensure that the ice condenser is not degraded. This is done by confirming the Ice Bed Temperature Monitoring System is OPERABLE with the ice bed temperature ≤ 27°F (value does not account for instrument error). This Action must be completed within 4 hours and each 4 hours thereafter. The Frequency of 4 hours is based on the fact that temperature changes cannot occur rapidly in the ice bed because of the large mass of ice involved. Since this is an indirect means of monitoring inlet door position, operation in MODE 1 may continue for a maximum of 14 days in this condition. If the ice bed temperature increases to above 27°F, the ice bed must be declared inoperable in accordance with Technical Specification 3.6.11, “Ice Bed.”

B.1

If the Required Action and associated Completion Time for Condition A are not met or if the Inlet Door Position Monitoring System is inoperable in MODES 2, 3, or 4, the Inlet Door Position Monitoring System must be restored to OPERABLE status within 48 hours. The 48 hour Completion Time is based on the fact that, with the very large mass of ice involved, it would not be possible for the temperature to increase to the melting point and a significant amount of ice to melt in a 48 hour period.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

If the Required Action and associated Completion Time of Condition B cannot be met, the plant must be placed in a condition where OPERABILITY of the Inlet Door Position Monitoring System is not required. This is accomplished by placing the plant in 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 MODES from full power in an orderly manner and without challenging plant systems.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.6.2.1

Performance of the CHANNEL CHECK for the Inlet Door Position Monitoring System 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 of something even more serious. Performance of the CHANNEL CHECK helps to ensure that the instrumentation continues to operate properly between each TADOT. The dual switch arrangement on each door allows comparison of open and shut indicators for each zone as well as a check with the annunciator window. An alternate to the use of the annunciator window as the channel check, is to perform a continuity check of the same circuit used by the annunciator window. This continuity check will confirm if one or more inlet door zone switch contacts are closed which would represent an open inlet door. The Surveillance Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Thus, TSR 3.6.2.1 ensures that loss of function will be identified within 12 hours.

TSR 3.6.2.2

TSR 3.6.2.2 is the performance of a TADOT every 18 months. It checks trip devices (limit switches) that provide actuation signals directly. The 18 month Frequency was developed considering the plant conditions needed to perform TSR 3.6.2.2. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.6.2.3

TSR 3.6.2.3 requires verification that the monitoring system correctly indicates the status of each inlet door as the door is opened and reclosed during its Technical Specification testing. This provides ongoing operational testing of the indicating system. The Frequency coincides with the Technical Specifications testing performed.

REFERENCES

1. Watts Bar FSAR, Section 6.7, "Ice Condenser System."

2. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989, as clarified by Reference 3.

3. TVA letter, "Watts Bar Nuclear Plant (WBN) Unit 1 - Technical Requirements Manual (TRM)," dated August 27, 1992 (ADAMS Accession No. ML073230174) including Enclosure 1, "Watts Bar Technical Requirements Manual," (ADAMS Accession No. ML073620391).

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Lower Compartment Cooling (LCC) System

BASES

BACKGROUND

The Lower Compartment Cooling (LCC) fans provide non-safety related cooling for the lower compartment spaces after all accidents, except those that initiate a Phase B Containment Isolation Signal (Ref. 1), when the non-safety related cooling coils and cooling water supply are available. LCC fans perform a safety related air recirculation function in the lower containment pocketed (dead ended) spaces after a main steam line break (MSLB) to prevent the formation of localized hot spots which could exceed the qualification temperatures of equipment required to operate post accident. The LCC fans are not required to operate during or after a loss of coolant accident (LOCA).

After an MSLB, one LCC train will be manually started a minimum of 1 1/2 hours, but less than 4 hours, after the accident to ensure that the dead ended compartment temperatures are kept below the environmental qualification limit. Each train consists of two 50% capacity fans, backdraft damper, instrumentation and controls, and associated ductwork. Each train is powered from separate class 1E power sources.

APPLICABLE SAFETY ANALYSES

The LCC fans recirculate air in the lower compartment spaces after an MSLB. Under these circumstances, the intact Reactor Coolant System piping will serve as a long term heat source. After the ice is melted, the heat from the Reactor Coolant System (RCS) will result in a gradual temperature increase in the sub-compartments of the lower containment. The LCC System may be used to provide long term containment cooling after a MSLB to assure electrical equipment qualification requirements are maintained. ~~If the recirculation of air should fail during or after the accident, the Containment Spray System and Air Return Fan System can be started to provide the necessary containment cooling. The temperatures in the sub-compartments of the lower containment are not input to the safety analyses. Containment area temperatures have not been identified as significant risk contributors~~.

(continued)

BASES (continued)

TR

The TR specifies the equipment which needs to be OPERABLE in order to ensure that air can be circulated in the sub-compartments if an accident should take place. At least one LCC train must be placed in operation after the accident. The LCC fans do not perform a cooling function, which means that the coils and the secondary cooling water circuits need not be OPERABLE. However, secondary side failures which could impair the operation of the fans and the circulation of the air must be prevented.

APPLICABILITY

The flow of air to the sub-compartments is necessary following an MSLB where the RCS represents a major heat source in lower containment. Based on the temperature of the RCS, this could be in MODES 1 through 4. In MODES 5 and 6, the probability for an accident is small and, in any case, the heat capacity of the RCS is limited and, therefore, not able to heat up the lower compartment spaces to such an extent that equipment could degrade. The specification is therefore only applicable in MODES 1, 2, 3 and 4.

ACTIONS

A.1

With one fan inoperable, the inoperable fan must be restored to OPERABLE status within 7 days. During this period, the remaining three fans are available to circulate the air in the lower compartments of the containment. However, only two fans are necessary to prevent the hot spots. Hence, there is one spare fan available. The 7 day Completion Time is based on the low probability of an event requiring emergency fan operation, the availability of one fan more than required, and the availability of alternate cooling means.

B.1

With two fans inoperable the plant has still adequate fan capacity to circulate air if an accident should take place. However, in this case no spare capacity is available. Hence, it is required to restore at least one fan to OPERABLE status within 72 hours. With one fan restored, three fans will be OPERABLE and Condition A must be entered. This will allow another 7 days to restore the last inoperable fan to OPERABLE status. The 72 hour Completion Time in Condition B is based on the low probability of an event requiring fan operation simultaneous with further fan deterioration, and the availability of alternate cooling means.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

If the Required Actions of A.1 and B.1 cannot be completed within the required Completion Time or if more than two fans are inoperable, the plant must be placed in a MODE in which the TR does not apply. This is done by placing the plant in at least MODE 3 within 6 hours and in MODE 5 within an additional 30 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.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.6.3.1

During normal operation, three of the four LCCs operate to remove heat from the lower compartments of the containment. This means that three of the four fans are operating at all times. Hence, this gives certainty that at least three fans are OPERABLE. The test is for the fan that has not been in operation during the preceding 31 days and to verify that all fans can be manually started from the control room. The 15 minutes running test is optional for fans that have been running the previous 31 days, or will be running after the Surveillance has been carried out.

REFERENCES

1. Watts Bar FSAR, Section 6.2.2, “Containment Heat Removal Systems.”

2. 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.”

B 3.7 PLANT SYSTEMS

B 3.7.1 Steam Generator Pressure / Temperature Limitations

BASES

BACKGROUND

In order to meet regulatory and code requirements with respect to material toughness, certain limits on steam generator pressure and temperature are established. Material toughness varies with temperature and is lower at room temperature than at operating temperature. One indicator of the temperature effect on ductility is the nil-ductility temperature (NDT). Therefore, a nil-ductility reference temperature (RT_{NDT}) has been determined by experimental means. The RT_{NDT} is that temperature below which brittle (non-ductile) fracture may occur. For the steam generators, the RT_{NDT} has been determined to be 10°F ~~for steam generators A, B, and C and 30°F for steam generator D~~ (Ref. 1). Considering uncertainties and proper margins, the minimum operating temperature has been determined to be 70°F. The 70°F temperature must be established before the pressure is increased to 200 psig. This limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits.

The fracture mechanic methodology, which is used to determine the stresses and material toughness, follows the guidance given by 10 CFR 50, Appendix G (Ref. 2). Reference 1 mandates the use of ASME Boiler and Pressure Vessel Code, Section III, Appendix G (Ref. 4).

APPLICABLE SAFETY ANALYSES

The RT_{NDT} limit is not derived from the Design Basis Accident analyses. The RT_{NDT} limit is imposed during normal operation to avert encountering pressure/temperature combinations which are not analyzed as part of the steam generator design. Unanalyzed pressure/temperature combinations could cause propagation of minor, undetected flaws, which could cause brittle failure of the pressure boundary. Because the RT_{NDT} limit is related to normal operation, the RT_{NDT} limit is not a consideration in designing the accident sequences for theoretical hazard evaluations (Ref. 5 & 6).

(continued)

BASES (continued)

TR

TR 3.7.1 requires that the pressures on the primary and the secondary sides in the steam generator are kept at or below 200 psig when the temperature is less than or equal to 70°F. The pressure induced stress from the 200 psig pressure is low enough to be insignificant, even at temperatures at or below RT_{NDT}.

APPLICABILITY

The operating requirements which must be observed to avoid a condition, which could lead to brittle failure, are not strictly limited to specific MODES. Hence, in general, Applicability should be At All Times. However, in practice it is unlikely that these limits will be violated in the lower numbered MODES, due to the high operating temperature on the primary as well as the secondary side in the steam generators. Accordingly, the limits are most easily violated at low temperature, during shutdown and startup of the plant. Applicability can therefore conveniently be limited to whenever the temperature on the primary or the secondary side is at or below 70°F. Since cooldown would make the steam generator more susceptible to brittle failure, a Note to the Applicability has been added to preclude cooldown in the primary or secondary to ≤ 70°F when the pressure is > 200 psig.

ACTIONS

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

With the combination of pressure and temperature not within limits, a reduction in pressure to ≤~~<~~ 200 psig is required within 30 minutes. An engineering evaluation must be performed to determine the effect on the structural integrity of the pressure boundary. The evaluation must be finished and the conclusion made that no hazard exists before the temperature is increased to more than 200°F. Condition A is modified by a Note which states that whenever Condition A is entered, ~~all~~ ACTIONS A.1 and A.2~~through A.3~~ must be completed.

(continued)

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.1.1

TSR 3.7.1.1 verifies that the pressures on the primary and the secondary sides in the steam generators are less than 200 psig (value does not account for instrument error). At temperatures below 70°F (value does not account for instrument error), the temperature margin to RT_{NDT} is diminished. Hence, the pressure must be checked every hour to ensure that the material toughness criteria are not violated. The 1 hour Frequency is based on engineering judgment and is consistent with industry practice.

Note: Instrument uncertainty has been considered in establishing these values and is discussed in this Bases section under Background.

REFERENCES

1. WCAP-13146, "Technical Basis for Determination of Secondary Side Pressure Test Temperatures in Sequoyah and Watts Bar Units 1 and 2 Steam Generators."

2. 10 CFR 50, Appendix G, "Fracture Toughness Requirements."

3. Not used.

4. ASME Boiler and Pressure Code, Section III.

5. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989, as modified by Reference 6.

6. TVA letter, "Watts Bar Nuclear Plant (WBN) Unit 1 - Technical Requirements Manual (TRM), dated August 27, 1992 (ADAMS Accession No. ML073230174) including Enclosure 1, "Watts Bar Technical Requirements Manual," (ADAMS Accession No. ML073620391).

B 3.7 PLANT SYSTEMS

B 3.7.2 Flood Protection Plan

BASES

BACKGROUND

Nuclear power plants are designed to prevent the loss of capability for cold shutdown and maintenance thereof resulting from the most severe flood conditions that can reasonably be predicted to occur at the site as a result of severe hydrometeorological conditions, seismic activity, or both (Ref. 1). Assurance that safety-related facilities are capable of surviving all possible flood conditions is provided by the flood protection plan.

The elevations of plant features which could be affected by the submergence during floods vary from 739.2~~08.014.5~~ ft Mean Sea Level (MSL) (excluding wave runup) ~~(access to electrical conduits)~~ to 741.~~0~~ 7 ft MSL (including wave runup). Plant grade is elevation 728.0 ft MSL which can be exceeded by extreme rainfall floods and closely approached by seismic-caused dam failure floods. A warning plan is needed to assure plant safety from floods.

The warning plan is divided into two stages. This two-stage plan is designed to allow adequate time for preparing the plant for operation in the flood mode and to avoid excessive economic loss in case a potential flood does not fully develop. Stage I warning, which is a minimum of 10 hours, allows preparation steps, causing some damage to be sustained, but will postpone major economic damage. Stage II warning, which is a minimum of 17 hours, is a warning that a forthcoming flood above grade is predicted.

Stage I procedures consist of a controlled reactor shutdown and other easily revokable steps, such as moving flood supplies above the probable maximum flood elevation and making temporary connections and load adjustments on the onsite power supply. After unit shutdown, the Reactor Coolant System will be cooled and the pressure will be reduced to less than 350 psig. Stage II procedures are the least easily revokable and more damaging steps necessary to have the plant in the flood mode when the flood exceeds plant grade. Heat removal from the steam generators will be accomplished by adding river water from the Fire Protection System, and relieving steam to the atmosphere through the steam generator power operated relief valves. Other essential plant cooling loads will be transferred from the Component Cooling Water System to the Essential Raw Cooling Water System (ERCW); the ERCW will also replace the Raw Cooling Water System to the ice condensers. The Radioactive Waste System will be secured by filling tanks below Design Bases Flood (DBF) level with enough water to prevent floatation;

(continued)

BASES

BACKGROUND (continued)

one exception is the waste gas decay tanks, which are sealed and anchored against floatation. Power and communication lines running beneath the DBF that are not required for submersed operation will be disconnected, and batteries below the DBF will be disconnected (Ref. 2).

APPLICABLE SAFETY ANALYSES

The flood protection plan specifies flood control measures to protect safety related equipment in the event that the maximum elevation for the ultimate heat sink or other body of water, as applicable, is exceeded. Because external flooding conditions present substantial warning time to achieve plant shutdown, this requirement is not a contributor to a dominant risk sequence (Ref. 3 & 4).

TR

TR 3.7.2 requires that the flood protection plan be ready for implementation to maintain the plant in a safe condition. This requirement ensures that facility protective actions will be taken and operation will be terminated in the event of flood conditions.

APPLICABILITY

The flood protection plan TR is applicable when one or more of the following conditions exist:

a. Extreme flood producing rainfall conditions in the east Tennessee watershed, or

b. Receipt of notification from TVA River Operations (RO) that predicted flood levels based on rainfall on the ground or potential failure problems with one or more dams combined with critical headwater elevations and flood-producing rainfall may result in subsequent issuance of a Stage I flood warning.

(continued)

BASES (continued)

ACTIONS

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

If a Stage I flood warning is issued, several actions are required to be taken. The first requires the plant to be placed in MODE 3 in 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 plant systems.

Upon issuance of a Stage I flood warning, initiate and complete the Stage I flood protection plan, which involves preparatory steps. The required Completion Time for this Required Action is 10 hours. The plant is also required to be brought from full power operation to a safe shutdown. This is accomplished by Required Action A.3. This Required Action requires the establishment of a SHUTDOWN MARGIN of at least 5% $\Delta k/k$ and T_{avg} less than or equal to 350°F. The Completion Time of 10 hours is reasonable to accomplish the required SHUTDOWN MARGIN and T_{avg}.

A.4.1 and A.4.2

Once a Stage I flood warning has been issued, it is necessary to maintain communications between the TVA RO Group and the Watts Bar Nuclear Plant. This is necessary because the TVA RO Group provides the flood forecasting for the Watts Bar Nuclear Plant. The Completion Time of 10 hours corresponds to the time specified to initiate and complete the Stage I flood protection plan.

If communications between the TVA RO Group and the Watts Bar Nuclear Plant have not been established within the required Completion Time, the Stage II flood protection procedure must be initiated and completed within 27 hours. The Completion Time of 27 hours corresponds to the minimum pre-flood preparation time. This is to ensure adequate warning time for safe plant shutdown.

B.1

If the Stage II flood warning has been issued, the Stage II flood protection plan must be initiated and completed within 17 hours or prior to flooding of the site. The Completion Time of 17 hours corresponds to the remaining hours of the 27 hour pre-flood preparation time after the Stage I flood warning consisting of 10 hours has expired, and is an adequate time period to complete Stage II preparations.

At any time it is determined that the potential for flooding at the site does not exist, the Stage I and Stage II flood protection plans are to be terminated immediately.

(continued)

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.2.1

This surveillance requires communications between Watts Bar Nuclear Plant and TVA RO Group be established and maintained every 3 hours. A Note for this surveillance states that this is required only when one of the applicability criteria is met. This communications requirement exists because the TVA RO Group provides the flood forecasting for Watts Bar Nuclear Plant. The 3 hour Frequency is adequate for early flood forecasting.

REFERENCES

1. Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants."

2. Watts Bar FSAR, Section 2.4.14, "Flooding Protection Requirements."

3. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989, as modified by Reference 4.

4. TVA letter, "Watts Bar Nuclear Plant (WBN) Unit 1 - Technical Requirements Manual (TRM), dated August 27, 1992 (ADAMS Accession No. ML073230174) including Enclosure 1, "Watts Bar Technical Requirements Manual," (ADAMS Accession No. ML073620391).

B 3.7 PLANT SYSTEMS

B 3.7.3 Snubbers

BASES

BACKGROUND

Component standard supports, are those metal supports which are designed to transmit loads from the pressure-retaining boundary of the component to the building structure. Although classified as component standard supports, snubbers require special consideration due to their unique function. Snubbers are either operated hydraulically or mechanically, depending on the nature of the support needed. They are designed to provide no transmission of force during normal plant operations, but function as a rigid support when subjected to dynamic transient loadings. Therefore, snubbers are chosen in lieu of rigid supports where restricting thermal grow during normal operation would induce excessive stresses in the piping nozzles or other equipment. The location and size of the snubbers are determined by stress analysis. Depending on the design classification of the particular piping, different combinations of load conditions are established. These conditions combine loading during normal operation, seismic loading and loading due to plant accidents/transients to four different loading sets. These loading sets are denominated: normal, upset, emergency, and faulted condition. The actual loading included in each of the four conditions, depends on the design classification of the piping. The calculated stresses in the piping and other equipment, for each of the four conditions, must be in conformance with established design limits.

Supports for pressure-retaining components are designed in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, Division 1 (Ref. 1). The combination of loadings for each support, including the appropriate stress levels, meet the criteria of Regulatory Guide 1.124, “Design Limits and Loading Combinations for Class 1 Linear-Type Component Supports” (Ref. 2), and Regulatory Guide 1.130, “Design Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports” (Ref. 3).

APPLICABLE SAFETY ANALYSES

Pipe and equipment supports, in general, are not directly considered in designing the accident sequences for theoretical hazard evaluations. Further, various Probabilistic Risk Assessment (PRA) studies have indicated that snubbers are not of prime importance in a risk significant sequence (Ref. 4 and 5). Therefore, the function of the snubbers is not essential in mitigating the consequences of a DBA or transient (Ref. 6).

(continued)

BASES (continued)

TR

TR 3.7.3 requires that all snubbers ~~utilized on safety related equipment~~ shall be OPERABLE. ~~Those snubbers that are utilized on non-safety related systems shall be OPERABLE if a failure could have adverse effect on a safety related system.~~ Individual snubbers may be removed from service for functional testing within the limits established herein without violating these requirements, although Required Actions and Completion Times still apply.

APPLICABILITY

The OPERABILITY of the snubbers is required in MODES 1, 2, 3, and 4. For MODES 5 and 6, the OPERABILITY is limited to the snubbers located on those systems which need to be OPERABLE in MODES 5 and 6.

ACTIONS

A.1~~.1, A.1.2, and A.2~~

If one or more snubbers are removed from service, the supported system must be~~have been~~ declared inoperable immediately and the appropriate LCO entered for that system.~~, the snubber(s) must be restored to OPERABLE status in 72 hours. Alternatively, the snubber(s) must be replaced in the 72 hours. In either case, an engineering evaluation per Table 3.7.3-5 must be performed during the 72 hours to:~~

B.1 and B.2

If one or more required snubbers are discovered inoperable, then the supported system must be declared inoperable immediately and the appropriate LCO entered for that system.

Discovery of any snubber as inoperable requires the performance of an engineering evaluation per Table 3.7.3-5 during the 72 hours to:

a) Determine the cause of the failure

As a result of this evaluation, the need for testing other snubbers will be considered. The results from the testing will be used to consider expanded functional testing and cause examination with consideration of manufacturing and design deficiency. It should be noted that the testing must be independent and not combined with TSR 3.7.3.3.

(continued)

BASES (continued)

B.1 and B.2 (continued)

b) Determine the impact on the supported component

This evaluation shall determine if the inoperable snubber has adversely affected the attached component.

The 72 hours is based on engineering experiences and is reasonable, considering the time it will take to identify the problem and take the proper corrective actions. This requirement is considered met for those snubbers rendered inoperable by removal for functional testing by the generic engineering evaluation included in Reference 9.

Another alternative is to perform an engineering evaluation to demonstrate inoperable snubber(s) do not impact the OPERABILITY of the supported system for the existing plant condition (Reference 10).

C.1

If Required Actions under Condition A or Condition B are not met within the required Completion Time, a Service Request shall be written to generate a Problem Evaluation Report for the occurrence.

(continued)

BASES

~~ACTIONS (continued)~~

~~A.3~~

~~Another alternative is to perform an engineering evaluation to demonstrate inoperable snubber(s) do not impact the OPERABILITY of the supported system for the existing plant condition (Reference 10).~~

~~B.1~~

~~If Required Actions under Condition A are not met within the 72 hours, the supported system or component is immediately declared inoperable.~~

TECHNICAL SURVEILLANCE REQUIREMENTS

The TSRs are preceded by three Notes. Note 1 states that the snubber inservice inspection program shall be carried out in accordance with the requirements in Tables 3.7.3-1, 3.7.3-2 and 3.7.3-3. This represents an enhanced snubber inservice inspection program compared to the Inservice Inspection Program which stipulates inservice inspection in accordance with ASME Section XI. The snubber inservice inspection program includes the requirements of Generic Letter 90-09, "Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions." ASME Section XI, 1989 Edition, Subsections IWF-5300(a) and (b) require that inservice examinations of snubbers, using the VT-3 visual examination method described in IWA-2213, and inservice tests of snubbers be performed in accordance with the first Addenda to ASME/ANSI OM-1987, Part 4. Note 2 requires repair or replacement of snubbers which fail inspection, and testing of repaired snubbers before installation. Note 3 indicates that a "snubber type," as used in this TR, is determined by the design and manufacturer, but not by size.

TSR 3.7.3.1

TSR 3.7.3.1 comprises a visual inspection of the snubbers. A pre-fuel load visual inspection and functional test has been performed on each snubber using the acceptance criteria listed in this TSR. The baseline considers that the snubbers have experienced thermal cycling and normal operating service as a result of previous hot functional testing. The initial inservice inspection must be performed on the snubbers prior to completion of the first refueling outage. The frequency of subsequent surveillances depends on the number of snubbers found inoperable from each previous inspection as provided in Table 3.7.3-2 and the Inservice Inspection Program. The acceptance criteria and the remedial ACTIONS are listed in Table 3.7.3-1.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.3.1 (continued)

The visual inspections are designed to detect obvious indications of inoperability of the snubbers. Removal of insulation or direct contact with the snubbers is not required initially. However, suspected causes of inoperability are to be investigated and all snubbers of the same type and all snubbers subjected to the same failure mode are to be inspected more frequently.

Until proven otherwise by functional testing, mechanical and hydraulic snubbers are considered OPERABLE, unless they are disconnected at either end, experienced gross deformation of the snubber or structural support, or the hydraulic fluid level is empty for hydraulic snubbers.

The visual inspection frequency is based upon the number of unacceptable snubbers found during the previous inspection. Therefore, the required inspection intervals vary inversely with the number of inoperable snubbers found during an inspection. If a snubber fails the visual acceptance criteria, the snubber is declared unacceptable and cannot be declared OPERABLE via functional testing. However, if the cause of rejection is understood and remedied for that type of snubber and for any other type of snubbers, that may be generically susceptible, and OPERABILITY verified by testing, that snubber may be reclassified acceptable for the purpose of establishing the next surveillance interval.

Snubbers may be categorized, according to accessibility, as noted in the Note to Table 3.7.3-2. The accessibility of each snubber is determined based on radiation level as well as other factors such as temperature, atmosphere, location, etc. The recommendations of Regulatory Guide 8.8, “Information Relevant to Maintaining Occupational Radiation Exposure as Low as Practicable,” (Ref. 7) and Regulatory Guide 8.10, “Operation Philosophy for Maintaining Occupational Radiation Exposure as Low as Practicable,” (Ref. 8), are considered in planning and implementing the visual inspection program.

TSR 3.7.3.2

TSR 3.7.3.2 comprises the inspection of all snubbers attached to systems that have experienced unexpected, potentially damaging transients. The potential impact of the transients is assessed by reviewing operating data and by visually inspecting the associated systems. The review and the inspection must be performed within six months of the event. In addition to the inspection, the freedom-of-motion of the mechanical snubber(s) is verified in accordance with Table 3.7.3-3.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.7.3.3

TSR 3.7.3.3 comprises the functional testing of hydraulic and mechanical snubbers. The testing for these snubbers have been separated into two sample plans. Sample Plan A (10%) is used for the hydraulic steam generator snubbers based on the small population. Sample Plan B (37 snubbers) is used for mechanical snubbers based on the large population. The plans, when used in combination, are a conservative approach versus using only the Sample Plan B for the entire population.

Snubber functional testing is performed prior to completion of each refueling outage. This frequency is based on engineering experience and is reasonable for testing of a representative sample of snubbers. Credit may be taken toward meeting minimum outage testing requirements for mechanical snubbers functionally tested within the refueling cycle. Snubbers may be removed from service for functional testing in ~~Modes~~ MODES 1 through 4 provided that the following administrative controls are implemented:

1. Required Actions and Completion Times must be met.

2. No more than one snubber may be removed from service at a time on any line and attached piping which is analyzed as a seismic subsystem. Multiple snubbers may be removed for testing simultaneously only if separated by two or more seismic anchors.

3. Snubbers on trained systems or portions of systems may be removed only on the train which is undergoing maintenance in that work week. Snubbers on non-trained systems or portions of systems may only be removed following a documented risk assessment. Snubbers may not be removed from service for testing on one train of a system while the other train has been declared inoperable for any reason.

4. Snubbers adjacent to equipment nozzles may not be removed for testing except in ~~Modes~~ MODES 5 and 6. In determining the applicability of this limitation, engineering judgment must be used regarding the placement of the snubber relative to the nozzle, the routing of the affected piping, and any other supports available to protect equipment function.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.7.3.4

The TSR is preceded by three Notes which underline the need for considering service life of sub-components and to replace these sub-components before the end of the respective service lives. The replacement of sub-components must be documented and the documentation retained for further reference. TSR 3.7.3.4 addresses the monitoring of the service life of the snubbers. The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. The expected service life is established by the manufacturer and is based on operating experience with critical snubber parts such as seals and springs in a radiation environment. The every refueling outage Frequency is based on engineering experience and is reasonable for the verification service life.

REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III and XI.
2. Regulatory Guide 1.124, “Design Limits and Loading Combinations for Class 1 Linear-Type Component Supports.”
3. Regulatory Guide 1.130, “Design Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports.”
4. “Zion Probabilistic Safety Study,” Commonwealth Edison Company, September 1981.
5. “Millstone Unit 3 Probabilistic Safety Study,” North-east Utilities Company, August 1983.
6. NRC Staff Review of Nuclear Steam Supply System Vendor Owners Groups' Application of The Commission's Interim Policy Statement Criteria to Standard Technical Specifications, Attachment to letter dated May, 1988 from T. E. Murley, NRC to W. S. Wilgus, Chairman The B&W Owners Group.
7. Regulatory Guide 8.8, “Information Relevant to Maintaining Occupational Radiation Exposure as Low as Practicable.”
8. Regulatory Guide 8.10, “Operating Philosophy for Maintaining Occupational Radiation Exposure as Low as Practicable.”
9. SA/SE WBPLCE-97-028-0, RIMS T28970829803.
10. Screening Review WBPLCE-02-003-0.

(continued)

BASES

REFERENCES (continued)

11. Screening Review WBPLCE-10-009-0.

12. American Society of Mechanical Engineers (ASME), Code for the Operation and Maintenance of Nuclear Power Plants (ASME OM Code), 2004 Edition through 2006 Addenda.

B 3.7 PLANT SYSTEMS

B 3.7.4 Sealed Source Contamination

BASES

BACKGROUND	A sealed source is any byproduct, source, or special nuclear material that is encased in a capsule designed to prevent leakage or escape of the material (Refs. 1 & 4). Sealed sources are classified into three groups according to their use (sources in use, not in use, and startup sources and fission detectors) and may contain alpha, beta, gamma, or neutron emitting material. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on References 2 & 4. Those sources that are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring, excore fission detector assemblies or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.
APPLICABLE SAFETY ANALYSES	The sealed source contamination requirement ensures that leakage from sealed sources will not exceed allowable intake values. This TR is important to the safety of plant personnel, however it is not required to mitigate the consequences of a DBA or transient (Refs. 3 & 4).
TR	TR 3.7.4 requires that the removable contamination shall be less than 0.005 microcuries for each sealed source containing the following radioactive material: a. Greater than 100 microcuries of beta and/or gamma emitting material; or b. Greater than 5 microcuries of alpha emitting material.
APPLICABILITY	Since the limits on the removable contamination for each sealed source containing radioactive material are not MODE dependent, this TR is applicable at all times.

(continued)

BASES (continued)

ACTIONS

Since this TR is applicable at all times, the Required Actions have been modified by a Note stating that the provisions of TR 3.0.3 do not apply.

A.1

With a sealed source having removable contamination in excess of the limits, the sealed source should be withdrawn from use immediately. The immediate Completion Time reflects the importance of preventing the contamination from spreading.

A.2.1 and A.2.2

If the sealed source contamination is not within the specified limit and the sealed source has been removed from use, the sealed source must be decontaminated and repaired, otherwise, disposal of the sealed source is required. If the sealed source is to be decontaminated and repaired, it must be done prior to returning the sealed source to use. If disposal of the sealed source is to be done, it must be completed in accordance with NRC regulations.

TECHNICAL SURVEILLANCE REQUIREMENTS

Notes have been added to this section stating that the licensee or other persons specifically authorized by the NRC shall perform the TSRs, and that the test methods used shall have a detection sensitivity of greater than or equal to 0.005 microcurie per test sample.

TSR 3.7.4.1

This surveillance determines every 6 months that the removable contamination is less than 0.005 microcuries for each sealed source. The 6 month Frequency is frequent enough to identify a leaking or contaminated sealed source without having extensive spreading of contamination.

This surveillance is modified by several Notes. The Notes state that this TSR is only applicable to sources in use, to sources with half-lives of more than 30 days, and to sources in any form other than gas. Also, this TSR is not applicable to startup sources and fission detectors previously subjected to core flux.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.7.4.2

This surveillance determines within 6 months prior to use or transfer to another licensee that the removable contamination is less than 0.005 microcuries for each sealed source and fission detector. This Frequency is adequate to identify a leaking or contaminated sealed source or fission detector to avoid extensive contamination.

This surveillance is modified by two Notes. The first Note states that this TSR is only applicable to sealed sources not in use. The second Note states that sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed in use.

TSR 3.7.4.3

This surveillance determines that the removable contamination is less than 0.005 microcuries for each startup source and fission detector. This test should be performed on each startup source and incore fission detector within 31 days prior to being installed in the core or being subjected to core flux. It also should be performed following any repairs or maintenance to the source. This Frequency ensures that the startup source or fission detector is not leaking or contaminated over the specified limit.

This Surveillance is modified by a Note stating this TSR only applies to startup sources and incore fission detectors that are not in use.

TSR 3.7.4.4

This surveillance determines that the removable contamination is less than 0.005 microcuries for each excore fission detector. This test should be performed on each excore fission detector assembly within 31 days prior to the detector assembly being installed in its permanent configuration. It also should be performed following any repairs or maintenance to the detector assembly. This frequency ensures that the excore detectors are not leaking or contaminated over the specified limit.

This Surveillance is modified by a Note stating this TSR only applies to excore fission detectors.

(continued)

BASES

REFERENCES

1. 10 CFR 70.4 “Definitions,” as clarified by Reference 4.

2. 10 CFR 70.39 “Specific Licenses for the Manufacture or Initial Transfer of Calibration or Reference Sources,” as clarified by Reference 4.

3. WCAP-11618, “MERITS Program-Phase II, Task 5, Criteria Application,” including Addendum 1 dated April, 1989, as clarified by Reference 4.

4. TVA letter, “Watts Bar Nuclear Plant (WBN) Unit 1 - Technical Requirements Manual (TRM), dated August 27, 1992 (ADAMS Accession No. ML073230174) including Enclosure 1, “Watts Bar Technical Requirements Manual,” (ADAMS Accession No. ML073620391).

B 3.7 PLANT SYSTEMS

B 3.7.5 Area Temperature Monitoring

BASES

BACKGROUND

Thermal-life of various electrical and mechanical equipment is one of several important aging concerns in the qualification of hardware. The requirement is that the equipment remains functional during and after specified design basis events. Design basis events consist of loss of offsite power and design basis accidents (DBA). In general, the following three groups of hardware are subjected to qualification:

a. Safety related equipment,

b. Non-safety related equipment (failure of which could prevent safety related equipment to operate as designed), and

c. Specific post-accident monitoring equipment.

The normal service temperatures of concern are relatively low, hence, most of the equipment requiring consideration are components in the electrical power supply and the instrumentation systems. Some of these components are designed for relatively low temperature with very little margin to normal operating temperatures in cabinets and buildings. The procedure for thermal qualification is normally to subject prototypes from the production line to life tests by natural or artificial (accelerated) aging to its end-of-installed life condition. Analyses with justifications of methods and assumptions are used to qualify the prototypes to the actual service conditions, which may differ from the test conditions. Although the equipment is qualified for an environment expected after a DBA, the components are only subjected to normal operating conditions for most of the design life. Therefore, the thermal aging due to normal operating conditions is of major importance and is the parameter which is controlled by the Technical Requirements. Accordingly, this particular requirement establishes temperature limits during normal operation for specific locations in various buildings, except the containment. The temperature limits are related to the expected thermal-life for the hardware which operates in the areas where the temperatures are monitored and controlled.

(continued)

BASES

BACKGROUND (continued)

Due to valve design, ambient temperatures can affect the setpoints of the main steam safety valves (MSSVs), whereby a decrease in valve body temperature causes an increase in setpoint, resulting in non-conservative relief pressure. Ambient temperatures are monitored within the main steam valve vaults to ensure that the MSSVs minimum temperatures are maintained to meet the 1% code allowable variance on setpoints. Detailed BASES for the MSSVs is provided in Technical Specification B 3.7.1.

The general guidelines, which are followed for the qualification of electrical equipment, are provided in 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants" (Ref. 1). Detailed requirements for the implementation of the general guidelines are provided in various Regulatory Guides and IEEE Standards. Basic requirements for the qualification of mechanical equipment are outlined in General Design Criteria 4 (Ref. 2).

APPLICABLE SAFETY ANALYSES

Certain components, which have the service temperatures controlled by this requirement, are part of the primary success path and function to mitigate DBAs and transients. However, the integrity/OPERABILITY of these components is addressed in the relevant specifications that cover individual components. The service temperatures and the thermal aging, which are controlled by observing the requirements of this TR, are not inputs to the safety analysis. Further, Probabilistic Risk Assessment studies, performed to date, do not explicitly model the function of area temperature monitors. In addition, this particular requirement covers only service temperatures and thermal aging of these components, which are not considerations in designing the accident sequences for theoretical hazard evaluations (Refs. 3 & 4).

TR

TR 3.7.5 provides nominal temperature limits in the vicinity of major equipment. The TR allows for each area shown in Table 3.7.5-1 to be higher or lower than the normal limit for a maximum of eight hours. Note that the temperature values listed in Table 3.7.5-1 do not account for instrument error.

APPLICABILITY

The limits on temperature and time apply whenever the affected equipment in an affected area is required to be OPERABLE.

(continued)

BASES (continued)

ACTIONS

A.1

Whenever the temperature in one or more areas has exceeded the normal temperature limits for more than eight hours, document the exceedance in accordance with the Corrective Action Program. The report must contain the cumulative time and the amount by which the temperature has exceeded the limits.

Condition A has been modified by a Note stating that the provisions of TR 3.0.3 do not apply.

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

Whenever the temperature in one or more areas has exceeded the abnormal temperature limits, the temperature must be restored to within the normal limits in 4 hours. The Completion Time of 4 hours is based on operator experience and is a reasonable time for restoring the temperature. Alternatively, the affected equipment must be declared inoperable and the inoperability documented in accordance with the Corrective Action Program along with the cumulative time and the amount by which the temperature has exceeded the limits. In addition, an analysis shall be prepared which demonstrates OPERABILITY of the affected equipment.

C.1 and C.2

Whenever the temperature in the Intake Pumping Station mechanical or electrical equipment rooms exceeds the lower limit of 40°F, actions must be initiated within 24 hours to ensure the temperature does not decrease below 32 °F. The 24 hour Completion Time hours is based on temperature analysis. Within 7 days, restore normal temperatures within the areas affected. The 7 day Completion Time is based on a reasonable repair duration, and compensatory actions available during the interim period to maintain temperatures above 32°F.

(continued)

BASES

ACTIONS (continued)

D.1 and D.2

If the temperature in the Intake Pumping Station mechanical or electrical equipment rooms decreases to 32°F or lower, the affected equipment must be immediately declared inoperable. The Completion Time is based on potential freezing of safety-related components. The inoperability of the equipment must be documented in the Corrective Action Program along with the cumulative time and amount by which the temperature has exceeded the limits. In addition, an analysis shall be submitted which demonstrates OPERABILITY of the affected equipment.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.5.1

The temperatures for the areas listed in Table 3.7.5-1 must be determined every 12 hours to ensure compliance with the limits. The 12 hour Frequency is based on engineering experience and is reasonable considering the time required for performing the surveillance and the probability for changes in the area temperatures. Note that the temperature values listed in Table 3.7.5-1 do not account for instrument error.

REFERENCES

1. 10 CFR 50.49 "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."

2. 10 CFR 50, Appendix A, General Design Criteria 4, "Environmental and Dynamic Effects Design Bases."

3. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989, as clarified by Reference 4.

4. TVA letter, “Watts Bar Nuclear Plant (WBN) Unit 1 - Technical Requirements Manual (TRM), dated August 27, 1992 (ADAMS Accession No. ML073230174) including Enclosure 1, “Watts Bar Technical Requirements Manual,” (ADAMS Accession No. ML073620391).

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 Isolation Devices

BASES

BACKGROUND

The onsite Class 1E AC and DC electrical power distribution system is divided by trains into two redundant and independent AC and DC electrical power distribution subsystems. Each AC and DC electrical power distribution subsystem is comprised of 6.9kV AC shutdown boards, 480V AC shutdown boards, associated motor control centers, and 120V AC power distribution panels, 120V AC vital buses, and 125V DC vital buses. Two trains (or subsystems) are required for safety function redundancy; any one train provides safety function, but without worst-case single-failure protection.

Because of the safety significance of the Class 1E AC and DC electrical power distribution subsystems and the equipment that they supply, unique requirements for OPERABILITY are imposed on these subsystems beyond those requirements applicable to non-qualified AC and DC distribution subsystems. As such, 1E buses must be protected from faults that could occur on loads not included as part of the Class 1E system, associated nonqualified cables routed in Class 1E cable trays or nonqualified cables insufficiently separated from Class 1E cables. Circuit breakers actuated by fault currents are used as isolation devices in this plant to detect and isolate faults. The OPERABILITY of these circuit breakers ensures that the Class 1E buses will be protected in the event of faults in nonqualified loads powered by the buses, located in associated nonqualified cables routed in Class 1E cable trays or in nonqualified cables insufficiently separated from Class 1E cables.

APPLICABLE SAFETY ANALYSES

The initial conditions of design basis transient and accident analyses in FSAR Chapter 6, “Engineered Safety Feature,” and Chapter 15, “Accident Analyses” (Ref. 1) assume ESF Systems are OPERABLE. The Class 1E AC and DC electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF Systems so that the fuel, Reactor Coolant System (RCS) and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Technical Specifications 3.2 (Power Distribution Limits), 3.4 (Reactor Coolant System), and 3.6 (Containment Systems).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The OPERABILITY of the AC and DC electrical power distribution systems is consistent with the initial assumptions of the accident analyses (Ref. 1) and is based upon meeting the design basis of the plant. This includes maintaining at least one train of the onsite or offsite AC electrical power sources, DC electrical power sources, and associated distribution systems OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite power or all onsite AC electrical power; and

b. A worst-case single failure.

Isolation devices help ensure the OPERABILITY of Class 1E AC and DC electrical power distribution systems by protecting them from faults on the non-Class 1E portion of the distribution system, on associated nonqualified cables routed in Class 1E cable trays, or on nonqualified cables insufficiently separated from Class 1E cables. However, these devices are not a structure, system, or component that is part of the primary success path and which actuates to mitigate a DBA or transient that either assumes a failure of or presents a challenge to the integrity of a fission product barrier (Ref. 2).

TR

TR 3.8.1 requires that all circuit breakers actuated by fault currents that are used as isolation devices protecting Class 1E buses from non-qualified loads, associated circuits or insufficiently separated cables shall be OPERABLE. These breakers are identified on Drawing Series 45A710 (Ref. 3). This Technical Requirement satisfies testing specified in Sections 8.3.3.3 (2) and 8.3.3.3 (3) of the Safety Evaluation Report (Ref. 4). The OPERABILITY of these devices helps ensure that the Class 1E subsystem will be protected from faults that occur on the non-Class 1E portion of the distribution system.

APPLICABILITY

The Class 1E AC and DC electrical distribution systems are required to supply power to those systems necessary to mitigate the consequences of DBAs or transients that could occur in MODES 1, 2, 3, or 4. Isolation devices are therefore required to protect the Class 1E distribution systems in these MODES.

(continued)

BASES (continued)

ACTIONS

A.1, A.2.1, and A.2.2

With one or more of the required circuit breakers inoperable, the Class 1E distribution system is not isolated from faults on non-Class 1E portions of the distribution system, on non-Class 1E associated cables routed in Class 1E cable trays, or on Non-Class 1E cables insufficiently separated from Class 1E cables.

Action must be taken to restore this isolation. One possible solution is to restore the circuit breaker(s) to OPERABLE status. If this cannot be done, the isolation can be achieved manually by tripping or removing the inoperable circuit breaker(s). Removing the inoperable breaker(s) ensures that they will not be inadvertently closed before they can be restored to OPERABLE status. The Completion Time of 8 hours takes into consideration the low probability of a fault occurring on the distribution system, on an associated non-Class 1E circuit or on an insufficiently separated non-Class 1E cable, concurrent with an event requiring the safety systems supplied by the Class 1E system. It also represents a reasonable time to repair or trip (or remove) the inoperable circuit breaker(s).

To ensure that the inoperable circuit breaker(s) are not inadvertently re-energized before they are returned to OPERABLE status, it is necessary to periodically verify that they remain tripped or removed. The period of 7 days takes into consideration the unlikelihood that a plant operation or maintenance activity would result in the re-energization of these breaker(s) from the de-energized condition.

B.1 and B.2

If the Required Action and associated Completion Time of Condition A cannot be met, the Class 1E system remains unprotected from faults on non-Class 1E portions of the distribution system, on non-Class 1E associated cables routed in Class 1E cable trays or on non-Class 1E cables insufficiently separated from Class 1E cables. Since this condition cannot be allowed for an extended period of time, it is necessary to place the plant in a condition where the isolation devices are not required to be OPERABLE. This is done by placing the plant in MODE 3 in 6 hours and then in MODE 5 in the next 30 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.

(continued)

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.8.1.1

This surveillance requires the performance of a functional test on a representative sample of ≥ 10% of each type of molded-case circuit breaker used as an isolation device. This sample size is sufficiently large to represent the actual failure distribution within the whole population of circuit breakers of a given type used in the plant. If there ~~are~~ is any failure mechanisms that could affect the OPERABILITY of the circuit breaker(s) they are likely to have occurred in the sample tested. The 18 month Frequency takes into consideration the infrequent operation of the breakers and their correspondingly low failure rate. The Surveillance is augmented by three Notes. The first Note states that the breakers shall be selected for testing on a rotating basis. This ensures that all of the breakers will eventually be tested and those failures that may not have been discovered in the initial 10% samples will eventually be discovered.

The second Note describes the functional test procedure and the response to be verified to ensure OPERABILITY.

The third Note states that for each molded case circuit breaker found inoperable during functional tests an additional representative sample of 10% of the defective type shall be functionally tested until no more failures are found or all of that type have been functionally tested. This helps to ensure that a failure discovered in the representative sample was not caused by a failure mechanism that could systematically affect other breakers in the overall population of breakers of the same type.

TSR 3.8.1.2

This surveillance requires the performance of a functional test on a representative sample of ≥ 10% of each type of electrically-operated circuit breaker used as an isolation device. This sample size is sufficiently large to represent the actual failure distribution within the whole population of circuit breakers of a given type used in the plant. If there are any failure mechanisms that could affect the OPERABILITY of the circuit breaker(s), they are likely to have occurred in the sample tested. The 18 month Frequency takes into consideration the infrequent operation of the breakers and their correspondingly low failure rate.

The Surveillance is augmented by three Notes. The first Note states that the breakers shall be selected for testing on a rotating basis. This ensures that all of the breakers will eventually be tested and those failures that may not have been discovered in the initial 10% samples will eventually be discovered. The second Note describes the functional test procedure and the response to be verified to ensure OPERABILITY.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.8.1.2 (continued)

The third Note states that for each electrically-operated circuit breaker found inoperable during functional tests an additional representative sample of 10% of the defective type shall be functionally tested until no more failures are found or all of that type have been functionally tested. This helps to ensure that a failure discovered in the representative sample was not caused by a failure mechanism that could systematically affect other breakers in the overall population of breakers of the same type.

TSR 3.8.1.3

This surveillance requires that the performance of a CHANNEL CALIBRATION of all protective relays associated with medium voltage (6.9 kV) isolation overcurrent devices. A CHANNEL CALIBRATION assures that the relays will be able to detect overcurrent conditions on the non-Class 1E loads. The Frequency of 18 months is consistent with the typical industry refueling cycle.

TSR 3.8.1.4

This surveillance requires the performance of an integrated system functional test which verifies that the relays and associated medium voltage (6.9 kV) circuit breakers function as designed to isolate fault currents. An integrated test assures that the individual elements of the protection scheme, the relays, breakers and other control circuits, interact as designed.

The surveillance has been modified by a Note stating that if a failure is discovered in the integrated functional test, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be tested to assure that there is no common cause failure mechanism that could systematically affect all breakers of a given type.

The Frequency of 18 months coincides with the typical industry refueling cycle.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.8.1.5

This surveillance requires the inspection of each circuit breaker and the performance of procedures prepared in conjunction with the manufacturer's recommendations. By performance of recommended maintenance, the likelihood for the circuit breakers to become inoperable can be minimized. The 72 months periodicity for Class 1E circuit breaker, (Ref. 5), takes into consideration the low frequency of operation of the circuit breakers and the low likelihood that operation and maintenance activities at the plant could adversely affect the OPERABILITY of the circuit breaker.

REFERENCES

1. Watts Bar FSAR, Section 6.0, “Engineered Safety Feature,” and Section 15.0, “Accident Analyses.”

2. TVA letter, “Watts Bar Nuclear Plant (WBN) Unit 1 - Technical Requirements Manual (TRM), dated August 27, 1992 (ADAMS Accession No. ML073230174) including Enclosure 1, “Watts Bar Technical Requirements Manual,” (ADAMS Accession No. ML073620391).~~Not Used.~~

3. Watts Bar Wiring Diagram Series 45A710, “Periodic Breaker Test.”

4. NUREG-0847, “Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2” including Supplements thereto.

5. EPRI NP-7410-V3, “Molded Case Circuit Breaker Application and Maintenance Guide,” Revision 1.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 Containment Penetration Conductor Overcurrent Protection Devices

BASES

BACKGROUND

General Design Criterion (GDC), "Containment Design Basis," of 10 CFR 50, Appendix A requires, in part, that the reactor containment structure be designed so that the containment structure can, without exceeding design leakage rate, accommodate the calculated pressure, temperature, and other environmental conditions resulting from any loss-of-coolant accident. One consideration in meeting the requirements of this GDC is the design of electrical penetrations.

Reference 1 describes a method of complying with GDC Appendix A with respect to the requirements for design, qualification, construction, installation and testing of electric penetration assemblies. It specifies that the electric penetration assembly should be designed to withstand, without loss of mechanical integrity, the maximum short-circuit current vs. time conditions that could occur given single random failures of circuit overload protection devices.

The function of electrical protective devices is to detect and isolate faults that could occur on the electrical distribution system. These devices therefore provide an effective means of preventing fault currents from challenging the design limit of the penetrations. Containment penetration conductor overcurrent protective devices are installed to further protect the penetration conductors from faults on components inside containment or improper operation of other protective devices in addition to that provided by the distribution system.

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment includes the requirement that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate. The design of the electrical penetrations must therefore provide that they remain intact in the event of faults on components inside containment or penetration conductors that supply these components. The containment penetration conductor overcurrent protective devices provide additional fault protection of the penetrations and help ensure that the design limits of the penetrations are not challenged. However, these overcurrent protective devices are not a structure, system, or component that is part of the primary success path and which actuates to mitigate a DBA or transient that either assumes a failure of or presents a challenge to the integrity of a fission product barrier (Refs. 2 & 5).

(continued)

BASES (continued)

TR

TR 3.8.2 requires that all containment penetration conductor overcurrent protection devices be OPERABLE. These protection devices are identified on Drawing Series 45A710 (Ref. 3). This assures that the design limits of the containment electrical penetrations will not be challenged as a result of electrical faults on the penetration conductors or the equipment that they supply in containment.

APPLICABILITY

The OPERABILITY of the containment penetration conductor overcurrent protection devices is required when the containment is required to be OPERABLE. In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced because of the pressure and temperature limitations of these MODES. The containment penetration conductor overcurrent protection devices are, therefore, required to be OPERABLE in MODES 1, 2, 3, and 4.

ACTIONS

A.1, A.2.1, A.2.2 and A.2.3

With one or more containment penetration conductor overcurrent protection devices inoperable, the circuit(s) associated with the inoperable protection device(s) must be placed in a condition that would preclude the possibility of a fault that could overload the circuit(s). To accomplish this, the circuit is de-energized by either tripping the circuit's backup circuit breaker or by removing the inoperable circuit breaker. Since systems or components supplied by the affected circuit will no longer have power, they must be declared inoperable.

The 72 hour Completion Time takes into account the design of the electrical penetration for maximum fault current, the availability of backup circuit protection on the distribution system and the low probability of a DBA occurring during this period. This Completion Time is also considered reasonable to perform the necessary repairs or circuit alterations to restore or otherwise de-energize the affected circuit.

In order to assure that any electrical penetration which is not protected by an overcurrent device remains de-energized, it is necessary to periodically verify that its backup circuit breaker is tripped or that the inoperable circuit breaker is removed. A Completion Time of 7 days is considered sufficient due to the infrequency of plant operations that could result in reenergizing a circuit that has been de-energized in this manner.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the inoperable containment penetration conductor overcurrent protection devices are not able to be restored to OPERABLE status and the associated circuit cannot be de-energized within 72 hours, the containment penetration is vulnerable to the mechanical effects of a short circuit, should one occur. These effects can challenge the design capability of the penetration and therefore pose a threat to containment integrity. To protect against this possibility, the plant must be placed in a condition where the TR is not applicable. This is done by placing the plant in MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are considered reasonable based on operating experience to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

TECHNICAL SURVEILLANCE REQUIREMENTS

As described by Technical Surveillance Requirements general surveillance Note 1, the surveillances for this TR are necessary to assure that the overcurrent protection devices given in Drawing Series 45A710 (excluding fuses) are demonstrated OPERABLE. Note 2 explains that the surveillance requirements apply to at least one Reactor Coolant Pump (RCP) such that all RCP circuits are demonstrated OPERABLE at least once per 72 month period. This recognizes the importance of the RCP circuits to the safe operation of the plant as well as the potentially large amount of short circuit current associated with a fault on these circuits.

TSR 3.8.2.1

This surveillance requires the performance of a CHANNEL CALIBRATION of all protective relays associated with medium voltage (6.9 kV) containment penetration overcurrent devices. A CHANNEL CALIBRATION assures that the relays will be able to detect overcurrent conditions on the penetration conductors. The Frequency of 18 months is consistent with the typical industry refueling cycle.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.8.2.2

This surveillance requires the performance of an integrated system functional test which verifies that the relays and associated medium voltage (6.9 kV) circuit breakers function as designed to isolate fault currents. An integrated test assures that the individual elements of the protection scheme, the relays, breakers and other control circuits, interact as designed.

The surveillance has been modified by a Note stating that if a failure is discovered in the integrated functional test, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be tested to assure that there is no common cause failure mechanism that could systematically affect all breakers of a given type.

The Frequency of 18 months coincides with the typical industry refueling cycle.

TSR 3.8.2.3

This surveillance requires the performance of a functional test on a representative sample of ≥ 10% of each type of molded-case circuit breaker used as penetration protection. This sample size is sufficiently large to represent the actual failure distribution within the whole population of circuit breakers of a given type used in the plant. If there are any failure mechanisms that could affect the OPERABILITY of the circuit breaker(s) they are likely to have occurred in the sample tested. The 18 month Frequency takes into consideration the infrequent operation of the breakers and their correspondingly low failure rate. The Surveillance is augmented by three Notes. The first Note states that the breakers shall be selected for testing on a rotating basis. This ensures that all of the breakers will eventually be tested and those failures that may not have been discovered in the initial 10% samples will eventually be discovered.

The second Note describes the functional test procedure and the response to be verified to ensure OPERABILITY.

The third Note states that for each molded case circuit breaker found inoperable during functional tests an additional representative sample of 10% of the defective type shall be functionally tested until no more failures are found or all of that type have been functionally tested. This helps to ensure that a failure discovered in the representative sample was not caused by a failure mechanism that could systematically affect other breakers in the overall population of breakers of the same type.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.8.2.4

This surveillance requires the performance of a functional test on a representative sample of ≥ 10% of each type of electrically-operated circuit breaker used as penetration protection. This sample size is sufficiently large to represent the actual failure distribution within the whole population of circuit breakers of a given type used in the plant. If there are any failure mechanisms that could affect the OPERABILITY of the circuit breaker(s), they are likely to have occurred in the sample tested. The 18 month Frequency takes into consideration the infrequent operation of the breakers and their correspondingly low failure rate.

The Surveillance is augmented by three Notes. The first Note states that the breakers shall be selected for testing on a rotating basis. This ensures that all of the breakers will eventually be tested and those failures that may not have been discovered in the initial 10% samples will eventually be discovered. The second Note describes the functional test procedure and the response to be verified to ensure OPERABILITY.

The third Note states that for each electrically-operated circuit breaker found inoperable during functional tests an additional representative sample of 10% of the defective type shall be functionally tested until no more failures are found or all of that type have been functionally tested. This helps to ensure that a failure discovered in the representative sample was not caused by a failure mechanism that could systematically affect other breakers in the overall population of breakers of the same type.

TSR 3.8.2.5

This surveillance requires the inspection of each circuit breaker and the performance of preventive maintenance in accordance with procedures prepared in conjunction with the manufacturers recommendation. Performance of recommended preventive maintenance helps ensure the operability of the circuit breakers. The 72 months periodicity for Class 1E and 96 months for non-Class 1E circuit breakers (Ref. 4) takes into consideration known failure rates for the circuit breakers and operating experience.

(continued)

BASES (continued)

REFERENCES

1. Regulatory Guide 1.63, “Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants,” Revision 3.

2. WCAP-11618, “MERITS Program-Phase II, Task 5, Criteria Application,” including Addendum 1 dated April, 1989, as clarified by Reference 5.

3. Watts Bar Wiring Diagram Series 45A710, “Periodic Breaker Test.”

4. EPRI NP-7410-V3, “Molded Case Circuit Breaker Application and Maintenance Guide,” Revision 1.

5. TVA letter, “Watts Bar Nuclear Plant (WBN) Unit 1 - Technical Requirements Manual (TRM), dated August 27, 1992 (ADAMS Accession No. ML073230174) including Enclosure 1, “Watts Bar Technical Requirements Manual,” (ADAMS Accession No. ML073620391).

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 Submerged Component Circuit Protection

BASES

BACKGROUND

Electrical equipment located inside containment has been designed to maintain equipment safety functions and to prevent unacceptable spurious actuations. All power cables feeding equipment inside containment are provided with individual breakers to protect the power sources (both Class 1E and non-Class 1E) from the effects of electrical shorts. Reactor coolant pumps have two circuit breakers. All other power cables are provided with a cable protector fuse which, in the event of a breaker failure, is designed to protect the containment penetration. These breakers and protector fuses ensure that, should an electrical short occur inside containment, the electrical power source will not be affected.

A failure analysis has been made on the ability of the electrical power (both AC and DC) systems to withstand failure of submerged electrical components from the postulated LOCA flood levels inside containment (Ref. 1 and 5). Some of the identified components are automatically de-energized in event of a LOCA. The remaining components that are powered from a Class 1E source were assumed to have a high impedance fault for the analysis. The magnitude of the leakage currents used in the analysis is the maximum value of current that each protective device would carry for an indefinite period (i.e., the protective device's thermal rating). The results of the evaluations indicate that the submergence of electrical components will not prevent the Class 1E electric (either AC or DC) systems from performing their intended safety function for the postulated submerged condition.

A listing of major electrical components located inside containment that may be inundated following a LOCA is found in Reference 2 along with an explanation of the safety significance of the failure of the equipment due to flooding. These components are automatically de-energized by the accident signal and the accident signal must be reset to remove the automatic trip signal from each component.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The Accident Analysis (Ref. 3) assumes the availability of the Engineered Safeguards Features to mitigate the consequences of a DBA or transient. The safety design basis of the containment includes the requirement that the containment must withstand the limiting DBA without exceeding the design leakage rate. Both of these requirements depend upon the actuation of motors and valves to perform their safety function. An electrical or mechanical failure on a submerged component has the potential to interfere with the ability of some other safety component or system to perform its intended function. By de-energizing their associated component on accident conditions, submerged component circuits minimize the potential for this type of interference with safety functions. They are not, however, systems or components that are part of the primary success path and which actuate to mitigate a DBA or transient that either assumes a failure of or presents a challenge to the integrity of a fission product barrier (Refs. 4 & 6).

TR

TR 3.8.4 requires that all submerged component circuits associated with valves 2-FCV-74-1, 2-FCV-74-2, 2-FCV-74-8, and 2-FCV-74-9 shall be de-energized and with each component listed in Table 3.8.4-1 shall be OPERABLE. The OPERABILITY of the submerged component circuits ensures that electrical or mechanical faults on submerged components will not interfere with the ability of other safety related equipment, or the Class 1E distribution, to perform its safety function.

APPLICABILITY

Electrical or mechanical faults on valves 2-FCV-74-1, 2-FCV-74-2, 2-FCV-74-8, and 2-FCV-74-9, and the components listed in Table 3.8.4-1 could potentially affect systems or components necessary to mitigate the consequences of DBAs or transients that could occur in MODES 1, 2, 3, or 4. The submerged component circuits are therefore required to be OPERABLE during these MODES in order to de-energize potentially submerged components.

(continued)

BASES (continued)

ACTIONS

A.1

With the circuits for valves 2-FCV-74-1, 2-FCV-74-2, 2-FCV-74-8 and 2-FCV-74-9 energized with RCS pressure ≥≥ 425 psig or with one or more submerged components circuits (Table 3.8.4-1) inoperable, the associated submerged components could remain energized in the event of an accident. In order to prevent the adverse effects of a potential fault on an energized submerged component during an accident, it is necessary to restore the ability to automatically de-energize the component under accident conditions or to maintain a de-energized configuration for components not required to be functional. This can be done by restoring the inoperable circuit to OPERABLE status. The Completion Time of 7 days takes into consideration the low probability of an accident occurring which would cause the components to be submerged. It is a reasonable amount of time to complete the work necessary to restore the circuits to OPERABLE status.

B.1 and B.2

If the submerged component circuits cannot be restored to OPERABLE status within the 7 day Completion Time, it is necessary to place the plant in a Condition where the function of the circuits is not needed. This can be accomplished by first placing the plant in MODE 3 and then in MODE 5. The Completion Time of 6 hours to reach MODE 3 and 36 hours to reach MODE 5 are considered to be reasonable times for placing the plant into a condition where the TR is not applicable in a controlled manner.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.8.4.1

This surveillance requires verification that valves 2-FCV-74-1, 2-FCV-74-2, 2-FCV-74-8, and 2-FCV-74-9 are de-energized by removal of power at the 480V motor control centers. These valves are required to be shut in MODES 1, 2, 3, and 4, and are interlocked so that they cannot be opened until RCS Pressure is reduced to < 425 psig. The Frequency of 31 days is considered reasonable based on plant operating experience.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.8.4.2

This surveillance requires verification that the components shown in Table 3.8.4-1 are automatically de-energized on a simulated accident signal. Since the function of OPERABLE submerged component circuits for the valves shown in the Table is to de-energize the components under accident conditions, verification that the valves are, in fact, de-energized on a simulated accident signal also constitutes verification that the submerged component circuits are OPERABLE. The 18 month Frequency corresponds to the availability of the components for testing during plant refueling.

REFERENCES

1. Watts Bar FSAR, Section 8.3.1.2.3, “Safety-Related Equipment in a LOCA Environment.”

2. Watts Bar FSAR, Table 8.3-28, “Major Electrical Equipment That Could Become Submerged Following a LOCA.”

3. Watts Bar FSAR, Section 15.0, “Accident Analyses.”

4. WCAP-13470, “Watts Bar Unit 1 Technical Specifications Criteria Application Report,” dated August, 1992, as clarified by Reference 6.

5. Watts Bar Electrical Calculation EDQ00299920080020.

6 TVA letter, “Watts Bar Nuclear Plant (WBN) Unit 1 - Technical Requirements Manual (TRM), dated August 27, 1992 (ADAMS Accession No. ML073230174) including Enclosure 1, “Watts Bar Technical Requirements Manual,” (ADAMS Accession No. ML073620391).

B 3.9 REFUELING OPERATIONS

B 3.9.1 ~~Decay Time~~Delete

~~BASES~~

~~BACKGROUND~~	~~Three analyses of a postulated fuel handling accident are performed: 1) a realistic analysis, 2) a conservative analysis, and 3) an analysis based on Regulatory Guide 1.25 (Ref. 1). Both the conservative analysis and the Regulatory Guide 1.25 analysis assume that the accident occurs 100 hours after plant shutdown. Radioactive decay of the fission product inventory during the interval between shutdown and placement of the first spent fuel assembly into the spent fuel pit is taken into account.~~ ~~It is also necessary to consider a fuel handling accident occurring inside the primary containment. The assumption that the accident occurs 100 hours after plant shutdown is also applicable to this analysis (Ref. 2).~~
~~APPLICABLE SAFETY ANALYSES~~	~~The minimum requirement of 100 hours of reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is an initial condition of a Postulated Fuel Handling Accident. Therefore, Reference 3 concludes that this requirement should be retained as a revised Technical Specification. However, in subsequent discussions with the NRC Staff, it was concluded that decay time was not strictly a process variable, and should be removed from the revised Technical Specifications.~~
~~TR~~	~~TR 3.9.1 requires the reactor to be subcritical for at least 100 hours. Implicit in this TR is the Applicability (during movement of irradiated fuel in the reactor vessel). This ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products, thus reducing the fission product inventory and reducing the effects of a fuel handling accident.~~
~~APPLICABILITY~~	~~This TR is applicable only during movement of irradiated fuel in the reactor vessel. Therefore, it effectively prohibits movement of irradiated fuel in the reactor vessel during the first 100 hours following reactor shutdown.~~

(continued)

~~BASES (continued)~~

~~ACTIONS~~

~~A.1~~

~~With the reactor subcritical less than 100 hours, all movement of irradiated fuel in the reactor vessel must be suspended. As stated above, movement of irradiated fuel in the reactor vessel is prohibited during the first 100 hours following reactor shutdown.~~

~~TECHNICAL SURVEILLANCE REQUIREMENTS~~

~~TSR 3.9.1.1~~

~~Since movement of irradiated fuel in the reactor vessel is prohibited during the first 100 hours following reactor shutdown, a verification of time subcritical must be made prior to movement of irradiated fuel in the reactor vessel. This is done by confirming the date and time of subcriticality and verifying that at least 100 hours have elapsed. The Frequency of "Prior to movement of irradiated fuel in the reactor vessel" ensures that the TR is met before irradiated fuel is moved in the reactor vessel.~~

~~REFERENCES~~

~~1. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."~~

~~2. Watts Bar FSAR, Section 15.5.6, "Environmental Consequences of a Postulated Fuel Handling Accident."~~

~~3. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.~~

B 3.9 REFUELING OPERATIONS

B 3.9.3 Refueling Machine

BASES

BACKGROUND

The refueling machine is used during CORE ALTERATIONS to either move fuel assemblies to new positions in the core, load new fuel assemblies, or unload spent fuel assemblies. The refueling machine consists of a rectilinear bridge and trolley crane with a vertical mast extending down into the refueling water. The bridge and trolley motions are used to position the vertical mast over a fuel assembly in the core. A long tube with a pneumatic gripper on the end is lowered down out of the mast to grip the fuel assembly and manipulate it so that it can be transported to its new position.

The refueling machine has two auxiliary monorail hoists which are located on each side of the bridge. The auxiliary hoists are used for the movement of control rod drive shafts in order to facilitate the refueling process. Before using the hoist, the drive shafts must be disconnected from their respective control rods and, with the upper internals, removed from the vessel (Ref. 1).

APPLICABLE SAFETY ANALYSES

This requirement ensures that the refueling machine and auxiliary hoists have sufficient load capacity to lift a fuel assembly or a drive shaft, respectively. This is to prevent a load from being accidentally dropped during the refueling process. The requirement also ensures that load limiting devices are available to prevent damage to a fuel assembly during fuel movement. These requirements have not been identified as a significant risk contributor (Refs. 2 & 3).

TR

TR 3.9.3 requires that the refueling machine and auxiliary hoist shall be used for the movement of fuel assemblies or drive shafts and that they shall be OPERABLE with certain requirements as discussed below. The refueling machine shall have a capacity of at least 3150 pounds, with two electrical overload cutoff limits of 2650 pounds, and 2800 pounds, respectively.

(continued)

BASES

TR (continued)

(Although the manufacturer's dynamic capacity rating of the refueling machine is 4000 pounds, only a capacity of 3150 pounds is required for movement of fuel assemblies or drive shafts). The auxiliary hoist shall have a capacity of at least 1200 pounds and a load indicator which shall be used to prevent the lifting of loads which are greater than 1190 pounds. These load requirements are specified in order to ensure that the equipment can handle the nominal weights of the components it must manipulate, while assuring that core components are not damaged from excessive lifting forces.

APPLICABILITY

TR 3.9.3 is applicable only during the movement of fuel assemblies or drive shafts within the reactor pressure vessel. The refueling machine's and auxiliary hoist's maximum loads and limitations are required when used for these purposes only, so the requirements are not applicable at any other times.

ACTIONS

A.1

If the refueling machine does not meet the requirements above, it is considered inoperable. Therefore, its use involving the movement of fuel assemblies within the reactor pressure vessel must be suspended immediately.

Suspension of the refueling operations shall not preclude completion of actions to establish a safe condition.

B.1

If the auxiliary hoist does not meet the requirements above, it is considered inoperable. Therefore, its use involving the movement of drive shafts within the reactor pressure vessel must be suspended immediately.

Suspension of the refueling operations shall not preclude completion of actions to establish a safe condition.

(continued)

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.9.3.1

TSR 3.9.3.1 requires the performance of three tests on the refueling machine. A load test of 3150 pounds must be performed on the refueling machine to verify its capacity. A test must be performed to demonstrate an automatic electrical load cutoff before the crane load is greater than 2650 pounds. A test must also be performed to demonstrate a second automatic electrical load cutoff before the crane load is greater than 2800 pounds. These tests verify that the capacity and the load limits are still within the Technical Requirements. The Surveillance Frequency of 18 months is based upon engineering judgment, manufacturer recommendation, the fact that the refueling machine is an infrequently used and highly reliable piece of equipment, and consistency with the typical industry refueling cycle.

TSR 3.9.3.2

TSR 3.9.3.2 requires a load test of at least 1200 pounds be performed on each required auxiliary hoist and its associated load indicator. This test verifies that the capacity is within the Technical Requirement and that the load indicator is functional. This surveillance is to be performed within 100 hours prior to starting the movement of the drive shafts within the reactor pressure vessel. The surveillance frequency is based on engineering judgment and the fact that the auxiliary hoist is an infrequently used and reliable piece of equipment.

REFERENCES

1. Watts Bar FSAR, Section 9.1.4, “Fuel Handling System.”

2. WCAP-11618, “MERITS Program-Phase II, Task 5, Criteria Application,” including Addendum 1 dated April, 1989, as clarified by Reference 3.

3. TVA letter, “Watts Bar Nuclear Plant (WBN) Unit 1 - Technical Requirements Manual (TRM),” dated August 27, 1992 (ADAMS Accession No. ML073230174) including Enclosure 1, “Watts Bar Technical Requirements Manual,” (ADAMS Accession No. ML073620391).

B 3.9 REFUELING OPERATIONS

B 3.9.4 Crane Travel - Spent Fuel Storage Pool Building

BASES

BACKGROUND

The spent fuel pool is a reinforced concrete structure with a stainless steel liner for leak tightness. The spent fuel storage racks consist of stainless steel structures with receptacles for nuclear fuel assemblies as they are used in a reactor, receptacles for neutron poison assemblies, and a supporting structure. Design of these storage racks is in accordance with Reference 1.

The racks can withstand the drop of a fuel assembly from its maximum supported height and the drop of tools used in the pool. Crane travel in the spent fuel storage pool building is limited through electrical and mechanical stops which prevent the movement of heavy objects, including shipping casks, over the spent fuel pool. The movement of casks is restricted to the cask loading area and areas away from the pool (Ref. 2).

APPLICABLE SAFETY ANALYSES

The release of radioactive material from fuel may occur during the refueling process, and at other times, as a result of fuel-cladding failures or mechanical damage caused by the dropping of fuel elements or the dropping of objects onto fuel elements (Ref. 1). The restriction on the movement of loads in excess of the nominal weight of a fuel and control rod assembly and the associated handling tool over other fuel assemblies in the storage pool areas ensures that, in the event this load is dropped, the activity release will be limited to that contained in a single fuel assembly, and that any possible distortion of fuel in the storage racks will not result in a critical array. These are design basis type accidents that have not been significant to risk when analyzed in environmental reports (Refs. 3 & 4).

TR

TR 3.9.4 requires that loads greater than 2059 pounds shall be prohibited from travel over fuel assemblies in the spent fuel pool. This ensures that objects traversing the pool are within the design basis and will not cause an unsafe condition if accidentally dropped.

(continued)

BASES (continued)

APPLICABILITY

TR 3.9.4 is applicable only when fuel assemblies are in the spent fuel pool. If there are no fuel assemblies in the pool, there is no danger of damaging a fuel assembly with a dropped load, therefore, the TR does not apply. The Applicability has been modified by a Note stating that the provisions of TR 3.0.3 do not apply.

ACTIONS

A.1

If a load in excess of 2059 pounds is allowed to traverse fuel assemblies in the spent fuel pool, the load must immediately be placed in a safe condition. This entails moving the load to a position which is not over the spent fuel pool.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.9.4.1

TSR 3.9.4.1 requires that the crane interlocks and physical stops, which prevent crane travel over fuel assemblies, are demonstrated to be OPERABLE. This surveillance must be performed within 7 days prior to using the crane and at least once per 7 days thereafter during crane operation. The 7 day Frequency corresponds to ANSI B30.2, "Frequent Inspection for Heavy to Severe Service."

REFERENCES

1. Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis."
2. Watts Bar FSAR, Section 9.1.2, "Spent Fuel Storage."
3. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989, as clarified by Reference 4.
4. TVA letter, "Watts Bar Nuclear Plant (WBN) Unit 1 - Technical Requirements Manual (TRM)," dated August 27, 1992 (ADAMS Accession No. ML073230174) including Enclosure 1, "Watts Bar Technical Requirements Manual," (ADAMS Accession No. ML073620391).