DRAFT

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment ~~(Ice Condenser)~~

BASES

BACKGROUND

The containment is a free standing steel pressure vessel ~~that is~~ surrounded by a reinforced concrete shield building. The containment vessel, including all its penetrations, is a low-leakage steel shell ~~that is~~ designed to contain the radioactive material that may be released from the reactor core following a Design Basis Accident (DBA)~~, such that offsite radiation exposures are maintained within limits~~. Additionally, the containment and shield building provide shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment vessel is a vertical cylindrical steel pressure vessel with hemispherical dome and a concrete base mat with steel membrane. It is completely enclosed by a reinforced concrete shield building. An annular space exits s between the walls and domes of the steel containment vessel and the concrete shield building to provide for the collection, mixing, holdup, and controlled release of containment out leakage. Ice Condenser containments utilize an outer concrete building for shielding and an inner steel containment for leak tightness.

Containment piping penetration assemblies provide for the passage of process, service, sampling, and instrumentation pipelines into the containment vessel while maintaining containment integrity. The shield building provides biological shielding and allows controlled filtered (1) release of the annulus atmosphere under accident conditions, ~~and~~ as well as environmental missile protection for the containment vessel and Nuclear Steam Supply System.

New paragraph

The inner steel containment and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. ~~Loss of containment OPERABILITY could cause site boundary doses, in the event of a DBA, to exceed values given in the licensing basis.~~ SR 3.6.1.1 leakage rate requirements ~~are in conformance~~ comply with 10 CFR 50, Appendix J (Ref. 1), as modified by approved exemptions.

Insert (A)

(continued)

Insert A

The isolation devices for the penetrations in the containment boundary are to maintain this leak tight barrier.

a. All penetrations required to be closed during accident conditions are either:

   1. capable of being closed by an OPERABLE automatic containment isolation system, OR

   2. closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in LCO 3.6.3.

b. Each air lock is OPERABLE except as provided in LCO 3.6.2.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rates.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures (SLB) are a loss of coolant accident (LOCA), a steam line break, and a rod ejection accident (REA) (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or a REA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of ~~[0.1%]~~ 0.25% of containment air weight per day (Ref. 3). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting DBA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a = ~~[0.1]%~~ 0.25% per day, and P_a = ~~[14.4]~~ 15.0 psig, which ~~results~~ bounds the calculated peak containment internal pressure resulting from the limiting design basis LOCA (Ref. 3). Satisfactory leakage rate test results ~~is~~ are a requirement for the establishment of containment OPERABILITY.

P

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The containment satisfies Criterion 3 of the NRC Policy Statement.

LCO

~~The requirements stated in this LCO define the performance of the containment fission product barrier. The containment design leakage rate (L_a) is an assumed initial condition.~~ By limiting leakage to within the acceptance criteria of 10 CFR 50, Appendix J (Ref. 1), containment OPERABILITY is maintained.

Compliance with ~~the~~ this LCO will ensure a containment configuration, including equipment hatches, that is structurally sound and will limit leakage to those leakage rates assumed in the safety analysis.

(continued)

BASES (continued)

Individual leakage rates specified for the containment air lock (LCO 3.6.2), and purge valves with resilient seals and shield building containment bypass leakage (LCO 3.6.3) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits only result in the containment being operable when the leakage results exceeding the acceptance criteria of Appendix J.

~~The containment LCO requires that containment OPERABILITY be maintained. Other LCOs support this LCO by ensuring:~~

~~a. All penetrations required to be closed during accidentconditions are either:~~

~~1. capable of being closed by an OPERABLE automatic containment isolation system, or~~

~~2. closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in LCO 3.6.3.~~

~~b. Each airlock is OPERABLE, except as provided in LCO 3.6.2.~~

~~c. The pressurized sealing mechanism associated with a penetration except as provided in LCO 3.6.[ ] is OPERABLE.~~

~~The Required Actions when other containment LCOs are not met have been specified in these LCOs and not in LCO 3.6.1.~~

APPLICABILITY

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~~ due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 ~~refueling operations~~ are addressed in LCO 3.9.4, "Containment Penetrations."

ACTIONS

Moved to Section 3.6.3

~~A.1~~

~~With the secondary containment bypass leakage rate not within limits, the assumptions of the radiological analysis (Ref.__) are not met. Therefore, the leakage must be restored to within limits within 4 hours. Restoration can be accomplished by isolating the penetration that caused the limit to be exceeded by use of one closed and deactivated~~

(continued)

BASES (continued)

~~automatic valve, closed manual valve, or blind flange. When a penetration is isolated, the leakage rate for the isolated penetration is assumed to be the actual pathway leakage through the isolation device. If two isolation devices are used to isolate the penetration, the leakage rate is assumed to be the lesser actual pathway leakage of the two devices. The 4 hour Completion Time is reasonable considering the time required to restore the leakage by isolating the penetration and the relative importance of secondary containment bypass leakage to the overall containment function.~~

A ~~B~~.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1-hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods where containment is inoperable is minimal.

B ~~C~~.1 and B ~~C~~.2

To satisfy this requirement, the plant must be braght to

The ~~unit~~ plant must be ~~placed in~~ brought to a MODE in which the LCO does not apply ~~if~~ containment cannot be restored to OPERABLE status ~~in the associated~~ within the required Completion Time. ~~This is done by placing the unit in~~ at least MODE 3 within 6 hours and ~~in~~ MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required ~~MODES~~ plant conditions from full power conditions in an orderly manner and without challenging ~~unit~~ plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of 10 CFR 50, Appendix J (Ref. 1), as modified by approved exemptions. ~~as contained in the containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to overall containment~~

W

(continued)

BASES (continued)

Failure to meet air lock leakage testing acceptance criteria (SR 3.6.2.1) or resilient seal containment purge valve leakage testing acceptance criteria does not necessarily result in a failure of this SR. The impact of the failure to meet ~~these two~~ this SR must be evaluated against the Type A, and Type B and C leakage tests of 10 CFR 50, Appendix J.

~~leakage (Type A leakage tests); leakage from equipment hatch, electrical penetrations with resilient seals, and other penetrations (Type B leakage tests) except air locks; and containment isolation valves (Type C leakage tests) except [42] inch purge valves. Leakage rate testing of the containment purge valves is addressed in LCO 3.6.3, "Containment Isolation Valves." Air lock leakage testing is addressed in LCO 3.6.2, "Containment Air Locks." Failure to meet air lock [secondary containment bypass leakage path] and purge valve, with resilient seal, specific leakage limits specified in LCO 3.6.2 and 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B and C leakage causes that to exceed limits.~~ SR Frequencies are as required by Appendix J as modified by approved exceptions. ~~or identified in the Containment Leakage Rate Testing Program.~~ Thus, SR 3.0.2 (which allows SR Frequency extensions) does not apply. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

Moved to Section 3.6.3

~~SR 3.6.1.2~~

~~This SR ensures that the leakage rate of secondary containment bypass leakage paths is less than the specified leakage rate. This provides assurance that the assumptions in the radiological evaluations of Reference 3 are met. The leakage rate of each bypass leakage path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves) unless the penetration is isolated by use of one closed and deactivated automatic valve, closed manual valve, or blind flange. In this case the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves. This method of quantifying maximum pathway leakage is only to be used for this SR (i.e., Appendix J maximum pathway leakage limits are to be quantified in accordance with Appendix 5.) The [18] month Frequency was developed considering it is prudent that this Surveillance be performed only during plant outage. Operating experience has shown that these components usually pass this SR when performed on the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability~~

(continued)

BASES (continued)

~~standpoint. A note has been added to this SR requiring the results to be evaluated against the acceptance criteria of SR 3.6.1.1.1. This ensures that shield building bypass leakage is properly accounted for in determining the overall primary containment leakage rate.~~

REFERENCES (R)

1. ~~10 CFR 50, Appendix J.~~ Title 10, Code of Federal Regulations, Part 50, Appendix J "Primary Reactor Containment Leakage testing for Water-Cooled Power Reactors." *
2. ~~FSAR, Section [15].~~ Watts Bar FSAR, Section 15, "Accident Analysis" *
3. ~~FSAR, Section [6.2].~~ Watts Bar FSAR, Section 6.2, "Containment Systems." *

(continued)

DRAFT

3.6 CONTAINMENT SYSTEMS

3.6.2 Containment Air Locks ~~(Atmospheric, Subatmospheric, Ice Condenser, and Dual)~~

LCO 3.6.2 [Two] containment air lock(s) shall be OPERABLE.

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

ACTIONS

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

1. Entry and exit is permissible to perform repairs of the affected air lock components.
2. Separate Condition entry is allowed for each air lock.
3. Enter applicable CONDITIONS and REQUIRED ACTIONS of LCO 3.6.1, "CONTAINMENT," when air lock leakage results in exceeding containment overall leakage rate.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment air locks with one containment air lock door inoperable.	------------NOTES------------ 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days to perform activities related to Technical Specification systems [if both air locks are inoperable]. 3. OPERABILITY may be verified by administrative means. -----------------------------	

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.1	Verify the OPERABLE door is closed in the affected air lock~~s~~.	1 hour
E	<u>AND</u>		
	A.2	Lock the OPERABLE door closed in the affected air lock~~s~~. W	24 hours
	<u>AND</u>		
	A.3	--------NOTE--------- Air lock doors in high radiation areas may be verified locked closed by administrative means. -------------------- Verify the OPERABLE door is locked closed in the affected air lock~~s~~. W	Once per 31 days
B. One or more containment air locks with containment air lock interlock mechanism inoperable.	------------NOTES------------ 1. Required Actions B.1, B.2, and B.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit of containment is permissible under the control of a dedicated individual. ~~3. OPERABILITY may be verified by administrative means.~~ ---------------------------- W		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.1	Verify an OPERABLE door is closed in the affected air lock. W	1 hour
	AND		
	B.2	Lock an OPERABLE door closed in the affected air lock. W	24 hours
	AND		
	B.3	--------NOTE--------- Air lock doors in high radiation areas may be verified locked closed by administrative means. ---------------------	
		Verify an OPERABLE door is locked closed in the affected air lock. W	Once per 31 days E
C. One or more containment air locks inoperable for reasons other than Condition A or B.	C.1	Initiate actions to evaluate containment overall leakage rate per LCO 3.6.1 "Containme ~~previous combined leakage rate for all penetrations and valves subject to Type B and C tests~~ using current air lock test results ~~per LCO 3.6.1.1~~.	Immediately
	AND		
	C.2	Verify a door is closed in the affected air lock. W	1 hour
	AND		

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.3 Restore air lock ~~to~~ W OPERABLE status.	24 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.1	-------------------NOTES------------------- 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria of SR 3.6.1.1 in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions as contained in the Containment Leakage Rate Test Program. -- Perform required air lock leakage rate testing in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions. [as contained in the Containment Leakage Rate Test Program]. W The acceptance criteria for air lock testing are: a. Overall air lock leakage rate is ≤ [0.05 L_a] when tested at ≥ ~~P~~ 15 psig. E b. For each door, leakage rate is ≤ [.01 L_a] when tested at ≥ [6 psig.] P	-----NOTE------ SR 3.0.2 is not applicable --------------- In accordance with 10 CFR 50, Appendix J, as modified by approved exemptions [as contained in the Containment Leakage Rate Test Program] W

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
[SR 3.6.2.2	------------------NOTE------------------ Only required to be performed ~~prior to~~ upon entry into containment. -- Verify only one door in the air lock can be opened at a time.	184 days]
~~[SR 3.6.2.3~~	~~Verify only one door in the air lock can be opened at a time.~~	~~[18 months (sub-atmospheric containments)]~~ N/A

DRAFT

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks ~~(Atmospheric, Subatmospheric, Dual, and Ice Condenser)~~

BASES

BACKGROUND

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

Each air lock is nominally a right circular cylinder, 10 feet in diameter, with a door~~s~~ at each end. ~~which are~~ The doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

Each personnel air lock is provided with limit switches on both doors that provide control room indication of door position. Additionally, control room indication is provided to alert the operator whenever an air lock door interlock mechanism is defeated.

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and ~~air~~ leak tightness is essential ~~to~~ for maintaining the containment leakage rates within limit ~~offsite doses from~~ in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in ~~offsite doses~~ a leakage rate in excess of ~~those described~~ that assumed in the ~~unit~~ safety analysis. SR 3.6.1.1 leakage rate requirements are in conformance with 10 CFR 50, Appendix J (Ref. 1), as modified by approved exemptions.

E

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(continued)

(2) A P_a value of 15.0 psig is utilized which bounds the calculated peak containment internal pressure following a DBA.

BASES (continued)

APPLICABLE SAFETY ANALYSES

~~The containment air lock LCO is derived from the requirements related to the control of offsite radiation does from major accidents by verifying that the actual containment leakage rate does not exceed the value assumed in the unit safety analysis.~~

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident ~~(LOCA)~~ and a rod ejection accident ~~(REA)~~ (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of (P) 0.25 ~~[0.1]~~% of containment air weight per day (Ref. 2). This leakage rate is defined in 10 CFR 50, Appendix J (Ref. 1), as L_a = ~~0.1%~~ per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) ~~= [47.4] psig~~ following a DBA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock(s).

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

LCO

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

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock ~~is necessary to~~

(continued)

BASES (continued)

is sufficient to provide a leak tight barrier
~~support Containment OPERABILITY~~ following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry and exit from containment.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 ~~refueling operations~~ are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

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

1

An additional Note has been added to provide clarification that for this LCO, separate condition entry is allowed for each air lock.

W

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

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

With one air lock door inoperable in one or more containment air locks, the OPERABLE door must be verified closed (A.1)
Required Action

(continued)

BASES (continued)

in each affected containment air lock. This ~~assures~~ ensures a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires containment be restored to OPERABLE status within 1 hour.

In addition, the affected air lock penetration must be isolated by locking closed the OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked closed OPERABLE air lock door. This ensures that an acceptable containment leakage boundary is maintained. The Completion Time of once per 31 days is based on engineering judgment and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. W

The Required Actions have been modified by ~~three~~ two notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C ~~is entered~~ are required if both doors in the air lock are inoperable. With both doors in the air lock inoperable, the Required Action to lock an OPERABLE door closed is not applicable. Required Actions C.1 and C.2 are the appropriate remedial actions. Note 2 allows use of the air lock for entry and exit for 7 days, to perform activities related to Technical Specification systems, if both air locks are inoperable (i.e., one door inoperable in both air locks). E ~~Primary~~ Containment entry ~~may be~~ is normally required on a periodic basis to perform Technical Specification Surveillances and Required Actions, as well as other activities on equipment inside ~~primary~~ containment, that is ~~either~~ required by Technical Specifications or that supports Technical Specification required equipment. This Note is not intended to preclude performing other activities (i.e., non-Technical Specification related activities) if the ~~primary~~ containment was entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the primary containment during the short time that the OPERABLE door is expected to be open. ~~Note 3 states that OPERABILITY may be verified by~~

W

(continued)

BASES (continued)

(W)

Note 3 applies to air lock doors located in high radiation areas or areas with limited access due to a subatmospheric environment and allows these doors to be verified locked closed by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

administrative means. The verification that a door is OPERABLE may be performed as an administrative check by examining logs or other information to determine if the door is OPERABLE. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of the door.

B.1, B.2 and B.3

(W)

With an air lock door interlock mechanism inoperable in one or more air locks, the Required Actions and associated Completion Times are consistent with those specified in Condition A ~~are applicable~~.

The Required Actions have been modified by ~~three~~ two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required ~~is entered~~ if both doors in the airlock are inoperable. With both doors in the airlock inoperable, ~~the Required Action to lock~~ an OPERABLE door closed is not available to be ~~is not applicable~~. Required Actions C.1 and C.2 are the appropriate remedial actions. Note 2 allows entry into and exit ~~into the primary~~ containment under the control of a dedicated individual stationed at the airlock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock). Note 3 states that OPERABILITY may be verified by administrative means. The verification that a door is OPERABLE may be performed as an administrative check by examining logs or other information to determine if the door is OPERABLE. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of the door.

(W)

C.1, ~~and~~ C.2, and C.3

With one or more air lock(s) inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be immediately initiated to evaluate previous combined leakage rates using current air lock test results ~~primary containment OPERABILITY~~. An evaluation is acceptable since it is overly conservative to immediately declare the primary containment inoperable if both doors in an air lock have failed a seal test or the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed) ~~primary~~ containment remains OPERABLE, yet only 1 hour (per LCO 3.6.1.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with

(continued)

BASES (continued)

New paragraph

both doors failing the seal test, the overall containment leakage rate can still be within limits. Required Action C.2 requires one door in the containment ~~A~~air ~~L~~lock must be verified to be closed within ~~a~~ 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, ~~"Containment,"~~ which requires that containment be restored to OPERABLE status within 1 hour.

Additionally, the affected air lock~~(s)~~ must be restored to OPERABLE status within the 24 hour Completion Time. The specified time period is considered reasonable for restoring an inoperable air lock~~s~~ to OPERABLE status assuming that at least one door is maintained closed in each affected air lock.

D.1 and D.2

the plant must be brought to MODE in which the LCO does not apply.

~~The unit must be placed in a MODE in which the LCO does not apply~~ ~~if~~ the inoperable containment ~~A~~air ~~L~~lock cannot be required restored to OPERABLE status within the ~~associated~~ Completion Time. ~~This is done by placing the unit in~~ at least MODE 3 within 6 hours and ~~in~~ MODE 5 within 36 hours. ~~The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems.~~

To achieve this status, the plant must be brought to

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of 10 CFR 50, Appendix J, as modified by approved exemptions. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The ~~Surveillance~~ Frequency is required by Appendix J (Ref. 1), as modified by approved exemptions. Thus, SR 3.0.2 (which allows ~~Surveillance~~ Frequency extensions) does not apply.

(continued)

BASES (continued)

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of an overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria of SR 3.6.1.1. This ensures that air lock leakage is properly accounted for in determining the overall containment leakage rate.

~~The SR has been modified by 2 Notes. Note 1 indicates an inoperable air lock door does not invalidate the previous successful performance of an overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission-product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria of SR 3.6.1.1.1. This ensures that air lock leakage is properly accounted for in determining the overall primary containment leakage rate.~~

SR 3.6.2.2

The air lock ~~door~~ interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed, and that simultaneous opening of the inner and outer doors ~~opening~~ will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is only challenged when containment is entered, this test is only required to be performed ~~prior to~~ upon entering containment, but is not required more frequently than every 184 days. The 184 day Frequency is based on engineering judgment and is considered adequate in view of other indications of door and interlock mechanism status available to operations personnel.

REFERENCES (R)

1. Title 10, Code of Federal Regulations, Part 50, ~~10 CFR 50~~, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."
2. Watts Bar FSAR, Section [6.2], "Containment Systems."

(continued)

3.6 CONTAINMENT SYSTEMS

CTS

3.6.3 Containment Isolation Valves ~~(Atmospheric, Subatmospheric, Ice Condenser, and Dual)~~ E

LCO 3.6.3 Each containment isolation valve shall be OPERABLE.

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

ACTIONS W

-----NOTES----- G

1. Penetration flow path(s) ~~Containment isolation valves [except for 42 inch purge valves]~~ may be opened intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for ~~supported~~ systems made inoperable by containment isolation valves.
4. Enter Applicable CONDITIONS And REQUIRED ACTIONS of LCO 3.6.1, "CONTAINMENT," when isolation valve leakage results in exceeding containment overall leakage rate. W

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Only applicable to penetration flow paths with two containment isolation valves. --------------------- One or more penetration flow paths with one containment isolation valve inoperable or shield building bypass ~~[except for purge valve leakage not within limit]~~.	A.1 Isolate the affected penetration flow path(s) by use of at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. W AND	4 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 --------NOTE--------- Valves and blind flanges in high radiation areas may be verified by use of administrative controls. --------------------- Verify the affected penetration flow path(s) is isolated. 6	Once per 31 days for isolation devices outside containment **AND** Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment
B. ---------NOTE--------- Only applicable to penetration flow paths with two containment isolation valves. ------------------------- One or more penetration flow paths OR shield building bypass with two containment isolation valves inoperable [except for purge valve leakage not within limits]. e	B.1 Isolate the affected penetration flow path(s) by use of at least one closed and deactivated automatic valve, closed manual valve, or blind flange. e W **AND** e E Δ	1 hour
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	--------NOTE--------- Valves and blind flanges in high radiation areas may be verified by use of administrative controls. --------------------- Verify the affected penetration is isolated.	~~Once per 31 days~~
C. ---------NOTE--------- Only applicable to penetration flow paths with only one containment isolation valve and a closed system. --------------------- One or more penetration flow paths with one containment isolation ~~valves~~ valve, inoperable.	C.1	Isolate the affected penetration flow path(s) by use of at least one closed and deactivated automatic valve, closed manual valve, or blind flange.	[4] hours
	AND		
	C.2	--------NOTE--------- Valves and blind flanges in high radiation areas may be verified by use of administrative controls. --------------------- Verify the affected penetration flow path(s) is isolated.	Once per 31 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
~~D~~ E. One or more penetration flow paths with one or more containment purge valves not within purge valve leakage limits. Insert from Page 3.6-4A W	~~D~~.1 E	Isolate ~~each~~ the affected penetration flow path(s) by use of at least one closed and deactivated automatic valve, closed manual valve, or blind flange.	24 hours
	AND		
	~~D.2~~ E 3	Perform SR 3.6.3.~~X~~6 for the resilient seal containment purge valve(s) closed to comply with Required Action ~~D~~ E.1.	Once per 92 days
~~E~~ F. Required Action and associated Completion Time not met.	~~E~~.1 F	Be in MODE 3.	6 hours
	AND		
	~~E~~.2 F	Be in MODE 5.	36 hours
Δ D. Shield Building bypass leakage not within limit.	D.1	Restore leakage within limit.	4 hours

INSERT

W

<u>AND</u> ~~D~~E.2	-------NOTE--------- Valves and blind flanges in high radiation areas may be verified by use of administrative controls. ---------------------	
	Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside containment <u>AND</u> Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
~~SR 3.6.3.1~~	~~Verify each [42] inch purge valve is sealed closed.~~	~~31 days~~
SR 3.6.3.2	W -------------------NOTE------------------- Not required to be met when the ~~[8] inch~~ containment purge valves are open for pressure control, ALARA, or air quality considerations for personnel entry, or for Surveillances ~~which~~ that require the valves to be open. ~~---~~ Verify each containment ~~[8] inch~~ purge valve is closed, except 4	31 days
SR 3.6.3.3	-------------------NOTE~~S~~------------------- ~~1.~~ Valves and blind flanges in high radiation areas may be verified by use of administrative controls. W ~~2. Not required to be met~~ for ~~on~~ containment isolation valves ~~which~~ that are open under administrative controls. --- Verify each containment isolation manual valve and blind flange which is located outside containment and required to be closed during accident conditions is closed, except	31 days
*SR 3.6.3.1	Verify the 24 inch containment lower compartment purge supply and/or exhaust isolation valves are physically restricted to ≤ 50° open.	~~31 days~~ 18 months

2

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.4	------------------NOTES------------------ 1. Valves and blind flanges in high rad areas may be verified by use of administrative controls. W ~~2. This SR is not required to be met~~ on containment isolation valves that are open under administrative controls. --- Verify each containment isolation manual valve and blind flange ~~which~~ that is located inside containment and required to be closed during accident conditions is closed, except	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.5	Verify the isolation time of each power operated and each automatic containment isolation valve is within limits.	[In accordance with the Inservice Testing Program, or 92 days]
~~[SR 3.6.3.6~~ 7	~~Cycle each weight- or spring-loaded check valve testable during plant operation through one complete cycle of full travel, and verify each check valve remains closed when the differential pressure in the direction of flow is ≤ [1.2] psid and opens when the differential pressure in the direction of flow is ≥ [1.2] psid and < [5.0] psid.~~	~~92 days]~~ ★

(continued)

E

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
[SR 3.6.3.~~7~~6	------------------NOTES------------------ Results shall be evaluated against acceptance criteria of SR 3.6.1.1 in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions. [as contained in the Containment Leakage Rate Test Program.] W --- Perform leakage rate testing for containment purge valves with resilient seals. [in accordance with the Containment Leakage Rate Test Program.]]	[184 days AND Within 92 days after opening the valve]
SR 3.6.3.~~8~~7	Verify each automatic containment isolation valve actuates to the isolation position on an actual or simulated actuation signal.	[18] months
~~[SR 3.6.3.9~~ 7	~~Cycle each weight- or spring-loaded check valve not testable during plant operation through one complete cycle of full travel, and verify each check valve remains closed when the differential pressure in the direction of flow is ≤ [1.2] psid and opens when the differential pressure in the direction of flow is ≥ [1.2] psid and < [5.0] psid.]~~	~~[18 months]~~ ★

SR 3.6.3.8 Insert M

Insert M

SR 3.6.3.8 ---------------------NOTE---------------------------------------
Results shall be evaluated against acceptance criteria of SR 3.6.1.1 in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, as contained in the Containment Leakage Rate Test Program.

W

Verify the combined leakage rate for all shield building bypass leakage rate for all shielc building bypass leakage paths is ≤ 0.25 La when pressurized to ≥ 15 psig.

Frequency - 18 months

DRAFT

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves ~~(Atmospheric, Subatmospheric, Ice Condenser, and Dual)~~ E

5

BASES

BACKGROUND

The containment isolation AS valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident-consequence-limiting systems to be provided with two isolation barriers that are closed on a containment isolation signal. These isolation devices are either passive or active (automatic). Manual valves, deactivated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. ~~Closed systems are those systems designed in accordance with GDC 57 (Ref. 1).~~ Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system ~~inside containment (in accordance with the requirements of 10 CFR 50, Appendix A, GDC 57)~~. These barriers (typically containment isolation valves) make up the containment isolation system.

Automatic isolation signals are produced during accident conditions. Containment Phase "A" isolation occurs upon receipt of a safety injection signal. The Phase "A" isolation signal isolates nonessential process lines in order to minimize leakage of fission product radioactivity. Containment Phase "B" isolation occurs upon receipt of a containment pressure High-High signal and isolates the remaining process lines, except systems required for accident mitigation. In addition to the isolation signals listed above, the purge and exhaust valves receive an isolation signal on a containment high radiation condition. As a result, the containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated from the environment in the event of a release of fission product radioactivity to the containment atmosphere as a result of a Design Basis Accident (DBA).

(continued)

BASES (continued)

The OPERABILITY requirements for containment isolation valves help ensure that containment is isolated within the time limits assumed in the safety analysis. Therefore, the OPERABILITY requirements provide assurance that containment leakage rates assumed in the safety analysis will not be exceeded.

WBN-specific write-up

1

See next page

~~Shutdown Purge System ([42]-inch purge valves)~~

~~The Shutdown Purge System operates to supply outside air into the containment for ventilation and cooling or heating, and may also be used to reduce the concentration of noble gases within containment prior to and during personnel access. The supply and exhaust lines each contain two isolation valves. Because of their large size, the 42-inch purge valves in some units are not qualified for automatic closure from their open position under DBA conditions. Therefore, the 42-inch purge valves are normally maintained closed in MODES 1 through 4 to ensure leak tightness.~~

~~Mini-Purge System ([8]-inch purge valves)~~

~~The Mini-Purge System operates to:~~

~~a. Reduce the concentration of noble gases within containment prior to and during personnel access, and~~

~~b. Equalize internal and external pressures.~~

~~Since the valves used in the Mini-Purge System are designed to meet the requirements for automatic containment isolation valves, these valves may be opened as needed in MODES 1, 2, 3 and 4.~~

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the requirements related to the control of leakage from the containment during ~~of offsite radiation doses resulting from~~ major accidents. This LCO is intended to ensure ~~that the offsite dose limits are not exceeded (i.e.,~~ that the actual containment leakage rate does not exceed the value assumed in the safety analysis~~)~~. As part of the containment boundary, containment isolation valve

(continued)

Reactor Building Purge Ventilation System

The Reactor Building Purge Ventilation System operates to supply outside air into the containment for ventilation and cooling or heating, to equalize internal and external pressures and to reduce the concentration of noble gases within containment prior to and during personnel access. The supply and exhaust lines each contain two isolation valves. Because of their large size and their exposure [illegible]gher containment pressure during accident conditions, the 24-inch containment lower compartment purge isolation valves are physically restricted to ≤ 50 degrees open.

Since the valves used in the Reactor Building Purge Ventilation System are designed to meet the requirements for automatic containment isolation valves, these valves may be opened as needed in MODES 1, 2, 3 and 4.

BASES (continued)

OPERABILITY supports containment OPERABILITY. Therefore, the safety analysis of any event requiring isolation of containment is applicable to this LCO.

Δ The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA) or a rod ejection accident (Ref. 1 ~~2~~). In the analysis for each of these accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential leakage paths to the environment through containment isolation valves (including containment purge valves) are minimized. ~~The offsite dose calculations~~ safety analyses ~~assumed~~ that the ~~[42]~~ 24 inch purge valves ~~were~~ are closed at event ~~initiation.~~ ~~Likewise, it is assumed that the containment is isolated such that release of fission products to the environment is controlled by the rate of containment leakage.~~

(P) (6)

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

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

(6) The purge valves may be unable to close in the environment following a LOCA. Therefore, each of the purge valves is required to remain sealed closed during MODES 1, 2, 3, and 4. In this case, the single failure criterion remains applicable to the containment purge valve due to failure in the control circuit associated with each valve. Again, the

(continued)

BASES (continued)

(6) purge system valve design precludes a single failure from compromising containment OPERABILITY as long as the system is operated in accordance with the subject LCO.

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

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valves' safety function is related to control containment leakage rates during ~~of offsite radiation exposures resulting from~~ a DBA. ~~This LCO addresses containment isolation valve OPERABILITY, stroke time, and containment purge valve leakage.~~ The other containment isolation valve leakage rates are addressed by LCO 3.6.1, "Containment," as ~~under~~ Type C testing. Δ

(2) MOVE

The automatic power operated containment lower compartment and shield building bypass isolation valves are considered OPERABLE when their isolation times are within limits and the valves actuate on an automatic isolation signal. The 24 ~~[42]~~ inch purge valves must be physically restricted (blocked) to ≤ 50 degrees open. ~~maintained sealed closed [or have blocks installed to prevent full opening]. and~~ Purge valves with resilient seals must meet additional leakage rate requirements ~~(SR 3.6.3.7)~~. ~~[also]~~ Blocked purge valves also actuate on an automatic signal. The valves covered by this LCO are listed along with their associated stroke times in the FSAR (Ref. ~~3~~ 2). Δ

MOVE

The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic valves are deactivated and secured in their closed position, and blind flanges and closed systems are in test ~~place~~. ~~Closed systems are those systems designed in accordance with GDC 57 (Ref. 1).~~ These passive isolation valves/devices are those listed in Reference ~~2~~ 1. Δ

This LCO provides assurance that the containment isolation valves ~~and purge valves~~ will perform their designed safety functions to control leakage from the containment during ~~mitigate the consequences of~~ accidents. MOVE Δ

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the

(continued)

In the event the air lock leakage results in exceeding the containment overall leakage rate, Note 4 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment".

W

probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE ~~and the containment purge valves are not required to be sealed closed~~ in MODE 5. The requirements for containment isolation valves ~~and containment purge valves~~ during MODE 6 ~~refueling operations~~ are addressed in LCO 3.9.4, "Containment Penetrations."

Δ

ACTIONS

W 6 Δ

The ACTIONS are modified by a Note allowing penetration flow paths ~~containment isolation valves, except the [42] inch purge valves,~~ to be opened intermittently under administrative control. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room ~~, at the controls of the valve~~. In this way, the penetration can be rapidly isolated ~~if a valid~~ when need for containment isolation ~~signal~~ is indicated. ~~Due to the size of the containment purge line penetration, and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, these valves may not be opened under administrative control.~~

6

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

The ACTIONS ~~Actions~~ are further modified by a third Note ~~which~~ that ~~requires the OPERABILITY of affected systems to be evaluated when a containment isolation valve is inoperable. This~~ ensures appropriate remedial actions are taken, if necessary, if the affected system(s) are rendered inoperable by ~~the~~ an inoperable ~~CIV.~~ containment isolation valve.

Δ

A.1 and A.2

W

In the event one containment isolation valve in one or more penetration flow paths is inoperable, except for purge valve or shield building bypass leakage not within limit, the affected penetration must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and deactivated

(continued)

BASES (continued)

automatic containment isolation valve, a closed manual valve, a blind flange, ~~or~~ and a check valve with flow through the valve secured. For A ~~For~~ penetrations isolated in accordance with Required Action A.2.2.1, the valve used to isolate the penetration should be the closest available one to containment. ~~One of these two~~ Required Action~~s~~ must be completed within 4-hours. The 4-hour Completion Time is reasonable considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.

For affected penetrations that cannot be restored to OPERABLE status within the 4-hour Completion Time and have been isolated in accordance with Required Action A.2.2.1, the affected penetrations must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations ~~that are~~ required to be isolated following an accident and ~~that~~ are no longer capable of being automatically isolated, will be in the isolation position should an event occur. The Completion Time ~~allowed for this verification is~~ of "once every 31 days" for isolation devices outside containment ~~and prior to entering MODE 4 from MODE 5 if not performed more often than once per 92 days for isolation devices inside containment. The Completion Time of once per 31 days was developed based upon Inservice Testing Program requirements to perform valve testing at least once per 92 days.~~ is appropriate considering the fact that the valves [illegible] operated under administrative control and the probability of their misalignment is low. This Required Action does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those isolation devices outside containment and capable of potentially being mispositioned are in the correct position.

MOVE

For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5, if not performed within the previous 92 days," is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device~~s~~ misalignment is an unlikely possibility.

Condition A has been modified by a Note indicating that this Condition is only applicable to those penetration flow paths with two containment isolation valves. ~~and a closed system inside containment (i.e., the containment penetration is isolated in accordance with 10 CFR 50, Appendix A, GDC 57,~~

(continued)

BASES (continued)

~~Reference 1)~~. For penetration flow paths with one containment isolation valve ~~CIV~~ and a closed system, Condition C provides the appropriate actions.

Required Action A.2 is modified by a Note that ~~which~~ applies to valves and blind flanges located in high radiation areas, and allows these valves to be verified closed by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

B.1

With two containment isolation valves in one or more flow paths inoperable, the affected penetration must be isolated within one hour. ~~The two inoperable valves must be in the same penetration flow path (but there can be more than one flow path meeting this Condition)~~. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and deactivated automatic valve, a closed manual valve, ~~or~~ and a blind flange. The ~~one~~ hour Completion Time is consistent with the ACTIONS of LCO 3.6.1 ~~importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.~~ In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to ~~ensure~~ provide assurance of leak tightness of containment ~~integrity is maintained~~ and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration is isolated is appropriate considering the fact that the valves are operated under administrative control and the probability of their misalignment is low.

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two isolation valves. Condition A of this LCO addresses the Condition of one isolation valve inoperable in this type of penetration flow path.

(continued)

BASES (continued)

C.1 and C.2

Δ With ~~When~~ one or more penetration flow paths with one containment isolation valve ~~are~~ inoperable, the inoperable valve ~~(s)~~ must be restored to OPERABLE status or the affected penetration must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and deactivated automatic valve, a closed manual valve, or a blind flange. A check valve may not be used to isolate the affected penetration, ~~since GDC 57 (Ref. 1) does not consider the check valve an acceptable automatic isolation valve. One of these~~ Required Action~~s~~ must be completed C.1 within the 4 hour Completion Time. The specified time period is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of maintaining containment integrity during MODES 1, 2, 3, and 4. In the event ~~that~~ the affected penetration is Δ isolated in accordance with Required Action C.1, the affected penetration must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure ~~ensure that~~ containment integrity is maintained and provide assurance of leak tightness and that containment penetrations requiring isolation ~~required to be isolated~~ following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration is isolated is appropriate because the valves are operated under administrative control and the probability of their misalignment is low.

Condition C is modified by a Note indicating that this Condition is only applicable to penetrations with only one containment isolation valve and a closed system. This Note is necessary since this Condition is written to specifically address those penetration flow paths in a closed system. ~~isolated consistent with 10 CFR 50, Appendix A, GDC 57~~ Δ ~~(Ref. 1). GDC 57 allows those lines that enter containment but are neither part of the reactor coolant pressure boundary nor connected directly to containment atmosphere to be isolated by means of one containment isolation valve.~~

Required Action C.2 is modified by a Note that ~~which~~ applies to valves and blind flanges located in high radiation areas, and allows these valves to be verified closed by use of administrative controls. Allowing verification by

(continued)

BASES (continued)

administrative controls is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

Insert (0) → D.1

~~E D.1 and E D.2~~

~~In the event that one or more penetration flow paths with one or more containment purge valves are not within the purge valve leakage limits, purge valve leakage must be restored to within limits or the affected penetration must be isolated. The method of isolation must use at least one of isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and deactivated automatic valve, a closed manual valve, or blind flange. One of these Required Actions must be completed within the 24 hour Completion Time. The specified time period is reasonable considering that the containment purge valves remain closed such so that a gross breach of containment does not exist. For containment purge valves that are isolated in accordance with Required Action D.1, SR 3.6.3.7 must be performed at least once per every 92 days. This ensures that degradation of the resilient seals is detected and confirms that the leakage rate of the containment purge valves does not increase during the time the penetration is isolated. The normal Frequency of SR 3.6.3.7 is 184 days and is based on an NRC initiative, Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 4). Since more reliance is being placed on a single valve during this condition, it is prudent to perform the SR more often. Therefore, a periodic interval Frequency of once per 92 days is appropriate.~~

be by the

without resilient seals

and a

(W)

INSERT FROM Page B 3.6-9A

F ~~E~~.1 and F ~~E~~.2

If the Required Actions ~~and associated~~ cannot be met within the required Completion Time~~s~~, ~~are not met, the plant~~ must be ~~placed in~~ brought to a MODE in which the LCO does not apply. ~~This is done by placing~~ To achieve this status, the plant ~~in~~ must be brought to at least MODE 3 within 6 hours and to at least MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required ~~MODES~~ plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

Insert O

D.1 With the Shield Building bypass leakage rate not within limits, the assumptions of the radiological analysis is not met. Therefore, the leakage must be restored to within limits within 4 hours. Restoration can be accomplished by isolating the penetrations that cause the limit to be exceeded, by use of closed and deactivated automatic valve, closed manual valve or blind flange. When a penetration is isolated, the leakage rate for the isolated penetration is assumed to be the actual pathway leakage through the isolation device. If two isolation devices are used to isolate the penetration, the leakage rate is assumed to be the lesser actual pathway leakage of the two devices. The 4 hour Completion Time is reasonable considering the time required to restore the leakage by isolating the penetration and relative importance of Shield Building bypass leakage to the overall containment functions.

INSERT

(W)

E.1, E.2, and E.3

In the event one or more containment purge valves are not within the purge valve leakage limits, purge valve leakage must be restored to within limits or the affected penetration must be isolated. The method of isolation must be by the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a [closed and deactivated automatic valve, a closed manual valve, or blind flange]. A purge valve with resilient seals utilzed to satisfy Required Action ~~D~~E.1 must have been demonstrated to meet the leakage requirements of SR 3.6.3.~~7~~6. The specified Completion Time is reasonable, considering that one containment purge valve remains closed ~~(refer to the Note to SR 3.6.3.1)~~ so that a gross breach of containment does not exist.

In accordance with Required Action ~~D~~E.2 this penetration flow path must be verified to be isolated on a periodic basis. The periodic verification is necessary to ensure that containment penetrations required to be isolated following an accident, which are no longer capable of being automatically isolated, will be in the isolation position should an event occur. This Required Action does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those isolation devices outside containment potentially capable of being mispositioned are in the correct position. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

For the containment purge valve with resilient seal that is isolated in accordance with Required Action ~~D~~E.1, SR 3.6.3.~~7~~6 must be performed at least once every ~~[72]~~ 92 days. This assures that degradation of the resilient seal is detected and confirms that the leakage rate of the containment purge valve does not increase during the time the penetration is isolated. The normal Frequency for SR 3.6.3.~~7~~6, 184 days, is based on an NRC initiative, Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. ~~8~~3). Since more reliance is placed on a single valve while in this Condition, it is prudent to perform the SR more often. Therefore, a Frequency of once per ~~[72]~~ 92 days was chosen and has been shown acceptable based on operating experience.

BASES (continued)

SURVEILLANCE REQUIREMENTS

2

WBN-specific write-up

See next page

SR 3.6.3.1

Each [42]-inch containment purge valve is required to be verified sealed closed at 31-day intervals. This SR is designed to ensure that a gross breach of containment is not caused by an inadvertent or spurious opening of a containment purge valve. Detailed analysis of the purge valves failed to conclusively demonstrate their ability to close during a LOCA in time to limit offsite doses. Therefore, these valves are required to be in sealed closed position during MODES 1, 2, 3, and 4. Containment purge valves that are sealed closed must have motive power to the valve operator removed. This can be accomplished by de-energizing the source of electric power or removing the air supply to the valve operator. In this application, the term "sealed" has no connotation of leak tightness. The Surveillance Frequency is a result of an NRC initiative, Generic Issue B-24, related to containment purge valve use during plant operations (Ref. 5).

SR 3.6.3.2

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This SR ensures that the ~~[8]-inch~~ purge valves are closed as required or, if open, open for an allowable reason. The SR is modified by a Note ~~which~~ that states that the SR is not required to be met when the purge valves are open for the reasons stated. The Note states that these valves may be opened for pressure control, ALARA or air quality considerations for personnel entry, ~~and~~ Surveillance tests that require the valve to be open. ~~The [8]-inch~~ All purge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31-day ~~Surveillance~~ Frequency is consistent with other containment isolation valve requirements discussed ~~under~~ in SR 3.6.3.3.

4

6

SR 3.6.3.3

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and required to be closed during accident

(continued)

SR 3.6.3.1

capable of closing during a DBA. Therefore, the site boundry dose guideline values of 10 CFR 100 would not be exceeded in the event of an accident

The 24-inch containment lower compartment purge supply and/or exaust isolation valves are required to be verified to be physically restricted to ≤ 50 degrees open every 31 days. This is to ensure that the proper closure times can be met upon receipt of an isolation signal. The 31 day frequency is based on ~~engineering judgement.~~

This SR ensures that the containment lower compartment purge valves are ~~closed as required or, if open, open for an allowable reason. This SR has been modified by a Note indicating that these valves may be opened for pressure control, as low as reasonably achievable (ALARA) air quality considerations for personnel entry, and Surveillance tests that require the valve to be open.~~ during containment purge operation.

ALARA radiation exposure considerations for personnel inspecting these remotely located valves, and on the very improbable chance that the mechanical device which physically restricts each valve will change position.

BASES (continued)

conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. ~~The Inservice Testing Program requires valve testing on a 92-day Frequency.~~ This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment, ~~and~~ capable of ~~potentially~~ being mispositioned, are in the correct position. Since verification of valve position for valves outside containment is relatively easy, the 31 day Frequency is based on engineering judgement and was chosen to provide added assurance of the correct positions.

W

Two Notes apply to this SR. A ~~The~~ first Note applies to valves and blind flanges located in high radiation areas, and allows these devices ~~valves~~ to be verified ~~as~~ closed by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small. A second Note has been included to clarify that The SR excludes valves that are open under administrative controls are not required to meet the SR during the time the valves are open.

W W

SR 3.6.3.4

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

W W

Two Notes have been added to this SR. A ~~This~~ first Note allows valves and blind flanges located in high radiation areas to be verified ~~as~~ closed by use of administrati~~on~~ve controls. Allowing verification by administrative controls

(continued)

BASES (continued)

(7)

is considered acceptable, since ~~the primary containment is inerted and~~ access to these areas is typically restricted during MODES 1, 2, ~~and~~ 3, for ALARA reasons. Therefore, the probability of misalignment of these valves, once they have been verified to be in their proper position, is small. A second Note has been included to clarify that valves that are open under administrative controls are not required to meet the SR during the time ~~that the valves~~ are open. Δ

and 4

(W) (W) Ae

they

This SR excludes

SR 3.6.3.5

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

N/A

(7)

~~SR 3.6.3.6~~

~~In subatmospheric containments, the check valves that serve a containment isolation function are weight- or spring-loaded to provide positive closure in the direction of flow. This ensures that these check valves will remain closed when the inside containment atmosphere returns to subatmospheric conditions following a DBA. SR 3.6.3.6 verifies the operation of the check valves that are testable during plant operation. The Frequency of 92 days is consistent with the Inservice Inspection and Testing Program requirement for valve testing on a 92 day Frequency.~~

SR 3.6.3.~~7~~6

Δ

For containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J ~~(Ref. 6)~~, is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other ~~types of seals.~~ seal types. Based on this observation, and the importance of maintaining this penetration leak tight (due to the direct path between

(continued)

BASES (continued)

containment and the environment), a Frequency of 184 days was established as part of the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. ~~4~~ 3).

Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that occurring to a valve that has not been opened). Thus, decreasing the interval (from 184 days) is a prudent measure after a valve has been opened.

Δ

A Note to this SR requires the results to be evaluated against the acceptance criteria of SR 3.6.1.1. This ensures that excessive containment purge valve leakage is properly accounted for in determining the overall containment leakage rate ~~to verify containment OPERABILITY~~.

Δ

SR 3.6.3.~~8~~ 7

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. The 18 month Frequency was developed considering it is prudent that this SR be performed only during a plant outage since isolation of penetrations would eliminate cooling water flow and disrupt the normal operation of many critical components. Operating experience has shown that these components usually pass this ~~SR~~ Surveillance when performed on the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

N/A

7

~~SR 3.6.3.9~~

~~In subatmospheric containments, the check valves that serve a containment isolation function are weight- or spring-loaded to provide positive closure in the direction of flow. This ensures that these check valves will remain~~

(continued)

BASES (continued)

(7)

closed when the inside containment atmosphere returns to subatmospheric conditions following a DBA. SR 3.6.3.9 verifies the operation of the check valves that are not testable during plant operation. The Frequency of 18 months is based on such factors as the inaccessibility of theses valves, the fact that the plant must be shut down to perform the tests, and the successful results of the tests on an 18-month basis during past plant operation.

Included in SR 3.6.3.1 for WBN

SR 3.6.3.10 (3)

[Reviewer Note: This SR is only required for those plants with resilient seal purge valves allowed to be open during [MODES 1, 2, 3, or 4] and having blocking devices on the valves which are not permanently installed.]

Verifying that each [ ]-inch primary containment purge valve is blocked to restrict opening to ≤ [50%] is required to ensure that the valves can close under DBA conditions within the times assumed in the analysis of references 3 and 6. If a LOCA occurs, the purge valves must close to maintain containment leakage within the values assumed in the accident analysis. At other times when purge valves are required to be capable of closing (e.g., during movement of irradiated fuel assemblies), pressurization concerns are not present, thus the purge valves can be fully open. The 18 month frequency was developed considering it is prudent that many Surveillances be performed only during a plant outage. Operating experience has shown that these components usually pass the SR when performed on the 18 month frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Δ Insert Q SR 3.6.3.8

REFERENCES (R) Δ

~~1.~~ ~~Title 10, Code of Federal Regulations, Part 50,~~ ~~10 CFR 50, Appendix A, GDC 50, GDC 52, GDC 53, GDC 54, GDC 56, GDC 57.~~

~~2.~~ 1. Watts Bar FSAR, Section 15, "Accident Analysis." ~~FSAR, Section [15].~~

~~3~~ 2. Watts Bar FSAR, Section 6.2.4.2, "Containment Isolation System Design", and Table 6.2.4-1, "Containment Penetrations and Barriers" ~~FSAR, Section [6.2].~~

~~4.~~ 3. Generic Issue B-20 "Containment Leakage Due to Seal Deterioration."

(continued)

Insert Q

SR 3.6.3.8

This SR ensures that the leakage rate of shield building bypass leakage paths is less than the specified leakage rate. This provides assurance that the assumptions in the radiological evaluations of Reference ~~3~~ 2 are met. The leakage rate of each bypass leakage path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves) unless the penetration is isolated by use of one closed and deactivated automatic valve, closed manual valve, or blind flange. In this case, the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves. This method of quantifying maximum pathway leakage is only to be used for this SR (i.e., Appendix J maximum pathway leakage limits are to be quantified in accordance with Appendix 5). The 18 month Frequency was developed considering it is prudent that this Surveillance be performed only during a plant outage. Operating experience has shown that these components usually pass this SR when performed on the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. A Note has been added to this SR requiring the results to be evaluated against the acceptance criteria of SR 3.6.1.1. This ensures that shield building bypass leakage is properly accounted for in determining the overall primary containment leakage rate.

BASES (continued)

△ ~~5. Generic Issue B-24 "Containment Purge Valve Reliability."~~

△ ~~6. 10 CFR 50, Appendix J.~~ a

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.4~~A~~9 Containment Pressure ~~(Atmospheric, Dual, and Ice Condenser)~~

DRAFT

LCO 3.6.4~~A~~9 Containment pressure shall be ≥ ~~[-0.3]~~ -0.1 ~~psig~~ and ≤ ~~[+1.5]~~ +0.3 psi~~g~~d ~~.~~ relative to the annulus.

P

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.4~~A~~.1 Verify containment pressure is within limits.	12 hours

DRAFT

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 ~~A~~ Containment Pressure ~~(Atmospheric, Dual, and Ice Condenser)~~ E

P

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential (-2.0 psig) with respect to the ~~outside~~ atmosphere in the event of inadvertent actuation of ~~the Containment Spray System~~ Shield Building Annulus. P

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a DBA (Design Basis Accident), post accident containment pressure could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered relative to containment pressure, are the LOCA and SLB, which are analyzed using computer pressure transients. The worst case LOCA generates larger mass and energy release than the worst case SLB. Thus, the LOCA event bounds the SLB event from the containment peak pressure standpoint (Ref. 1).

The initial pressure condition used in the containment analysis was ~~[17.7]~~ 15.0 psia (~~[3.0]~~ 0.3 psig). This resulted in a maximum peak pressure from a LOCA of ~~[53.9]~~ 11.21 psig. The containment analysis (Ref. 2) shows that the maximum peak ~~calculated~~ containment pressure, P_a, bounds the calculated results from the limiting LOCA.

~~The initial pressure condition used in the containment analysis was [17.7] 15.0 psia ([3.0] 0.3 psig)~~ (15 psig). The maximum containment pressure resulting from the worst case LOCA,

(continued)

BASES (continued)

11.21 ~~[44.1] psig~~, does not exceed the containment design pressure, 13.5 ~~[55]~~ psig.

(P) The containment was also designed for an external pressure load equivalent to -2.0 psig ~~[2.5] psig~~. The inadvertent actuation of the Containment Spray System was analyzed to determine the resulting reduction in containment pressure. The initial pressure condition used in this analysis was -0.1 ~~[0.3]~~ psig. This resulted in a minimum pressure inside containment of 1.4 psig, ~~[2.0] psig~~, which is less than the design load.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. Therefore, for the reflood phase, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. 2).

Containment pressure satisfies Criterion 2 of the NRC Policy Statement.

LCO

Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following the inadvertent actuation of the Containment Spray System.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analysis are maintained, the LCO is applicable in MODES 1, 2, 3 and 4.

(continued)

BASES (continued)

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODES 5 ~~or~~ and 6.

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, it must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1-hour.

B.1 and B.2

If containment pressure cannot be restored to OPERABLE status within ~~limits in~~ the ~~associated~~ required Completion Time, the ~~unit~~ plant must be ~~placed in~~ brought to a MODE in which the LCO does not apply. ~~This is done by placing~~ To achieve this status the ~~unit in~~ plant must be brought to at least MODE 3 within 6 hours and ~~in~~ to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required ~~MODES~~ plant conditions from full power conditions in an orderly manner and without challenging ~~unit~~ plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.4~~A~~9.1

Verifying that containment pressure is within limits ensures that ~~unit~~ operation remains within the limits assumed in the containment analysis. The 12 hour Frequency of this SR was developed based on operating experience related to trending of ~~both~~ containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.

(continued)

BASES (continued)

REFERENCES (R)

1. Watts Bar FSAR, Section [6.2], "Containment Functional Design."
2. Title 10, Code of Federal Regulations, Part 50, [redacted] Appendix K, "ECCS Evaluation Models."

(continued)

DRAFT

3.6 CONTAINMENT SYSTEMS

3.6.5~~B~~9 Containment Air Temperature ~~(Ice Condenser)~~ E

LCO 3.6.5B Containment average air temperature shall be:

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

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

---------------------------NOTE---------------------------
The minimum ~~C~~containment average air temperature in MODES 2, 3, and 4 may be reduced to [60]°F.

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5~~B~~9.1	Verify containment upper compartment average air temperature is within limits.	24 hours
SR 3.6.5~~B~~9.2	Verify containment lower compartment average air temperature is within limits.	24 hours

DRAFT

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5~~B~~ Containment Air Temperature ~~(Ice Condenser)~~ (E)

BASES

BACKGROUND

The containment structure serves to contain radioactive material which may be released from the reactor core following a Design Basis Accident (DBA). The containment average air temperature is limited, during normal operation, to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB).

The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during unit operations. The total amount of energy to be removed from containment by the containment spray and cooling systems during post accident conditions, is dependent upon the ~~quantity of~~ energy released to the containment due to the event, as well as the initial containment temperature and pressure. The higher the initial temperature, the more energy that must be removed, resulting in a higher peak containment ~~pressure and~~ temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

APPLICABLE SAFETY ANALYSES

New Paragraph

Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analyses for containment (Ref. 1).

The limiting DBAs considered relative to containment OPERABILITY are the LOCA and SLB. The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously, or consecutively. The postulated DBAs are analyzed ~~in~~ with regard

(continued)

BASES (continued)

to Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train each of Containment Spray System, Residual Heat Removal System, and Air Return System being rendered inoperable.

The limiting DBA for the maximum peak containment air temperature is an SLB. For the upper compartment, the initial containment average air temperature assumed in the design basis analyses (Ref. 1) is [110]°F. For the lower compartment, the initial average containment air temperature assumed in the design basis analyses is [120]°F. This resulted in a maximum containment air temperature of [326]°F. The design temperature is [250]°F.

The temperature upper limits are used to establish the environmental qualification operating envelope for both containment compartments. The maximum peak containment air temperature for both containment compartments was calculated to exceed the containment design temperature for only a few seconds during the transient. The basis of the containment design temperature, however, is to ensure the performance of safety related equipment inside containment (Ref. 2). Thermal analyses showed that the time interval during which the containment air temperature exceeded the containment design temperature was short enough that the equipment surface temperatures remained below the design temperature. Therefore, it is concluded that the calculated transient containment air temperatures are acceptable for the DBA SLB.

The temperature upper limits are also used in the depressurization analyses to ensure that the minimum pressure limit is maintained following an inadvertent actuation of the Containment Spray System for both containment compartments.

The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure is ~~the~~ a LOCA. The temperature lower limits, [85]°F for the upper compartment and [100]°F for the lower compartment, are used in ~~this~~ analyses to ensure that, in the event of an accident, this the maximum containment internal pressure will not be exceeded in either containment compartment.

(continued)

BASES (continued)

Containment average air temperature satisfies Criterion 2 of the NRC Policy Statement.

LCO

During a DBA, with an initial containment average temperature within the LCO temperature limits, the resultant peak accident temperature ~~is calculated to remain within acceptable limits.~~ maintained below the containment design temperature. As a result, the ability of containment to perform its design function is ensured. In MODES 3 and 4, containment air temperature may be as low as 60°F, because the resultant calculated peak containment accident pressure would not exceed the design pressure due to a lesser amount of energy released from the pipe break in these MODES.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment average air temperature within the limit is not required in MODE 5 or 6.

ACTIONS

A.1

When containment average air temperature in the upper or lower compartment is not within the limit of the LCO, the average air temperature in the affected compartment must be restored within 8 hours. This Required Action is necessary to return operation to within the bounds of the containment analysis. The 8-hour Completion Time is acceptable considering the sensitivity of the analysis to variations in this parameter, and provides sufficient time to correct minor problems. ~~or to prepare the unit for an orderly shutdown.~~

B.1 and B.2

If the containment average air OPERABLE status temperature cannot be restored to ~~within its limits~~ in the ~~associated~~ required Completion Time, the unit must be ~~placed in~~ brought to a MODE in which the LCO

(continued)

BASES (continued)

does not apply. To achieve this status, plant must be brought ~~This is done by placing~~ the ~~unit in~~ to at least MODE 3 within 6 hours and to ~~in~~ MODE 5 within 36 hours. The allowable Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging ~~unit~~ plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.5~~B~~.1 and SR 3.6.5~~B~~.2

Verifying that ~~the~~ containment average air temperature is within the LCO limits ensures that containment operation remains within the limits assumed for the containment analyses. In order to determine the containment average air temperature, a weighted average is calculated using measurements taken at locations within the containment selected to provide a representative sample of the overall containment atmosphere. The 24 hour Frequency of these SRs is considered acceptable based on observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containments).

Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment temperature condition.

REFERENCES (R)

1. Watts Bar FSAR, Section [6.2], "Containment Systems."
2. ~~10 CFR 50.49.~~ Title 10, Code of Federal Regulations, Part 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.6~~C~~ 9 Containment Spray System (E) ~~(Ice Condenser)~~

DRAFT

LCO 3.6.6~~C~~ 9 Two containment spray trains and two RHR spray trains shall be OPERABLE.

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

4

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6~~C~~ 9.1	Verify each containment spray manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.6.2	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.6.6.3	Verify each containment spray automatic valve in the flow path actuates to the correct position on an actual or simulated actuation signal.	[18] months
SR 3.6.6.4	Verify each containment spray pump starts automatically on an actual or simulated actuation signal.	[18] months
SR 3.6.6.5	Verify each spray nozzle is unobstructed.	First refueling outage AND 10 years
* SR 3.6.6.6	Perform SR 3.5.2.2 and SR 3.5.2.4 for the RHR spray system.	In accordance with applicable SRs.

4

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6~~C~~ Containment Spray System ~~(Ice Condenser)~~ E

BASES

BACKGROUND

W

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

1

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

6

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

The Containment Spray System and RHR System provide a spray of cold or subcooled borated water ~~into~~ onto the upper and lower regions of ~~C~~containment and in dead-ended volumes to limit the containment pressure and temperature during a DBA. The RWST solution temperature is an important factor in

(continued)

BASES (continued)

determining the heat removal capability of the Containment Spray System during the injection phase. In the recirculation mode of operation, heat is removed from the containment sump water by the Containment Spray System and RHR heat exchangers. Each train of the Containment Spray System, supplemented by a train of RHR spray, provides adequate spray coverage to meet the system design requirements for containment heat removal.

2

The Spray Additive System injects a sodium hydroxide (NaOH) solution into the spray. The resulting alkaline pH of the spray enhances the ability of the spray to scavenge iodine fission products from the containment atmosphere. The NaOH added in the spray also ensures an alkaline pH for the solution recirculated in the Containment sump. The alkaline pH of the containment sump water minimizes the evolution of iodine and the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid.

6

The Containment Spray System is actuated ~~either~~ automatically, by a containment ~~High-3~~ High-High pressure signal, or manually. An automatic actuation opens the containment spray pump discharge valves, starts the two containment spray pumps, and begins the injection phase. A manual actuation of the Containment Spray System requires the operator to actuate two separate switches on the main control board to begin the same sequence. The injection phase continues until an RWST level Low-Low alarm is received. The Low-Low alarm for the RWST ~~actuates valves to align the containment spray pump suction to the containment sump and/or~~ signals the operator to manually align the system to recirculation mode. The Containment Spray System in recirculation mode maintains an equilibrium temperature between the containment atmosphere and the recirculated sump water. Operation of the Containment Spray System in the recirculation mode is controlled by the operator in accordance with the emergency operation procedures. T 3

6
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The RHR spray operation is initiated manually, when required by the emergency operating procedures, after the Emergency Core Cooling System (ECCS) is operating in the recirculation mode. This additional spray capacity would typically be used after the ice bed has been depleted and in the event that containment pressure rises above a predetermined limit.

(continued)

BASES (continued)

The Containment Spray System is an ESF system. It is designed to ensure that the heat removal capability required during the post accident period can be attained.

The operation of the Containment Spray System, together with the ice condenser, is adequate to assure pressure suppression during the initial blowdown of steam and water from a DBA. During the post blowdown period, the Air Return System (ARS) is automatically started. The ARS returns upper compartment air through the divider barrier to the lower compartment. This serves to equalize pressures in containment and to continue circulating heated air and steam through the ice condenser, where heat is removed by the remaining ice.

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~~After the ECCS is aligned to the recirculation mode, the RHR spray is aligned to the recirculation mode.~~ The RHR sprays are available to supplement the Containment Spray System, if required, in limiting containment pressure.

W

The Containment Spray System ~~supports containment OPERABILITY by~~ limits ~~limiting~~ the temperature and pressure that could be expected following a DBA. Protection of containment integrity limits leakage of fission product radioactivity from containment to the environment. ~~Loss of containment integrity could cause site boundary doses, in the event of a DBA, to exceed values given in the licensing basis.~~ Δ

APPLICABLE SAFETY ANALYSES

The limiting DBAs considered relative to containment operability are the loss of coolant accident (LOCA) and the steam line break (SLB). The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant Containment pressure and temperature transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed, in regard to containment ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train of the Containment Spray System, the RHR System, and the ARS being rendered inoperable (Ref. 2). W

P 11.21

The DBA analyses show that the maximum peak containment pressure of 11.21 ~~[44.1]~~ psig results from the LOCA analysis, and is calculated to be less than the containment design pressure. The maximum peak containment atmosphere

(continued)

BASES (continued)

temperature ~~of [306]°F~~ results from the SLB analysis and was calculated to exceed the containment design temperature for a few seconds during the DBA SLB. The basis of the containment design temperature, however, is to ensure the OPERABILITY of safety related equipment inside Containment (Ref. 3). Thermal analyses showed that the time interval during which the containment atmosphere temperature exceed the containment design temperature was short enough that the equipment surface temperatures remained below the design temperature. Therefore, it is concluded that the calculated transient containment atmosphere temperatures are acceptable for the DBA SLB.

The modeled Containment Spray System actuation from the Containment analysis is based on a response time associated with exceeding the containment ~~High-3~~ High-High pressure signal setpoint to achieving full flow through the containment spray nozzles. A delayed response time initiation provides conservative analyses of peak calculated Containment temperature and pressure responses. The Containment Spray System total response time of ~~[45]~~ 221 seconds is composed of signal delay, diesel generator startup, and system startup time.

T

P

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the ECCS cooling effectiveness during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 4).

Inadvertent actuation of the Containment Spray System is evaluated in the analysis, and the resultant reduction in containment pressure is calculated. The maximum calculated ~~reduction in containment pressure resulted in a containment external pressure load of [1.2] psid~~ steady state pressure differential relative to the Shield Building annulus is 1.4 psid, which is below the containment design external ~~pressure load~~ of 2.0 psid.

P

The Containment Spray System satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

T

LCO 4

During a DBA, one train of Containment Spray System and RHR Spray System is required to provide the heat removal capability assumed in the safety analyses. ~~Additionally, a minimum of one train of the Containment Spray System, with spray pH adjusted by the Spray Additive System, is required to scavenge iodine fission products from the containment atmosphere and ensure their retention in the containment sump water.~~ To ensure that these requirements are met, two containment spray trains and two RHR spray trains must be OPERABLE with power from two safety related independent power supplies. Therefore, in the event of an accident, at least one train in each system ~~occurs~~ operates.

N/A

6

Each Containment Spray ~~System~~ train typically includes a spray pump, headers, valves, heat exchangers, nozzles, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an ESF actuation signal and automatically transferring suction to the containment sump.

Each RHR Spray train includes a pump, headers, valves, heat exchangers, nozzles, piping, instruments and controls to ensure an OPERABLE flow path capable of taking suction from the containment sump and supplying flow to the spray header

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the Containment Spray System.

In MODES 5 and 6, the probability and consequences of these events are reduced because of the pressure and temperature limitations of these MODES. Thus, the Containment Spray System is not required to be OPERABLE in MODE 5 or 6.

T

ACTIONS 4

A Note has been added which states the RHR spray trains are not required in MODE 4. The containment spray system does not require supplemental cooling from RHR spray in MODE 4.

A.1

T

With one Containment Spray System train or one RHR Spray System train inoperable, the affected train must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the heat removal and iodine removal needs after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal and iodine removal capabilities afforded by the OPERABLE train and the low probability of a DBA occurring during this period.

(continued)

BASES (continued)

B.1 and B.2

~~In the event the affected~~ If containment spray train or RHR spray train ~~is not~~ cannot be restored to OPERABLE status within the ~~associated~~ required Completion Time, the ~~unit~~ plant must be ~~placed in~~ brought to a MODE in which the LCO does not apply. ~~This is accomplished by placing the unit in~~ To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and ~~in~~ to MODE 5 within 84 hours. ~~The 6 hours allotted to reach MODE 3 is a reasonable time, based on operating experience, to shut down the unit from full power conditions in an orderly manner and without challenging unit systems. The extended interval to reach MODE 5 allows additional time and is reasonable when considering that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.~~ The allowed Completion Time of [redacted] hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment of manual, power operated, and automatic valves, excluding check valves, in the Containment Spray System provides assurance that the proper flow path exists for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since they were verified in the correct position prior to being secured. ~~The 31 day Frequency of this SR was developed based on Inservice Testing Program requirements to perform valve testing at least once per 92 days.~~ This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned, are in the correct position.

SR 3.6.6.2

Verifying that each containment spray pump developed head at the flow test point is ~~≥~~ greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by Section XI of the ~~American Society of Mechanical Engineers (ASME)~~ Code (Ref. 5). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested

(continued)

BASES (continued)

on bypass flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.6.3 and SR 3.6.6.4 (W)

These SRs requires verification ~~ensure~~ that each automatic Containment spray valve actuates to its correct position and each containment spray pump starts on receipt of an actual or simulated containment spray actuation signal. The 18 month Frequency was developed considering it is prudent that many Surveillances be performed only during a ~~unit~~ outage. This is due to the ~~unit~~ conditions needed to perform the SR and the potential for unnecessary ~~unit~~ transients if the SR is performed with the reactor at power. Operating experience has shown these components usually pass the SR when performed on the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(5) Containment spray pump start verification may be performed by testing breaker actuation without pump start (breaker is racked out in its "test position") and observation of the local or remote pump start lights (breaker energization light).

The Surveillance of containment sump isolation valves is also required by SR 3.6.6.3. A single Surveillance may be used to satisfy both requirements.

SR 3.6.6.5

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and that spray coverage of the containment during an accident is not degraded. Because of the passive design of the nozzle, the first refueling outage and ~~a~~ 10 year ~~test~~ intervals are ~~is~~ considered adequate to detect obstruction of the spray nozzles.

(4) SR 3.6.6.6
The Surveillance descriptions from Bases 3.5.2 for SR 3.5.2.2 and 3.5.2.4 apply as applicable to the RHR spray system.

REFERENCES (R)

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Criteria for Nuclear Power Plants" ~~10 CFR 50, Appendix A, GDC 38, GDC 40, GDC 41, GDC 42, GDC 43, GDC 50, GDC 39,~~
2. Watts Bar FSAR, Section 6.2, "Containment Systems."

(continued)

BASES (continued)

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

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

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

(continued)

DRAFT

3.6 CONTAINMENT SYSTEMS

3.6.~~8~~ 7 Hydrogen Recombiners E ~~(Atmospheric, Subatmospheric, Ice Condenser, and Dual) (if permanently installed)~~

LCO 3.6.~~8~~ 7 Two hydrogen recombiners shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.~~8~~7.1	Perform a system functional test for each hydrogen recombiner.	[18] months
SR 3.6.~~8~~7.2	Visually examine each hydrogen recombiner enclosure and ensure there is no evidence of abnormal conditions.	[18] months
SR 3.6.~~8~~7.3	Perform a resistance to ground test ~~of~~ for each heater phase.	[18] months

B 3.6 CONTAINMENT SYSTEMS

B 3.6.~~8~~ 7 Hydrogen Recombiners ~~(Atmospheric, Subatmospheric, Ice Condenser, and Dual)~~ E

BASES

BACKGROUND

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

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

Design Basis Accident

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

APPLICABLE SAFETY ANALYSES

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

(continued)

BASES (continued)

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

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

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

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

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

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

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

The hydrogen recombiners are designed such that, with the conservatively calculated hydrogen generation rates discussed above, a single recombiner is capable of limiting the peak hydrogen concentration in containment to less than 4.0 v/o (Ref. 3). The Hydrogen Purge System is similarly designed such that one of two redundant trains is an adequate backup to the redundant hydrogen recombiners.

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

(continued)

BASES (continued)

LCO

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

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

APPLICABILITY

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

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

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

ACTIONS

A.1

With one containment hydrogen recombiner inoperable, the inoperable recombiner must be restored to OPERABLE status within 30 days. In this ~~C~~condition, the remaining OPERABLE hydrogen recombiner is adequate to perform the hydrogen control function. However, the overall reliability is reduced because a single failure in the OPERABLE recombiner could result in reduced hydrogen control capability. ~~The 30-day Completion Time is based on the low probability of the occurrence of a LOCA or SLB that would generate hydrogen in amounts capable of exceeding the flammability limit, the length of time after the event that operator action would be required to prevent hydrogen accumulation from exceeding~~

(continued)

BASES (continued)

The 30 day Completion Time is based on the small probability of a LOCA or SLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit); the amount of time available after a LOCA or SLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit; and the low probability of failure of the OPERABLE hydrogen recombiner.

~~this limit, and the low probability of failure of the OPERABLE hydrogen recombiner.~~

Required Action A.1 has been modified by a Note which states the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one recombiner is inoperable. ~~This allowance is provided because the probability of the occurrence of a LOCA that would generate hydrogen in amounts capable of exceeding the flammability limit is low, the probability of the failure of the OPERABLE recombiner is low and the length of time after a postulated LOCA before operator action would be required to prevent exceeding the flammability limit.~~ {copy note again}

B.1 and B.2

(E) ~~[Reviewers note - This Condition is only allowed for units with an alternate hydrogen control system acceptable to the technical staff.]~~

With two hydrogen recombiners inoperable, the ability to perform the hydrogen control function via alternate capabilities must be verified by administrative means within 1 hour. The alternate hydrogen control capabilities are provided by ~~[the containment hydrogen purge system/hydrogen recombiner/hydrogen ignitor system/hydrogen mixing system/containment air dilution system/containment inerting system].~~ the hydrogen mitigation system (1). The 1 hour Completion Time allows a reasonable period of time to verify that a loss of hydrogen control function does not exist. ~~[Reviewer note - The following is to be used if a non-Technical Specification alternate hydrogen control function is used to justify this Condition:~~ In addition, the alternate hydrogen control system capability must be verified every 12 hours thereafter to ensure its continued availability. ~~]~~ ~~[~~Both~~]~~ the ~~[~~initial~~]~~ verification ~~[~~and all subsequent verifications~~]~~ may be performed as an administrative check by examining logs or other information to determine the availability of the alternate hydrogen control system. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of the alternate hydrogen control system. If the ability to perform the hydrogen control function is maintained, continued operation is permitted with two ~~[~~hydrogen recombiners~~/hydrogen mixing trains/containment atmosphere~~

(continued)

BASES (continued)

~~dilution trains]~~ inoperable for up to 7 days. Seven days is a reasonable time to allow two [hydrogen recombiners/ ~~hydrogen mixing trains/containment atmosphere dilution trains]~~ to be inoperable because the hydrogen control function is maintained and the low probability of the occurrence of a LOCA that would generate hydrogen in the amounts capable of exceeding the flammability limit.

C.1

~~The unit must be placed in a MODE in which the LCO does not apply if~~ PLANT the inoperable hydrogen recombiner(s) cannot be restored to OPERABLE status ~~in~~ within the ~~associated~~ required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, ~~This is done by placing~~ the ~~unit in~~ PLANT must be brought to at least MODE 3 within 6 hours. ~~The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems.~~ PLANT

SURVEILLANCE REQUIREMENTS

SR 3.6.~~8~~7.1

Performance of a system functional test for each hydrogen recombiner ensures the recombiners are operational and can attain and sustain the temperature necessary for hydrogen recombination. In particular, this SR verifies that the minimum heater sheath temperature increases to ≥ 700°F in ≤ 90 minutes. After reaching 700°F, the power is increased to maximum power for approximately 2 minutes and power is verified to be ≥ 60 kW.

The 1~~8~~9-month Frequency for this SR was developed considering ~~such factors as the following:~~ ~~a.~~ the incidence of hydrogen recombiners failing the SR in the past is low.

~~b. Even when hydrogen recombiner failure has been detected, there has been, in all instances, a backup available either from the other recombiner or from a diverse system such as the Hydrogen Purge System; and~~

(continued)

BASES (continued)

c. Since the ~~hydrogen~~ recombiner is manually started many hours after a LOCA ~~occurs,~~ there is time available to either restore a recombiner to ~~OPERABLE~~ status, or activate an alternative.

SR 3.6.~~8~~7.2

This SR ensures there are no physical problems that could affect recombiner operation. Since the recombiners are mechanically passive, they are not subject to mechanical failure. The only credible failure involves loss of power, blockage of the internal flow, missile impact, etc.

A visual inspection is sufficient to determine abnormal conditions that could cause such failures. The 18-month Frequency for this SR was developed considering ~~such factors as the following:~~

~~a.~~ ~~T~~the incidence of hydrogen recombiners failing the SR in the past is low~~;~~.

b. Even when hydrogen recombiner failure has been detected, there has been, in all instances, a backup available either from the other recombiner or from a diverse system such as the Hydrogen Purge System; and

c. Since the hydrogen recombiner is manually started many hours after a LOCA occurs, there is time available to either restore a combiner to OPERABLE status or activate an alternative.

SR 3.6.~~8~~7.3

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

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

(continued)

BASES (continued) Title 10, Code of Federal Regulations, Part

REFERENCES (R)

1. ~~10 CFR~~ Title 10, Code of Federal Regulations, Part 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Power Reactors."
2. ~~10 CFR~~ Title 10, Code of Federal Regulations, Part 50, Appendix A, GOC 41, "Containment Atmosphere Cleanup."
3. Regulatory Guide 1.7, Revision ~~[1]~~ 2, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," U.S. Nuclear Regulatory Commission.
4. Watts Bar FSAR, Chapter ~~[15]~~, "Accident Analyses."

(continued)

DRAFT

3.6 CONTAINMENT SYSTEMS

3.6.~~10~~ 8 * Hydrogen Mitigation ~~Ignition~~ System T (HMS) ~~(HIS) (Ice Condenser)~~ E

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

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One HMS ~~HIS~~ train inoperable.	A.1	Restore HMS ~~HIS~~ train to OPERABLE status.	7 days
	<u>OR</u>		
	A.2	Perform SR 3.6.10.1 on the OPERABLE train.	Once per 7 days
B. One containment region with no OPERABLE hydrogen ignitor.	B.1	Restore one hydrogen ignitor in the affected containment region to OPERABLE status.	7 days
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.~~10~~.1 8	Energize each ~~HIS~~ HMS train power supply breaker and verify ≥ [~~32~~] ignitors are energized in each train. 33* P	92 days
SR 3.6.~~10~~.2 8	Verify at least one hydrogen ignitor OPERABLE in each containment region.	92 days
SR 3.6.~~10~~.3 8	Energize each hydrogen ignitor and verify temperature ≥ [1700]°F.	[18 months]

DRAFT

B 3.6 CONTAINMENT SYSTEMS

B 3.6.~~10~~ 8 Hydrogen ~~Ignition~~ Mitigation System ~~(HIS)~~ (HMS) T ~~(Ice Condenser)~~ E

Replace HIS with HMS T

BASES

BACKGROUND

in a post accident environment

The HMS consists of two groups of 34 igniters distributed throughout the containment.

P

The ~~HIS~~ HMS reduces the potential for breach of primary containment due to a hydrogen/oxygen reaction in a post accident environment. The HIS is required by 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors" (Ref. 1), and Appendix A, GDC 41, "Containment Atmosphere Cleanup" (Ref. 2), to reduce the hydrogen concentration in the primary containment following a degraded core accident. The HIS must be capable of handling an amount of hydrogen equivalent to that generated from a metal water reaction involving 75% of the fuel cladding surrounding the active fuel region (excluding the plenum volume).

10 CFR 50.44 (Ref. 1) requires units with ice condenser containments to install suitable hydrogen control systems that would accommodate an amount of hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding with water. The ~~HIS~~ HMS provides this required capability. This requirement was placed on ice condenser units because of their small containment volume and low design pressure (compared with pressurized water reactor dry containments). Calculations indicate that if hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding with water were to collect in the primary containment, the resulting hydrogen concentration would be far above the lower flammability limit such that, if ignited from a random ignition source, the resulting hydrogen burn would seriously challenge the containment and safety systems in the containment.

The ~~HIS~~ HMS is based on the concept of controlled ignition using thermal ignitors, designed to be capable of functioning in a post accident environment, seismically supported, and capable of actuation from the control room. A total of P 68 ~~[64]~~ ignitors are distributed throughout the various regions of containment in which hydrogen could be released or to which it could flow in significant quantities. The ignitors are arranged in two independent trains such that each containment region has at least two ignitors, one from each train, controlled and powered redundantly so that ignition would occur in each region even if one train failed to energize.

(continued)

BASES (continued)

When the ~~HIS~~ HMS is initiated, the ignitor elements are energized and heat up to a surface temperature ≥ [1700] °F. At this temperature they ignite the hydrogen gas that is present in the airspace in the vicinity of the ignitor. The HIS depends on the dispersed location of the ignitors so that local pockets of hydrogen at increased concentrations would burn before reaching a hydrogen concentration significantly higher than the lower flammability limit. Hydrogen ignition in the vicinity of the ignitors is assumed to occur when the local hydrogen concentration reaches [8.0] volume percent (v/o) and results in [85]% of the hydrogen present being consumed.

APPLICABLE SAFETY ANALYSES

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

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

by only E

LCO

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

(continued)

BASES (continued)

P

For this unit, an OPERABLE ~~HIS~~ HMS train consists of ~~32~~ 33 of ~~33~~ 34 ignitors energized on the train. T

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

APPLICABILITY

Requiring OPERABILITY in MODES 1 and 2 for the HIS ensures its immediate availability after safety injection and scram actuated on a LOCA initiation. In the post accident environment, the two HIS subsystems are required to control the hydrogen concentration within ~~primary~~ containment to near its flammability limit of 4.1 v/o assuming a worst case single failure. This prevents overpressurization of containment and damage to safety related equipment and instruments located within ~~primary~~ containment.

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

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

ACTIONS

A.1

With one HIS train inoperable, the inoperable train must be restored to OPERABLE status within 7 days. The 7-day Completion Time is based on the low probability of the occurrence of a degraded core event that would generate hydrogen in amounts equivalent to a metal-water reaction of 75% of the core cladding, the length of time after the event that operator action would be required to prevent hydrogen

(continued)

BASES (continued)

accumulation from exceeding this limit, and the low probability of failure of the OPERABLE ~~HIS~~ HMS (T) train.

INSERT FROM PAGE B 3.6-4A

B.1

Condition B is one containment region with no OPERABLE hydrogen ignitor. The only way this ~~C~~condition can exist in only one containment region is for each train of HIS to have only one ignitor inoperable, and the one inoperable ignitor in each train to be in the same containment region. In this Condition both trains of HIS would be OPERABLE per SR 3.6.10.1, and there would be two OPERABLE ignitors in all other containment regions, including the regions adjacent to the region with no OPERABLE ignitors. Furthermore, even if one train of HIS is inoperable (putting the unit into Conditions A and B simultaneously), the OPERABLE train would provide at least one ignitor in each adjacent region. Lastly, if both trains of HIS are inoperable, or if more than one containment region has no OPERABLE ignitor, LCO 3.0.3 would be immediately entered. Thus, while in Condition B, there would always be ignition capability in the adjacent containment regions that would provide redundant capability by flame propagation to the region with no OPERABLE ignitors.

Required Action B.1 calls for the restoration of one hydrogen ignitor in each region to OPERABLE status within 7 days. The ~~7~~-day Completion Time is based on the same reasons given under Required Action A.1.

C.1

If the HMS subsystem(s) cannot be restored to OPERABLE status within the required Completion Time, ~~the~~ plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours

~~The unit must be placed in a MODE in which the LCO does not apply if the HIS subsystem(s) cannot be restored to OPERABLE status in the associated Completion Time. This is done by placing the unit in at least MODE 3 within 6 hours.~~ The allowed Completion Time is reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner and without challenging unit systems.

(continued)

INSERT

A.2

If the inoperable HMS train cannot be restored within the Completion time of 7 days, it is acceptable to continue operation provided SR 3.6.8.1 is performed on the OPERABLE train with the specified frequency of 7 days. This SR verifies at least one hydrogen igniter OPERABLE in each containment zone. The igniters are simple glow plug devices, not likely to fail between surveillance periods with a frequency of 7 days. As such, the frequency is based on engineering judgement, and has been shown to be acceptable through operating experience.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.~~10~~8.1 (P)

This SR confirms that ≥ ~~32~~ 33 of ~~33~~ 34 hydrogen ignitors can be successfully energized in each train. The ignitors a.e simple resistance elements. Therefore, energizing provides assurance of OPERABILITY. The allowance of one inoperable hydrogen ignitor is acceptable because, although one inoperable hydrogen ignitor in a region would compromise redundancy in that region, the containment regions are interconnected so that ignition in one region would cause burning to progress to the others (i.e., there is overlap in each hydrogen ignitor's effectiveness between regions). The Frequency of 92 days is based on the Inservice Testing Program requirements for determining equipment OPERABILITY, and has been shown to be acceptable through operating experience.

SR 3.6.~~10~~8.2

This SR confirms that the two inoperable hydrogen ignitors allowed by SR 3.6.10.1 (i.e., one in each train) are not in the same containment region. The containment regions and hydrogen igniter locations are provided in Reference 4. (1) Δ As such, failure of this SR results in entry into Condition B. See ~~ACTION~~ Required Action B.1 for a discussion regarding how Conditions A and B and the associated Required Actions ~~assure~~ ensure that no more than one containment region can be without an OPERABLE hydrogen ignitor for any length of time without commencing a shutdown.

SR 3.6.~~10~~8.3

A more detailed functional test is performed every 18 months to verify system OPERABILITY. Each glow plug is visually examined to ensure that it is clean, and that the electrical circuitry is energized. All ignitors (glow plugs), including normally inaccessible ignitors, are visually checked for a glow to verify that they are energized. Additionally, the surface temperature of each glow plug is measured to be ≥ [1700]°F to demonstrate that a temperature sufficient for ignition is achieved. The 18-month Frequency was developed considering it is prudent that many Surveillances be performed only during a unit outage. This

(continued)

BASES (continued)

is due to the unit conditions needed to perform the SR and the potential for an unplanned unit transient if the SR is performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed on the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES (R)

1. Title 10, Code of Federal Regulations, Part ~~10 CFR~~ 50.44, "Standards for Combustible Gas Control Systems in Light Water-Cooled Power Reactors."
2. Title 10, Code of Federal Regulations, Part ~~10 CFR~~ 50, Appendix A, General Design Criterion 41, "Containment Atmosphere Cleanup."
3. Watts Bar FSAR, Section ~~[~~6.2~~]~~, "Containment Systems."

(1) 4. Watts Bar FSAR, Section 6.2.5A.2, "Hydrogen Mitigation System Description."

(continued)

DRAFT

3.6 CONTAINMENT SYSTEMS

3.6.~~13~~ 9 ~~Shield Building Air Cleanup System SBACS (Dual and Ice Condenser)~~ Emergency Gas Treatment System (EGTS) E

LCO 3.6.~~13~~ 9 Two ~~SBACS~~ EGTS trains shall be OPERABLE. T

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One ~~SBACS~~ EGTS train inoperable.	A.1	Restore ~~SBACS~~ EGTS train to OPERABLE status.	7 days
B. Annulus pressure not within limits.	B.1	Restore annulus pressure within limits.	8 hours
~~B.~~ C. Required Action and associated Completion Time not met.	~~B~~.1 C.1	Be in MODE 3.	6 hours
	AND		
	~~B~~.2 C.2	Be in MODE 5.	36 hours

2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.~~13~~ 9.1	Operate each ~~SBACS~~ EGTS train for ~~[~~ ≥ 10 continuous hours with heaters operating ~~or (for systems without heaters) ≥ 15 minutes]~~.	31 days

NOTE
Annulus pressure requirement is not applicable during ventilating operations, required annulus entries, or Auxilary Building isolations not exceeding 1 hour in duration.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.~~13~~ 9.2	Perform required ~~SBACS~~ EGTS (T) filter testing in accordance with the Ventilation Filter Testing Program (VFTP). (W)	In accordance with the VFTP ~~Ventilation Filter Testing Program~~
SR 3.6.~~13~~ 9.3 (P) (8)	Verify each ~~SBACS~~ EGTS (T) train actuates on an actual or simulated actuation signal and achieves a system flow of 4000 ±10% cfm within 20 seconds.	~~[18]~~ months
(7) ~~SR 3.6.13.4~~ 9	~~Verify each SBACS~~ ~~EGTS~~ (T) ~~filter bypass damper can be opened.~~	~~[18] months~~
SR 3.6.~~13~~ 9.~~5~~ 4	Verify each ~~SBACS~~ EGTS (T) train produces ~~a~~ an annulus pressure equal to or more negative than ~~[~~-0.5~~]~~ (E) inch water gauge ~~in the annulus within [22] seconds after a start signal.~~ ~~160*~~ (8) with respect to the El. 772 mechanical equipment room with an inleakage of ≤ 250 cfm at a system flow of 4000 ±10% cfm. (P)	~~[18]~~ months on a STAGGERED TEST BASIS
* SR 3.6.9.5 (2) (P)	Verify annulus pressure ~~≥~~ (6) equal to or more negative than -5 inches water gauge ~~vacuum~~ with respect to ~~penetration room on El. 737.~~ Atmosphere.	24 hours

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.~~13~~ 9 ~~Shield Building Air Cleanup System (SBACS) (Dual and Ice Condenser)~~ Emergency Gas Treatment System (EGTS) (T)

Replace SBACS with EGTS

BASES

BACKGROUND

The ~~SBACS~~ EGTS is required by 10 CFR 50 Appendix A, CDC 41, "Containment Atmosphere Cleanup" (Ref. 1), to ensure that radioactive materials that leak from the primary containment into the shield building (secondary containment) following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment.

The ~~(dual/ice condenser)~~ containment~~s~~ ~~have~~ has a secondary containment called the shield building, which is a concrete structure that surrounds the steel primary containment vessel. Between the containment vessel and the shield building inner wall is an annular space that collects any containment leakage that may occur following a loss of coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the steel containment vessel.

The ~~SBACS~~ EGTS establishes a negative pressure in the annulus between the Shield Building and the steel containment vessel. Filters in the system then control the release of radioactive contaminants to the environment. Shield building OPERABILITY is required to ensure retention of primary containment leakage and proper operation of the ~~SBACS~~ EGTS.

6 The ~~SBACS~~ EGTS consists of two separate and redundant trains. Each train includes a heater, ~~[cooling coils,]~~ a prefilter, a moisture separator~~s~~, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of radioiodines, and fan. Ductwork, valves, and/or dampers, and instrumentation also form part of the system. The moisture separators function to reduce the moisture content of the airstream. A second bank of HEPA filters follows the adsorber sections to collect carbon fines. ~~and provide backup in case of failure of the main HEPA filter bank~~. Only the upstream HEPA filter and the charcoal adsorber section are credited in the analysis. The system initiates and maintains a negative air pressure in the shield building by means of filtered exhaust ventilation of the shield building following receipt of a safety injection signal. The system is described in Reference 2.

6

(continued)

BASES (continued)

Cross over flow ducts are provided between the two trains to allow the active train to draw air through the inactive train and cool the air to keep the charcoal beds on the inactive train becoming too hot due to absorption of fission products.

The prefilters remove ~~any~~ large particles in the air, and the moisture separators remove ~~any~~ entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal absorbers. Heaters may be included to reduce the relative humidity of the air stream on systems that operate in high humidity. Continuous operation of each train, for at least 10 hours per month, with heaters on, reduces moisture buildup on their HEPA filters and absorbers. ~~[The cooling coils cool the air to keep the charcoal beds from becoming too hot due to absorption of fission product.]~~

~~During normal operation, the Shield Building Cooling System is aligned to bypass the SBACS's HEPA filters and charcoal absorbers. For SBACS operation following a DBA, however, the bypass dampers automatically reposition to draw the air through the filters and absorbers.~~

The containment annulus vacuum fans maintain the annulus at -5 inches water gauge vacuum during normal operations. During accident conditions, they are isolated from the air cleanup portion of the system. (2)

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

APPLICABLE SAFETY ANALYSES

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

WBN-specific write up (3)

~~The modeled SBACS EGTS actuation in the safety analysis is based upon a worst case response time following a safety injection initiated at the limiting setpoint. The total response time, from exceeding the signal setpoint to attaining the negative pressure of [0.5] inch water gauge in the shield building, is [22] seconds. This response time is composed of signal delay, diesel generator startup and sequencing time, system startup time, and time for the system to attain the required pressure after starting.~~

The ~~SBACS~~ EGTS satisfies Criterion 3 of the NRC Policy Statement.

(continued)

APPLICABLE SAFETY ANALYSES (continued)

The modeled EGTS actuation in the safety analysis assumes an initial annulus vacuum pressure of -5.0 inches water gauge upon receipt of the Phase A isolation signal. The fans automatically start within 20 seconds (excluding 10 seconds for diesel generator start) after receipt of the initiating signal. The analysis shows that after an initial step increase, the pressure will rise to a peak value of -0.13 inches water gauge in approximately 90 seconds after the LOCA and does not go positive. The annulus pressure then returns to the post accident setpoint of -0.5 inches water gauge.

BASES (continued)

LCO

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

T

APPLICABILITY

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

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

ACTIONS

A.1

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

(continued)

2 BASES (continued)

B.1 WBN-specific write-up

C.1 and C.2

If ~~the SBACS~~ an EGTS train cannot be restored to OPERABLE status or annulus pressure restored to within limits ~~within~~ by the required ~~in the associated~~ Completion Time, the ~~unit~~ plant must be ~~placed in~~ brought to a MODE in which the LCO does not apply. ~~This is done by placing the unit in~~ To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and ~~in~~ to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required ~~MODES~~ plant conditions from full power conditions in an orderly manner and without challenging ~~unit~~ plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.~~13~~9.1

~~Operating each SBACS train for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action.~~ For systems with heaters, operation with the heaters on (automatic heater cycling to maintain temperature) for ≥ 10 hours eliminates moisture on the adsorbers and HEPA filters. Experience from filter testing at operating units indicates that the 10 hour period is adequate for moisture elimination on the adsorbers and HEPA filters. The 31-day Frequency was developed in consideration of the known reliability of fan motors and controls, the two-train redundancy available, ~~and the iodine removal capability of the Containment Spray System.~~ 5

SR 3.6.~~13~~9.2

4 ~~This SR verifies that the required EVS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. The EVS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 4).~~ The Ventilation Filter Testing Program (VFTP) (Specification 5.7.2) encompasses all the EGTS filter tests consistent with Regulatory Guide 1.52 (Ref. 4). The [VFTP] includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the [VFTP].

(continued)

B.1

With annulus pressure not within limits during normal operation, the initial conditions of the accident analysis are not met. The Completion Time of 8 hours is based on engineering judgment. A note has been provided which makes the requirement to maintain the annulus pressure within limits not applicable during venting operations, required annulus entries, or Auxiliary Building isolations not exceeding 1 hour in duration.

BASES (continued)

SR 3.6.~~13~~9.3

P

The automatic ~~start~~up ensures that each SBACS train responds properly and achieves rated flow within the time assumed in the accident analysis which ensures annulus pressure remains negative. The 18 month Frequency was developed considering it is prudent that this Surveillance be performed only during a unit outage. This is due to the unit conditions needed to perform the SR and the potential for unplanned unit transients if the SR is performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed on the 18-month Frequency. Therefore the Frequency was concluded to be acceptable from a reliability standpoint. Furthermore, the SR interval was developed considering that the SBACS equipment OPERABILITY is demonstrated on a 31 day Frequency by SR 3.6.~~13~~9.1.

SR 3.6.~~13~~9.4

7

The filter bypass dampers are tested to verify OPERABILITY. The dampers are in the bypass position during normal operation and must reposition for accident operation to draw air through the filters. The 18 month Frequency is considered to be acceptable based on damper reliability and design, mild environmental conditions in the vicinity of the dampers, and the fact that operating experience has shown that the dampers usually pass the 18-month SR.

SR 3.6.~~13~~9.~~5~~4 4

6

The proper functioning of the fans, dampers, filters, absorbers, etc. as a system is verified by the ability to achieve total system flow rate of 4000 CFM ± 10%, maintain the maximum inleakage ≤ 250 CFM, and produce the required negative pressure equal to or more negative than ~~≥~~ -0.5 inch water gauge with respect to the El. 772 Mechanical Equipment Room during testing operation. ~~during test operation within one minute.~~ The negative pressure assures that the building is adequately sealed and that leakage from the building will be prevented, since outside air will be drawn in by the low pressure. The negative pressure must be established and verified under steady state conditions at the specified total system flow and leakage requirements ~~within the time limit~~ to ensure that no significant quantity of radioactive materials leak from the shield building ~~prior to developing the negative pressure.~~

The 18 month on a STAGGERED TEST BASIS Frequency is consistent with Regulatory Guide 1.52 (Ref. 4) guidance for functional testing.

(continued)

BASES (continued)

REFERENCES (R)

1. ~~10 CFR 50~~ Title 10, Code of Federal Regulations, Part 50, Appendix A, ~~GDC 41.~~ General Design Criteria 41, "Containment Atmosphere Cleanup."
2. Watts Bar FSAR, Section [6.5], "Fission Product Removal + Control Systems"
3. Watts Bar FSAR, Section [15], "Accident Analysis."
4. Regulatory Guide 1.52, Revision ~~[1]~~ 2, "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Absorption Units of Light-Water Cooled Nuclear Power Plants."

SR 3.6.9.~~6~~ 5

Verifying that shield building annulus pressure is within limits ensures that operation remains within the limits assumed in the containment analysis. The 24 hour Frequency of this SR was developed considering operating experience related to shield building annulus pressure variations and pressure instrument drift during the applicable MODES.

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.~~14~~ 10 Air Return System (ARS) (E) ~~(Ice Condenser)~~

LCO 3.6.~~14~~ 10 Two ARS trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ARS train inoperable.	A.1 Restore ARS train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. **AND** B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

or simulated actuation

SURVEILLANCE		FREQUENCY
SR 3.6.~~14~~ 10.1	Verify each ARS fan starts on an actual or simulated actuation signal, after a delay of ≥ ~~[9.0]~~ 8.0 (P) minutes and ≤ ~~[11.0]~~ (P) 10.0 minutes, and operates for ≥ 15 minutes.	[92 days]

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.~~14~~.2 10	Verify with the ARS fan dampers closed, each ARS fan motor current is ≥ ~~[20.5]~~ 54 and ≤ ~~[35.5]~~ 94 amps ~~[when the fan speed is ≥ [840] rpm and ≤ [900] rpm]~~. (4) (P) (P)	92 days
SR 3.6.~~14~~.3 10	Verify with the ARS fan not operating, ~~that~~ each ARS fan damper opens when ~~≥ [11.0] lb~~ ≤ 150 in•lbs ~~are~~ is applied to the counterweight. (P)	92 days
~~SR 3.6.14.4~~	~~[Verify each motor operated valve in the hydrogen collection header opens on an actual or simulated actuation signal after a delay of ≥ [9.0] minutes and ≤ [11.0] minutes.~~	~~92 days]~~ N/A

(1)

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.~~14~~ 10 Air Return System (ARS) ~~(Ice Condenser)~~ (E)

BASES

BACKGROUND

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

(1)

The ARS draws air from the dome of the containment vessel, from the reactor cavity, and from the ten dead ended (pocketed) spaces in the containment where there is potential for the accumulation of hydrogen

(E)

The ARS provides post accident for hydrogen mitigation ~~hydrogen~~ mixing of air in selected areas of containment. ~~The associated Hydrogen Skimmer System consists of hydrogen collection headers routed to potential hydrogen pockets in containment, terminating on the suction side of either of the two ARS fans at the header isolation valves.~~ The minimum design flow from each potential hydrogen pocket is sufficient to limit the local concentration of hydrogen.

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

8 to 10 (P)

The ARS fans are automatically started and the hydrogen collection header isolation valves are opened by the containment pressure High-High signal ~~10~~ minutes after the containment pressure reaches the pressure setpoint. The time delay ensures that no energy released during the initial phase of a DBA will bypass the ice bed through the ARS fans ~~or Hydrogen Skimmer System.~~

(1)

into the upper containment compartment.

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

(continued)

BASES (continued)

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

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

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

APPLICABLE SAFETY ANALYSES

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

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

(continued)

BASES (continued)

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

T
INSERT FROM PAGE B 3.6-3A

6 ~~The analysis for minimum internal containment pressure (i.e., maximum external differential containment pressure) assumes inadvertent simultaneous actuation of both the ARS and the Containment Spray System.~~ 5 ~~The containment vacuum relief valves are designed to accommodate inadvertent actuation of either or both systems.~~

The modeled ARS actuation from the containment analysis is based upon a response time associated with exceeding the containment pressure High-High signal setpoint to achieving full ARS air flow. A delayed response time initiation provides conservative analyses of peak calculated containment temperature and pressure responses. The ARS total response time of ~~600~~ 540 ± 60 seconds consists of the built-in signal delay. P

The ARS satisfies Criterion 3 of the NRC Policy Statement.

LCO

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

APPLICABILITY

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

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature

(continued)

INSERT

The maximum peak containment atmosphere temperature of 32~~5~~6°F results from the SLB analysis and was calculated to exceed the containment design temperature of 250° F for a short time. This analysis is discussed in the Bases for LCO 3.6.5, "Containment Air Temperature". Thermal analyses show that the time interval during which the containment atmosphere temperature exceeds the containment design temperature is short enough that equipment surface temperatures remain below the design temperature. Therefore, it is concluded that the calculated transient containment atmosphere temperatures are acceptable for the SLB.

limitations of these MODES. Therefore, the ARS is not required to be OPERABLE in these MODES.

ACTIONS

A.1

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

(1)

B.1 and B.2 △

If the ARS train cannot be restored to plant OPERABLE status ~~in the associated~~ within the required Completion Time, the ~~unit~~ plant must be ~~placed in~~ brought to a MODE in which the LCO does not apply. ~~This is done by placing the unit~~ To achieve this status, the plant in at least MODE 3 within 6 hours and ~~into~~ to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required ~~MODES~~ plant conditions from full power conditions in an orderly manner and without challenging ~~unit~~ plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.14.1

Verifying that each ARS fan starts on an actual or simulated actuation signal, After a delay of ≥ 8.0 minutes and ≤ 10.0 minutes, and operates for ≥ 15 minutes is sufficient to ensure that all fans are OPERABLE and that all associated controls and time delays are functioning properly. ~~It also ensures that blockage, fan and/or motor failure, or excessive vibration can be detected for corrective action.~~ The 92-day Frequency was developed considering the known reliability of fan motors and controls and the two train redundancy available.

(2)

(P)

BASES (continued)

SR 3.6.14.2

backdraft

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

SR 3.6.14.3

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

~~SR 3.6.14.4~~

~~Verifying the OPERABILITY of the motor operated valve in the Hydrogen Skimmer System collection header to the lower containment compartment provides assurance that the proper flow path will exist when the valve receives an actuation signal. This Surveillance also confirms that the time delay to open is within specified tolerances. The 92-day Frequency was developed considering the known reliability of the motor operated valves and controls and the two train redundancy available. Operating experience has also shown this Frequency to be acceptable.~~

REFERENCES

1. Watts Bar FSAR, Section [6.2], "Containment Systems."
2. Title 10, Code of Federal Regulations, Part 50, ~~10 CFR 50~~, Appendix K, "ECCS Evaluation Models."

~~(continued)~~

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3.6 CONTAINMENT SYSTEMS

3.6.~~15~~ 11 Ice Bed E ~~(Ice Condenser)~~

LCO 3.6.~~15~~ 11 The ice bed shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Ice bed inoperable.	A.1 Restore ice bed to OPERABLE status.	48 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.~~15~~.1 11 Verify maximum ice bed temperature is ≤ [27]°F.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.~~15~~ 11 .2	Verify total weight of stored ice is ≥ ~~[2,721,600]~~ 2,360,875 P lb by: a. Weighing a representative sample of ≥ 144 ice baskets and verifying each basket contains ≥ ~~[1400]~~ 1214 P lb of ice; and b. Calculating total weight of stored ice, at a 95% confidence level, using all ice basket weights determined in SR 3.6.~~16~~ 11 .2.a.	9 months
SR 3.6.~~15~~ 11 .3	Verify azimuthal distribution of ice at a 95% confidence level by subdividing weights, as determined by SR 3.6.~~16~~ 11 .2.a, into the following groups: a. Group 1—bays 1 through 8; b. Group 2—bays 9 through 16; and c. Group 3—bays 17 through 24. The average ice weight of the sample baskets in each group from radial rows 1, 2, 4, 6, 8, and 9 shall be ≥ ~~[1400]~~ 1214 P lb.	9 months
SR 3.6.~~15~~ 11 .4	Verify, by visual inspection, accumulation of ice or frost on structural members comprising flow channels through the ice condenser is ≤ [0.38] inches thick.	9 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.~~15~~.5 11	Verify by chemical analyses of ~~≥ 9~~ at least nine representative samples of stored ice: a. Boron concentration is ≥ [1800] ppm; and AND b. pH ≥ [9.0] and ≤ [9.5].	[18] months
SR 3.6.~~15~~.6 11	Visually inspect, for detrimental structural wear, cracks, corrosion, or other damage, ~~2~~ two ice baskets from each azimuthal group of bays. See SR 3.6.~~15~~ 11.3.	40 months

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.~~15~~ 11 Ice Bed ~~(Ice Condenser)~~ E

BASES

BACKGROUND

The ice bed consists of over ~~2,721,600~~ 2,360,875 P lb of ice stored in baskets within the ice condenser. Its primary purpose is to provide a large heat sink in the event of a release of energy from a Design Basis Accident (DBA) in containment. The ice would absorb energy and limit containment peak pressure and temperature during the accident transient. Limiting the pressure and temperature ~~would ensure containment integrity and~~ reduces the release of fission product radioactivity from containment to the environment in the event of a DBA.

The ice condenser is an annular compartment enclosing approximately 300° ~~degrees~~ of the perimeter of the upper containment compartment, but penetrating the operating deck so that a portion extends into the lower containment compartment. The lower portion has a series of hinged doors exposed to the atmosphere of the lower containment compartment which, for normal ~~unit~~ plant operation, are designed to remain closed. At the top of the ice condenser is another set of doors exposed to the atmosphere of the upper compartment, which also remain closed during normal ~~unit~~ plant operation. Intermediate deck doors, located below the top deck doors, form the floor of a plenum at the upper part of the ice condenser. These doors also remain closed during normal ~~unit~~ plant operation. The upper plenum area is used to facilitate surveillance and maintenance of the ice bed.

The ice baskets held in the ice bed within the ice condenser are arranged to promote heat transfer from steam to ice. This arrangement enhances the ice condenser's primary function of condensing steam and absorbing heat energy released to the containment during a DBA.

In the event of a DBA, the ice condenser inlet doors (located below the operating deck) open due to the pressure rise in the lower compartment. This allows air and steam to flow from the lower compartment into the ice condenser. The resulting pressure increase within the ice condenser causes the intermediate deck doors and the top deck doors to open, which allows the air to flow out of the ice condenser into the upper compartment. Steam condensation within the ice

(continued)

BASES (continued)

condenser limits the pressure and temperature buildup in containment. A divider barrier separates the upper and lower compartments and ensures that the steam is directed into the ice condenser.

The ice, together with the containment spray, is adequate to absorb the initial blowdown of steam and water from a DBA and the additional heat loads that would enter containment during several hours following the initial blowdown. The additional heat loads would come from the residual heat in the reactor core, the hot piping and components, and the secondary system, including the steam generators. During the post blowdown period, the Air Return System (ARS) returns upper compartment air through the divider barrier to the lower compartment. This serves to equalize pressures in containment and to continue circulating heated air and steam from the lower compartment through the ice condenser where the heat is removed by the remaining ice.

As ice melts, the water passes through the ice condenser floor drains into the lower compartment. Thus, a second function of the ice bed is to be a large source of borated water (via the containment sump) for long term Emergency Core Cooling System (ECCS) and Containment Spray System heat removal functions in the recirculation mode.

A third function of the ice bed and melted ice is to remove fission product iodine that may be released from the core during a DBA. Iodine removal occurs during the ice melt phase of the accident and continues as the melted ice is sprayed into the containment atmosphere by the Containment Spray System. The ice is adjusted to an alkaline pH that facilitates removal of radioactive iodine from the containment atmosphere. The alkaline pH also minimizes the occurrence of the chloride and caustic stress corrosion on mechanical systems and components exposed to ECCS and Containment Spray System fluids in the recirculation mode of operation.

It is important for the ice to be uniformly distributed around the 24 ice condenser bays and for open flow paths to exist around ice baskets. This is especially important during the initial blowdown so that the steam and water mixture entering the lower compartment do not pass through only part of the ice condenser, depleting the ice there while bypassing the ice in other bays.

(continued)

BASES (continued)

Two phenomena that can degrade the ice bed during the long service period are:

a. Loss of ice by melting or sublimation; and

b. Obstruction of flow passages through the ice bed due to buildup of frost or ice. Both of these degrading phenomena are reduced by minimizing air leakage into and out of the ice condenser.

The ice bed limits the temperature and pressure that could be expected following a DBA, thus limiting leakage of fission product radioactivity from containment to the environment.

APPLICABLE SAFETY ANALYSES

The limiting DBAs considered relative to containment temperature and pressure are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. DBAs are not assumed to occur simultaneously or consecutively.

Although the ice condenser is a passive system that requires no electrical power to perform its function, the containment Spray System and the ARS also function to assist the ice bed in limiting pressures and temperatures. Therefore, the postulated DBAs are analyzed in regards to containment Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active failure and results in one train each of the Containment Spray System and ARS being inoperable.

The limiting DBA analyses (Ref. 1) show that the maximum peak containment pressure results from the LOCA analysis, and is calculated to be less than the containment design pressure.

For certain aspects of the transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the ECCS during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is

(continued)

BASES (continued)

calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures, in accordance with 10 CFR 50, Appendix K (Ref. 2).

The maximum peak containment atmosphere temperature results from the SLB analysis and is discussed in the Bases for Specification 3.6.5, "Containment Air Temperature."

In addition to calculating the overall peak containment pressures, the DBA analyses include calculation of the transient differential pressures that occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand these local transient pressure differentials for the limiting DBAs.

The ice bed satisfies Criterion 3 of the NRC Policy Statement.

LCO

The ice bed LCO requires the existence of the required quantity of stored ice, appropriate distribution of the ice and the ice bed, open flow paths through the ice bed, and appropriate chemical content and pH of the stored ice. The stored ice functions to absorb heat during a DBA, thereby limiting containment air temperature and pressure. The chemical content and pH of the ice provide core shutdown margin (boron content) and remove radioactive iodine from the containment atmosphere when the melted ice is recirculated through the ECCS and the Containment Spray System, respectively.

APPLICABILITY

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

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

(continued)

BASES (continued)

ACTIONS

A.1

If the ice bed is inoperable, it must be restored to OPERABLE status within 48 hours. The Completion Time was developed based on operating experience, which confirms that due to the very large mass of stored ice, the parameters comprising OPERABILITY do not change appreciably in this time period. Because of this fact, the Surveillance Frequencies are long (months), except for the ice bed temperature, which is checked every 12 hours. If a degraded condition is identified, even for temperature, with such a large mass of ice it is ~~not possible~~ improbable 1 for the degraded condition to significantly degrade further in a 48-hour period. Therefore, 48 hours is a reasonable amount of time ~~to allow~~ to correct a degraded condition before initiating a shutdown.

B.1 and B.2

If the ice bed cannot be restored to OPERABLE status within ~~limits in~~ the required ~~associated~~ Completion Time, the plant ~~unit~~ must be ~~placed in~~ brought to a MODE in which the LCO does not apply. ~~This is done by placing the unit in~~ To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and ~~in~~to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required ~~MODES~~ plant conditions from full power conditions in an orderly manner without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.~~15~~11.1

Verifying that the maximum temperature of the ice bed is ≤ [27]°F ensures that the ice is kept well below the melting point. The 12-hour Frequency was based on operating experience, which confirmed that, due to the large mass of stored ice, it is ~~not possible~~ improbable 1 for the ice bed temperature to degrade significantly in a 12-hour period, and on assessing the proximity of the LCO limit to the melting temperature.

Furthermore, the 12-hour Frequency is considered adequate in view of indications in the control room, including the

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BASES (continued)

alarm, to alert the operator to an abnormal ice bed temperature condition. This SR may be satisfied by use of the Ice Bed Temperature Monitoring System.

SR 3.6.~~15~~11.2 E

The weighing program is designed to obtain a representative sample of 6 the ice baskets. The representative sample shall include ~~six~~ baskets from each of the 24 ice condenser bays, and consist of one basket from radial rows 1, 2, 4, 6, 8, and 9. If no basket from a designated row can be obtained for weighing, a basket from the same row of an adjacent bay shall be weighed.

The rows chosen include the rows nearest the inside and outside walls of the ice condenser (rows 1 and 2, and 8 and 9, respectively), where heat transfer into the ice condenser is most likely to influence melting or sublimation. Verifying the total weight of ice ensures that there is adequate ice to absorb the required amount of energy to mitigate the DBAs.

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If a basket is found to contain < ~~[1400]~~ 1214 lb of ice, a representative sample of 20 additional baskets from the same bay shall be weighed. The average weight of ice in these 21 baskets (the discrepant basket and the 20 additional baskets) shall be ≥ ~~[1400]~~ 1214 lb at a 95% confidence level.

Weighing 20 additional baskets from the same bay in the event a Surveillance reveals that a single basket contains < ~~[1400]~~ 1214 lb ensures that no local zone exists that is grossly deficient in ice. Such a zone could experience early melt-out during a DBA transient, creating a path for steam to pass through the ice bed without begin condensed. The Frequency of 9 months was based on ice storage tests and the allowance built into the required ice mass over and above the mass assumed in the safety analyses. Operating experience has verified that, with the 9 month Frequency, the weight requirements are maintained with no significant degradation between Surveillances.

(continued)

BASES (continued)

SR 3.6.~~15~~ 11.3

This SR ensures that the azimuthal distribution of ice is reasonably uniform, by verifying that the average ice weight in each of three azimuthal groups of ice condenser bays is within the limit. The Frequency of 9 months was based on ice storage tests and the allowance built into the required ice mass over and above the mass assumed in the safety analyses. Operating experience has verified that, with the 9-month Frequency, the weight requirements are maintained with no significant degradation between Surveillances.

SR 3.6.~~15~~ 11.4

This SR ensures that the flow channels through the ice condenser have not accumulated an excessive amount of ice or frost blockage. The visual inspection shall be made for two or more flow channels per ice condenser bay and shall include the following specific locations along the flow channel:

a. Past the lower inlet plenum support structures and turning vanes;

b. Between ice baskets;

c. Past lattice frames;

d. Through the intermediate floor grating; and

e. Through the top deck floor grating.

The allowable [0.38]-inch thick buildup of frost or ice is based on analysis of containment response to a DBA with partial blockage of the ice condenser flow passages. If a flow channel in a given bay is found to have an accumulation of frost or ice greater than [0.38] inches thick, a representative sample of 20 additional flow channels from the same bay shall be visually inspected.

If these additional flow channels are all found to be acceptable, the discrepant flow channel may be considered single, unique, and acceptable deficiency. More than one discrepant flow channel in a bay is not acceptable, however. These requirements are based on the sensitivity of the

(continued)

BASES (continued)

partial blockage analysis to additional blockage. The Frequency of 9 months was based on ice storage tests and the allowance built into the required ice mass over and above the mass assumed in the safety analyses.

SR 3.6.~~15~~11.5

Verifying the chemical composition of the stored ice ensures that the stored ice has a boron concentration of [at least 1800] ppm as sodium tetraborate and a high pH, ≥ [9.0] and ≤ [9.5], in order to meet the requirement for borated water when the melted ice is used in the ECCS recirculation mode of operation. Sodium tetraborate has been proven effective in maintaining the boron content for long storage periods, and it also enhances the ability of the solution to remove and retain fission product iodine. The high pH is required to enhance the effectiveness of the ice and the melted ice in removing iodine from the containment atmosphere. This pH range also minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to ECCS and Containment Spray System fluids in the recirculation mode of operation. The Frequency of 18 months was developed considering these facts:

a. Long-term ice storage tests have determined that the chemical composition of the stored ice is extremely stable;

b. Operating experience has demonstrated that meeting the boron concentration and pH requirements has never been a problem; and

c. Someone would have to enter the containment to take the sample, and if the unit is at power, that person would receive a radiation dose.

SR 3.6.~~15~~11.6

This SR ensures that a representative sampling of ice baskets, which are relatively thin walled perforated cylinders, have not been degraded by wear, cracks, corrosion, or other damage. Each ice basket shall be raised at least ~~12~~ 10 feet for this inspection. However, for baskets where vertical lifting height is restricted due to overhead obstructions, a camera shall be used to perform the inspection. The ~~Surveillance~~ Frequency of 40 months for a visual inspection of the

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BASES (continued)

structural soundness of the ice baskets is based on engineering judgment, and considers such factors as the thickness of the basket walls relative to corrosion rates expected in their service environment and the results of the long term ice storage testing.

REFERENCES R

1. Watts Bar FSAR, Section [6.2], "Containment Systems."
2. Title 10, Code of Federal Regulations Part 50 ~~10 CFR 50,~~ Appendix K, "ECCS Evaluation Models."

(continued)

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3.6 CONTAINMENT SYSTEMS

3.6.~~16~~ 12 Ice Condenser Doors E ~~(Ice Condenser)~~

LCO 3.6.~~16~~ 12 The ice condenser inlet doors, intermediate deck doors, and top deck [doors] shall be OPERABLE and closed.

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

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each ice condenser door.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more ice condenser inlet doors inoperable due to being physically restrained from opening.	A.1	Restore inlet door(s) to OPERABLE status. W	1 hour
B. One or more ice condenser doors inoperable for reasons other than Condition A or not closed.	B.1	Verify maximum ice bed temperature ≤ [27]°F.	Once per 4 hours
	<u>AND</u>		
	B.2	W Restore ice condenser door(s) to OPERABLE status and closed positions.	14 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition B not met.	C.1	Restore W ice condenser door~~s~~ to OPERABLE status and closed positions.	48 hours
D. Required Action and associated Completion Time of Condition A or C not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.~~16~~ 12.1	Verify all inlet doors indicate closed by the Inlet Door Position Monitoring System.	12 hours
SR 3.6.~~16~~ 12.2	Verify, by visual inspection, each intermediate deck door is closed and not impaired by ice, frost, or debris.	7 days
SR 3.6.~~16~~ 12.3	Verify, by visual inspection, each inlet door is not impaired by ice, frost, or debris.	~~[3 months during first year after receipt of license AND]~~ [18] months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.~~16~~.4 12	Verify ~~Demonstrate~~ torque required to cause each inlet door to begin to open is ≤ [675] ~~inch~~-lbs. W	[3 months during first year after receipt of license AND] [18] months
SR 3.6.~~16~~.5 12	Perform a torque test on [a sampling of ≥ ~~25%~~ 50% of the] inlet doors. P	[3 months during first year after receipt of license AND] [18] months
SR 3.6.~~16~~.6 12	Verify for each intermediate deck door: a. No visual evidence of structural deterioration; b. Free movement of the vent assemblies; and c. Free movement of the door.	[3 months during first year after receipt of license AND] [18] months
SR 3.6.~~16~~.7 12	Verify, by visual inspection, each top deck [door]: a. Is in place; and c ~~b~~. Has no condensation, frost, or ice formed on the [doors] that would restrict their opening. b Free movement of the top deck vent assembly; and 1	92 days

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.~~16~~ 12 Ice Condenser Doors ~~(Ice Condenser)~~ E

BASES

BACKGROUND

The ice condenser doors consist of the inlet doors, the intermediate deck doors, and the top deck doors. The functions of the doors are to:

a. Seal the ice condenser from air leakage during the lifetime of the unit; and

b. Open in the event of a Design Basis Accident (DBA) to direct the hot steam-air mixture from the DBA into the ice bed, where the ice would absorb energy and limit Containment peak pressure and temperature during the accident transient.

Limiting the pressure and temperature following a DBA reduces the release of fission product radioactivity from containment to the environment.

The ice condenser is an annular compartment enclosing approximately 300° of the perimeter of the upper containment compartment, but penetrating the operating deck so a portion extends into the lower containment compartment. The inlet doors separate the atmosphere of the lower compartment from the ice bed inside the ice condenser. The top deck doors are above the ice bed and exposed to the atmosphere of the upper compartment. The intermediate deck doors, located below the top deck doors, form the floor of a plenum at the upper part of the ice condenser. This plenum area is used to facilitate surveillance and maintenance of the ice bed.

The ice baskets held in the ice bed within the ice condenser are arranged to promote heat transfer from steam to ice. This arrangement enhances the ice condenser's primary function of condensing steam and absorbing heat energy released to the containment during a DBA.

In the event of a DBA, the ice condenser inlet doors (located below the operating deck) open due to the pressure rise in the lower compartment. This allows air and steam to flow from the lower compartment into the ice condenser. The resulting pressure increase within the ice condenser causes the intermediate deck doors and the top deck doors to open,

(continued)

BASES (continued)

which allows the air to flow out of the ice condenser into the upper compartment. Steam condensation within the ice condensers limits the pressure and temperature buildup in containment. A divider barrier separates the upper and lower compartments and ensures that the steam is directed into the ice condenser.

The ice, together with the containment spray, serves as a containment heat-removal system and is adequate to absorb the initial blowdown of steam and water from a DBA as well as the additional heat loads that would enter containment during the several hours following the initial blowdown. The additional heat loads would come from the residual heat in the reactor core, the hot piping and components, and the secondary system, including the steam generators. During the post blowdown period, the Air Return System (ARS) returns upper compartment air through the divider barrier to the lower compartment. This serves to equalize pressures in containment, and to continue circulating heated air and steam from the lower compartment through the ice condenser, where the heat is removed by the remaining ice.

The water from the melted ice drains into the lower compartment where it serves as a source of borated water (via the containment sump) for the Emergency Core Cooling System (ECCS) and the Containment Spray System heat removal functions in the recirculation mode. The ice (via the Containment Spray System) and the recirculated ice melt also serve to clean up the containment atmosphere.

The ice condenser doors ~~assure~~ ensure Δ that the ice stored in the ice bed is preserved during normal operation (doors closed), and that the ice condenser functions as designed if called upon to act as a passive heat sink following a DBA.

APPLICABLE SAFETY ANALYSES

The limiting DBAs considered relative to containment pressure and temperature are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. DBAs are assumed not to occur simultaneously or consecutively.

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BASES (continued)

Although the ice condenser is a passive system that requires no electrical power to perform its function, the Containment Spray System and ARS also function to assist the ice bed in limiting pressures and temperatures. Therefore, the postulated DBAs are analyzed with respect to Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active failure and results in one train each of the Containment Spray System and the ARS being rendered inoperable.

The limiting DBA analyses (Ref. 1) show that the maximum peak containment pressure results from the LOCA analysis and is calculated to be less than the containment design pressure. For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the ECCS's cooling effectiveness during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures, in accordance with 10 CFR 50, Appendix K (Ref. 2).

The maximum peak containment atmosphere temperature results from the SLB analysis and is discussed in the Bases ~~B~~ for specification 3.6.5B, "Containment Air Temperature." Δ

An additional design requirement was imposed on the ice condenser door design for a small break accident in which the flow of heated air and steam is not sufficient to fully open the doors.

For this situation, the doors are designed so that all of the doors would partially open by approximately the same amount. Thus, the partially opened doors would modulate the flow so that each ice bay would receive an approximately equal fraction of the total flow.

This design feature ~~assures~~ ensures Δ that the heated air and steam will not flow preferentially to some ice bays and deplete the ice there without utilizing the ice in the other bays.

In addition to calculating the overall peak containment pressures, the DBA analyses include the calculation of the transient differential pressures that would occur across

(continued)

BASES (continued)

subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand the local transient pressure differentials for the limiting DBAs.

The ice condenser doors satisfy Criterion 3 of the NRC Policy Statement.

LCO

This LCO establishes the minimum equipment requirements to assure that the ice condenser doors perform their safety function. The ice condenser inlet doors, intermediate deck doors, and top deck doors must be closed to minimize air leakage into and out of the ice condenser, with its attendant leakage of heat into the ice condenser and loss of ice through melting and sublimation. The doors must be OPERABLE to ensure the proper opening of the ice condenser in the event of a DBA. OPERABILITY includes being free of any obstructions that would limit their opening, and for the inlet doors, being adjusted such that the opening and closing torques are within limits. The ice condenser doors function with the ice condenser to limit the pressure and temperature that could be expected following a DBA.

APPLICABILITY

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

The probability and consequences of these events in MODES 5 and 6 are reduced due to the pressure and temperature limitation of these MODES. Therefore, the ice condenser doors are not required to be OPERABLE in these MODES.

ACTIONS

Δ

A Note ~~has been~~ ~~added~~ ~~to~~ provides clarification that for this LCO, separate Condition entry is allowed for each ice condenser door.

(continued)

BASES (continued)

A.1

If one or more inlet doors are physically restrained from opening, the door(s) must be restored to OPERABLE status within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1-hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires containment to be restored to OPERABLE status in 1 hour.

B.1 and B.2

If one or more doors are determined to be partially open or otherwise inoperable for reasons other than Condition A, or a door is found that is not closed, it is acceptable to continue unit operation for up to 14 days provided the ice bed temperature instrumentation is monitored once per 4 hours to ensure that the open or inoperable door is not allowing enough air leakage to cause the maximum ice bed temperature to approach the melting point. The Frequency of 4 hours is based on the fact that temperature changes ~~cannot~~ occur slowly ~~rapidly~~ in the ice bed because of the large mass of ice involved. The 14-day Completion Time is based on long-term ice storage tests that indicate that if the temperature is maintained below [27]°F, there would not be a significant loss of ice from sublimation. If the maximum ice bed temperature is > [27]°F at any time, the situation reverts to Condition C and a Completion Time of 48 hours is allowed to restore the inoperable door to OPERABLE status or enter into Required Actions D.1 and D.2. Ice bed temperature must be verified to be within the specified Frequency as augmented by the provisions of SR 3.0.2. If this verification is not made, Required Actions D.1 and ~~Required Action~~ D.2, not Required Action C.1, must be taken.

2

C.1

2

If ~~the~~ Required Actions B.1 or B.2 are not met, the doors 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~~ highly improbable for the temperature to decrease to the melting point and a significant amount of ice to melt in a 48 hour period. Condition C is entered from Condition B only when the

(continued)

BASES (continued)

Completion Time of Required Action B.2 is not met or the ice ~~B~~bed temperature has not been verified at the required Frequency.

<u>D.1 and D.2</u>

Δ

If the ice condenser doors required cannot be restored to ~~within~~ OPERABLE sta ~~limits~~ within the ~~associated~~ Completion Time, the unit must be brought to a ~~placed in~~ MODE in which the LCO does not apply. ~~This is done by placing the unit in~~ To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to ~~in~~ MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required ~~MODES~~ plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

<u>SR 3.6.~~16~~12.1</u>

Verifying, by means of the Inlet Door Position Monitoring System, that the inlet doors are in their closed positions makes the operator aware of an inadvertent opening of one or more doors. The ~~Surveillance~~ Frequency of 12 hours ensures that operators on each shift are aware of the status of the doors.

<u>SR 3.6.~~16~~12.2</u>

Verifying by visual inspection that each intermediate deck door is closed and not impaired by ice, frost, or debris provides assurance that the intermediate deck doors (which form the floor of the upper plenum where frequent maintenance on the ice bed is performed) have not been left open or obstructed. The Frequency of 7 days is based on engineering judgment and takes into consideration such factors as the frequency of entry into the intermediate ice condenser deck, the time required for significant frost buildup, and the probability that a DBA will occur.

<u>SR 3.6.~~16~~12.3</u>

Verifying by visual inspection that the ice condenser inlet doors are not impaired by ice, frost, or debris provides

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BASES (continued)

assurance that the doors are free to open in the event of a DBA. For this facility, the Frequency of [18] months ([3 months during the first year after receipt of license]) is based on door design, which ~~does not allow water~~ inhibits condensation to freeze, and operating experience, which indicates that the inlet doors ~~very rarely fail to~~ usually meet their SR acceptance criteria. Because of high radiation in the vicinity of the inlet doors during power operation, this Surveillance is normally performed during a shutdown. (2)

SR 3.6.~~16~~12.4

Verifying the opening torque of the inlet doors provides assurance that no doors have become stuck in the closed position. The value of [675] inch-lb is based on the design opening pressure on the doors of 1.0 lb/ft^2. For this facility, the Frequency of [18] months ([3 months during the first year after receipt of license]) is based on the passive nature of the closing mechanism (i.e., once adjusted, there are no known factors that would change the setting, except possibly a buildup of ice; ice buildup is not likely, however, because of the door design, which ~~does not allow water~~ inhibits condensation to freeze). Operating experience indicates that the inlet doors ~~very rarely fail to~~ usually meet their SR acceptance criteria. Because of high radiation in the vicinity of the inlet doors during power operation, this Surveillance is normally performed during a shutdown. Δ (2)

SR 3.6.~~16~~12.5

The torque test Surveillance ensures that the inlet doors have not developed excessive friction and that the return springs are producing a door return torque within limits. The torque test consists of the following:

1. Verify that the torque, T(OPEN), required to cause opening motion at the [40]° open position, is ≤ [195] inch-lb;

2. Verify that the torque, T(CLOSE), required to hold the door stationary (i.e., keep it from closing) at the [40]° open position, is ≥ [78] inch-lb; and

(continued)

BASES (continued)

3. Calculate the frictional torque, T(FRICT) = 0.5 {T(OPEN) - T(CLOSE)}, and verify that the T(FRICT) is ≤ [40] inch-lb.

The purpose of the friction and return torque specifications is to ensure that, in the event of a small break LOCA or SLB, all of the 24 door pairs open uniformly. This assures that, during the initial blowdown phase, the steam and water mixture entering the lower compartment does not pass through part of the ice condenser, depleting the ice there, while bypassing the ice in other bays. ~~For this facility,~~ the Frequency of [18] months ([3 months during the first year after receipt of license]) is based on the passive nature of the closing mechanism (i.e., once adjusted, there are no known factors that would change the setting, except possibly a buildup of ice; ice buildup is not likely, however, because of the door design, which ~~does not allow water~~ inhibits condensation to freeze). Operating experience indicates that the inlet doors very rarely fail to meet their SR acceptance criteria. Because of high radiation in the vicinity of the inlet doors during power operation, this Surveillance is normally performed during a shutdown. Δ 2

SR 3.6.~~16~~ 12.6

Verifying the OPERABILITY of the intermediate deck doors provides assurance that the intermediate deck doors are free to open in the event of a DBA. The verification consists of visually inspecting the intermediate doors for structural deterioration, verifying free movement of the vent assemblies, and ascertaining free movement of each door when lifted with the applicable force shown below:

	Door	Lifting Force
a.	Adjacent to crane wall	< 37.4 lb
b.	Paired with door adjacent to crane wall	≤ 33.8 lb
c.	Adjacent to containment wall	≤ 31.8 lb
d.	Paired with door adjacent to containment wall	≤ 31.0 lb

The 18-month Frequency ([3 months during the first year after receipt of license]) is based on the passive design of the intermediate deck doors, the frequency of personnel entry into the intermediate deck, and the fact that SR 3.6.~~17~~ 12.2

(continued)

BASES (continued)

confirms on a ~~7~~ 9 day Frequency that the doors are not impaired by ice, frost, or debris, which are ways a door would fail the opening force test (i.e., by sticking or from increased door weight).

SR 3.6.~~16~~ 12.7

Verifying by visual inspection that the top deck doors are in place and not obstructed And verifying free movement of the vent Assembly provides assurance that the doors are performing their function of keeping warm air out of the ice condenser during normal operation, and would not be obstructed if called upon to open in response to a DBA. The Frequency of 92 days is based on engineering judgment, which considered such factors as the following:

a. The relative inaccessibility and lack of traffic in the vicinity of the doors make it unlikely that a door would be inadvertently left open;

b. Excessive air leakage would be detected by temperature monitoring in the ice condenser; and

c. The light construction of the doors would ~~assure~~ ensure that, in the event of a DBA, air and gases passing through the ice condenser would find a flow path, even if a door were obstructed.

REFERENCES (R)

1. Watts Bar FSAR, Section [15], "Accident Analysis."
2. Title 10, Code of Federal Regulations, Part 50, ~~10 CFR 50~~ Appendix K, "ECCS Evaluation Models."

(continued)

DRAFT

3.6 CONTAINMENT SYSTEMS

3.6.~~17~~ 13 Divider Barrier Integrity E ~~(Ice Condenser)~~

LCO 3.6.~~17~~ 13 Divider barrier integrity shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ---------NOTE--------- For this action, separate condition entry is allowed for each personnel access door or equipment hatch. --------------------- One or more personnel access doors or equipment hatches open or inoperable, other than for personnel transit entry.	A.1	Restore personnel access doors and equipment hatches to OPERABLE status and closed positions.	1 hour
B. Divider barrier seal inoperable.	B.1	Restore seal to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.~~17~~.1 13	Verify, by visual inspection, all personnel access doors and equipment hatches between upper and lower containment compartments closed.	Prior to entering MODE 4 from MODE 5
SR 3.6.~~17~~.2 13	Verify, by visual inspection, that the seals and sealing surfaces of each personnel access door and equipment hatch have: a. No detrimental misalignments; b. No cracks or defects in the sealing surfaces; and c. No apparent deterioration of the seal material.	Prior to final closure after each opening AND -----NOTE------ Only required for seals made of resilient materials --------------- 10 years
SR 3.6.~~17~~.3 13	Verify, by visual inspection, each personnel access door or equipment hatch that has been opened for personnel transit entry is closed.	After each opening
SR 3.6.~~17~~.4 13	Remove ~~2~~ two divider barrier seal test coupons and verify: E Both test coupons' tensile strength ≥ ~~[120]~~ 60 psi. P ~~[and]~~ ~~b. Both test coupons' elongation ≥ [100]%.~~ N/A	[18] months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.~~17~~13.5	Visually inspect ≥ [95]99 % of the divider barrier seal length, and verify: a. Seal and seal mounting bolts are properly installed; and b. Seal material shows no evidence of deterioration due to holes, ruptures, chemical attack, abrasion, radiation damage, or changes in physical appearance.	[18]99 months

DRAFT

B 3.6 CONTAINMENT SYSTEMS

B 3.6.~~17~~ 13 Divider Barrier Integrity ~~(Ice Condenser)~~ Ⓔ

BASES

BACKGROUND

The divider barrier consists of the operating deck and associated seals, personnel access doors, and equipment hatches that separate the upper and lower containment compartments. Divider barrier integrity is necessary to minimize bypassing of the ice condenser by the hot steam and air mixture released into the lower compartment during a Design Basis Accident (DBA). This ensures that most of the gases pass through the ice ~~B~~ed, which condenses the steam and limits pressure and temperature during the accident transient. Limiting the pressure and temperature reduces the release of fission product radioactivity from containment to the environment in the event of a DBA.

In the event of a DBA, the ice condenser inlet doors (located below the operating deck) open due to the pressure rise in the lower compartment. This allows air and steam to flow from the lower compartment into the ice condenser. The resulting pressure increase within the ice condenser causes the intermediate deck doors and the door panels at the top of the condenser to open, which allows the air to flow out of the ice condenser into the upper compartment. The ice condenses the steam as it enters, thus limiting the pressure and temperature buildup in containment. The divider barrier separates the upper and lower compartments and ensures that the steam is directed into the ice condenser. The ice, together with the containment spray, is adequate to absorb the initial blowdown of steam and water from a DBA as well as the additional heat loads that would enter containment over several hours following the initial blowdown. The additional heat loads would come from the residual heat in the reactor core, the hot piping and components, and the secondary system, including the steam generators. During the post blowdown period, the Air Return System (ARS) returns upper compartment air through the divider barrier to the lower compartment. This serves to equalize pressures in containment and to continue circulating heated air and steam from the lower compartment through the ice condenser, where the heat is removed by the remaining ice.

(continued)

BASES (continued)

Divider barrier integrity ensures that the high energy fluids released during a DBA would be directed through the ice condenser, and that the ice condenser would function as designed if called upon to act as a passive heat sink following a DBA.

APPLICABLE SAFETY ANALYSES

Divider barrier integrity ensures the functioning of the ice condenser to the limiting containment pressure and temperature that could be experienced following a DBA. The limiting DBAs considered relative to containment temperature and pressure are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. DBAs are assumed not to occur simultaneously or consecutively.

Although the ice condenser is a passive system that requires no electrical power to perform its function, the Containment Spray System and the ARS also function to assist the ice ~~bed~~ in limiting pressures and temperatures. Therefore, the postulated DBAs are analyzed, with respect to containment Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active failure and results in the inoperability of one train in both the Containment Spray System and the ARS.

The limiting DBA analyses (Ref. 1) show that the maximum peak containment pressure results from the LOCA analysis and is calculated to be less than the containment design pressure. The maximum peak containment temperature results from the SLB analysis and is discussed in the Bases ~~B~~ for specification 3.6.5B, "Containment Air Temperature."

In addition to calculating the overall peak containment pressures, the DBA analyses include calculation of the transient differential pressures that occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand these local transient pressure differentials for the limiting DBAs.

The divider barrier satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

This LCO establishes the minimum equipment requirements to ensure that the divider barrier performs its safety function of ensuring that bypass leakage, in the event of a DBA, does not exceed the bypass leakage assumed in the accident analysis. Included are the requirements that the personnel access doors and equipment hatches in the divider barrier are OPERABLE and closed, and that the divider barrier seal is properly installed and has not degraded with time. An exception to the requirement that the doors be closed is made to allow personnel transit entry through the divider barrier. The basis of this exception is the assumption that, for personnel transit, the time during which a door is open will be short (i.e., shorter than the Completion Time of 1 hour for Condition A). The divider barrier functions with the ice condenser to limit the pressure and temperature that could be expected following a DBA.

APPLICABILITY

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

The probability and consequences of these events in MODES 5 and 6 are low due to the pressure and temperature limitations of these MODES. As such, divider barrier integrity is not required in these MODES.

ACTIONS

A.1

If one or more personnel access doors or equipment hatches is inoperable or open, except for personnel transit entry, 1 hour is allowed to restore the door(s) to OPERABLE status and the closed position. The 1 hour Completion Time is consistent with LCO 3.6.1 "Containment" which requires that containment be restored to OPERABLE status within 1 hour.

Δ Condition A ~~Required Action A.1~~ has been modified by a Note to provide clarification that for this LCO, separate Condition entry is allowed for each personnel access door and equipment hatch.

(continued)

BASES (continued)

B.1

If the divider barrier seal is inoperable, 1 hour is allowed to restore the seal to OPERABLE status. The 1 hour Completion Time is consistent with LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

C.1 and C.2

If the Divider Barrier integrity cannot be restored to OPERABLE status ~~If the Required Actions are not met~~ within the required Completion Time, the plant ~~unit~~ must be brought to ~~placed in~~ a MODE ~~where the requirement~~ in which the LCO does not apply. ~~This is performed by placing the unit in~~ To achieve this status, the plant must be brought to MODE 3 within 6 hours and to ~~in~~ MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required ~~MODES~~ plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(2) SURVEILLANCE REQUIREMENTS

SR 3.6.~~17~~13.1

Verification by visual inspection that all personnel access doors and equipment hatches between the upper and lower containment compartments are closed provides assurance that divider barrier integrity is maintained prior to the reactor being taken from MODE 5 to MODE 4. The visual inspection shall include the canal gate and control rod drive missile shield which penetrate the divider barrier. This SR is necessary because many of the doors and hatches may have been opened for maintenance during the shutdown.

SR 3.6.~~17~~13.2

Verification that the personnel access door and equipment-hatch seals, sealing surfaces, and alignments are acceptable provides assurance that divider barrier integrity is maintained. This inspection cannot be made when the door or hatch is closed. Therefore, SR 3.6.17.2 is required for each door or hatch that has been opened, prior to the final closure. Some doors and hatches may not be opened for long periods of time. Those that use resilient materials in the seals must be opened and inspected at least once every

(continued)

BASES (continued)

10 years to provide assurance that the seal material has not aged to the point of degraded performance. The Frequency of 10 years is based on the known resiliency of the materials used for seals, the fact that the openings have not been opened (to cause wear), and operating experience that confirms that the seals inspected at this Frequency have been found to be acceptable.

SR 3.6.~~17~~13.3

Verification after each opening of a personnel access door or equipment hatch that it has been closed makes the operator aware of the importance for closing it and thereby provides additional assurance that divider barrier integrity is maintained while in applicable MODES.

SR 3.6.~~17~~13.4

Conducting periodic physical property tests on divider barrier seal test coupons provides assurance that the seal material has not degraded in the containment environment, including the effects of irradiation with the reactor at power. The required test~~s~~ include~~s~~ a tensile strength test. ~~and a test for elongation.~~ (1) The Frequency of 18 months was developed considering such factors as the known resiliency of the seal material used, the inaccessibility of the seals and absence of traffic in their vicinity, and the unit conditions needed to perform the SR. Operating experience has shown that these components usually ~~pass the Surveillance test~~ pass the SR ~~tes in 3.6.7.4~~ when performed on the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Δ

SR 3.6.~~17~~13.5

Visual inspection of the seal around the perimeter provides assurance that the seal is properly secured in place. The Frequency of 18 months was developed considering such factors as the inaccessibility of the seals and absence of traffic in their vicinity, the strength of the bolts and mechanisms used to secure the seal, and the unit conditions needed to perform the SR. Operating experience has shown

(continued)

BASES (continued)

that these components usually pass the Surveillance test when performed on the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES R 1. Watts Bar FSAR, Section 6.2, "Containment Systems."

(continued)

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3.6 CONTAINMENT SYSTEMS

3.6.~~18~~ 14 Containment Recirculation Drains E ~~(Ice Condenser)~~

LCO 3.6.~~18~~ 14 The ice condenser floor drains and the refueling canal drains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ice condenser floor drain inoperable.	A.1 Restore ice condenser floor drain to OPERABLE status.	1 hour
B. One refueling canal drain inoperable.	B.1 Restore refueling canal drain to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE REQUIREMENTS	FREQUENCY
SR 3.6.~~18~~ 14.1	Verify by visual inspection that: a. Each refueling canal drain plug is removed; b. Each refueling canal drain is not obstructed by debris; and c. No debris is present in the upper compartment or refueling canal that could obstruct the refueling canal drain.	92 days AND Prior to entering MODE 4 from MODE 5 after each partial or complete fill of the canal
SR 3.6.~~18~~ 14.2	Verify for each ice condenser floor drain that the: a. Valve opening is not impaired by ice, frost, or debris; b. Valve seat shows no evidence of damage; c. Valve opening force is ≤ [~~66~~] 100 lb; and P d. Drain line from the ice condenser floor to the lower compartment is unrestricted.	[18] months

DRAFT

B 3.6 CONTAINMENT SYSTEMS

B 3.6.~~18~~ 14 Containment Recirculation Drains ~~(Ice Condenser)~~ (E)

BASES

BACKGROUND

The containment recirculation drains consist of the ice condenser drains and the refueling canal drains. The ice condenser is partitioned into 24 bays, each having a pair of inlet doors that open from the bottom plenum to allow the hot steam/air mixture from a Design Basis Accident (DBA) to enter the ice condenser. Twenty of the 24 bays have an ice condenser floor drain at the bottom to drain the melted ice into the lower compartment (in the 4 bays that do not have drains, the water drains through the floor drains in the adjacent bays). Each drain leads to a drain pipe that drops down several feet, then makes one or more 90° bends and exits into the lower compartment. A check (flapper) valve at the end of each pipe keeps warm air from entering during normal operation, but when the water exerts pressure, it opens to allow the water to spill into the lower compartment. This prevents water from backing up and interfering with the ice condenser inlet doors. The water delivered to the lower containment serves to cool the atmosphere as it falls through to the floor and provides a source of borated water at the containment sump for long term use by the Emergency Core Cooling System (ECCS) and the containment Spray System during the recirculation MODE of operation.

The two refueling canal drains are at low points in the refueling canal. During a refueling, plugs are installed in the drains and the canal is flooded to facilitate the refueling process. The water acts to shield and cool the spent fuel as it is transferred from the reactor vessel to storage. After refueling, the canal is drained and the plugs removed. In the event of a DBA, the refueling canal drains are the main return path to the lower compartment for Containment Spray System water sprayed into the upper compartment.

The ice condenser drains and the refueling canal drains functional with the ice bed, the Containment Spray System, and the ECCS, to limit the pressure and temperature that could be expected following a DBA.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The limiting DBAs considered relative to containment temperature and pressure are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. DBAs are assumed not to occur simultaneously or consecutively. Although the ice condenser is a passive system that requires no electrical power to perform its function, the Containment Spray System and the Air Return System (ARS) also function to assist the ice bed in limiting pressures and temperatures. Therefore, the analysis of the postulated DBAs, with respect to Engineered Safety Feature (ESF) systems, assumes the loss of one ESF bus, which is the worst case single active failure and results in one train of the Containment Spray System and one train of the ARS being rendered inoperable.

A the for specification

The limiting DBA analyses (Ref. 1) show that the maximum peak containment pressure results from the LOCA analysis and is calculated to be less than the containment design pressure. The maximum peak containment atmosphere temperature results from the SLB analysis and is discussed in Bases ~~B~~ 3.6.5, "Containment Air Temperature." In addition to calculating the overall peak containment pressures, the DBA analyses include calculation of the transient differential pressures that occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand these local transient pressure differentials for the limiting DBAs.

The containment recirculation drains satisfy Criterion 3 of the NRC Policy Statement.

LCO

This LCO establishes the minimum requirements to ensure that the containment recirculation drains perform their safety functions. The ice condenser floor drain valve disks must be closed to minimize air leakage into and out of the ice condenser during normal operation and must open in the event of a DBA when water begins to drain out. The refueling canal drains must have their plugs removed and remain clear

(continued)

BASES (continued)

to ensure the return of Containment Spray System water to the lower containment in the event of a DBA. The containment recirculation drains function with the ice condenser, ECCS, and Containment Spray System to limit the pressure and temperature that could be expected following a DBA.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature, which would require the operation of the containment recirculation drains. Therefore, the LCO is applicable in MODES 1, 2, 3, and 4.

The probability and consequences of these events in MODES 5 and 6 are low due to the pressure and temperature limitations of these MODES. As such, the containment recirculation drains are not required to be OPERABLE in these MODES.

ACTIONS

A.1

If one ice condenser floor drain is inoperable, 1 hour is allowed to restore the drain to OPERABLE status. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1-hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which require that containment be restored to OPERABLE status within 1 hour.

B.1

If one refueling canal drain is inoperable, 1 hour is allowed to restore the drain to OPERABLE status. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1-hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which require that containment be restored to OPERABLE status in 1 hour.

(continued)

BASES (continued)

C.1 and C.2

If the affected drain(s) cannot be restored to OPERABLE status within the required ~~in the associated~~ Completion Time, the unit must be brought to ~~placed in~~ a MODE in which the LCO does not apply. ~~This is done by placing the unit in~~ To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to ~~in~~ MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required ~~MODES~~ plant conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.~~18~~ 14.1

Verifying the OPERABILITY of the refueling canal drains ensures that they will be able to perform their functions in the event of a DBA. This Surveillance confirms ~~extra~~ that the refueling canal drain plugs have been removed and that the drains are clear of any obstructions that could impair their functioning. In addition to debris near the drains, attention must be given to any debris that is located where it could be moved to the drains in the event that the Containment Spray System is in operation and water is flowing to the drains. SR 3.6.18.1 must be performed before entering MODE 4 from MODE 5 after every filling of the canal to ensure that the plugs have been removed and that no debris that could impair the drains was deposited during the time the canal was filled. The 92-day Frequency was developed considering such factors as the inaccessibility of the drains, the absence of traffic in the vicinity of the drains, and the redundancy of the drains.

SR 3.6.~~18~~ 14.2

Verifying the OPERABILITY of the ice condenser floor drains ensures that they will be able to perform their functions in the event of a DBA. Inspecting the drain valve disk ensures that the valve is performing its function of sealing the drain line from warm air leakage into the ice condenser during normal operation, yet will open if melted ice fills the line following a DBA. Verifying that the drain lines are not obstructed ensures their readiness to drain water from the ice condenser. The 18-month Frequency was

(continued)

BASES (continued)

developed considering such factors as the inaccessibility of the drains during power operation; the design of the ice condenser, which precludes melting and refreezing of the ice; and operating experience that has confirmed that the drains are found to be acceptable when the Surveillance is performed on an 18 month Frequency. Because of high radiation in the vicinity of the drains during power operation, this Surveillance is normally done during a shutdown.

REFERENCES (R) 1. Watts Bar FSAR, Section [6.2] "Containment Systems."

(continued)

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3.6 CONTAINMENT SYSTEMS

3.6.~~19~~ 15 Shield Building ~~(Dual and Ice Condenser)~~ (E)

LCO 3.6.~~19~~ 15 The shield building shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Shield building inoperable.	A.1 Restore shield building to OPERABLE status.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
~~[SR 3.6.19.1~~	~~Verify annulus negative pressure > [5] inches water gauge.~~	~~12 hours]~~

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.~~19.2~~ 15.1 E 2	Verify the ~~each~~ door in each access opening is closed except when the access opening is being used for normal transient entry and exit~~s~~. ~~then, at least 1 door shall be closed.~~	31 days
SR 3.6.~~19.3~~ 15.2 E	Verify shield building structural integrity by performing a visual inspection of the exposed interior and exterior surfaces of the shield building.	During shutdown for SR 3.6.1.1 Type A tests

B 3.6 CONTAINMENT SYSTEMS

B 3.6.~~19~~ 15 Shield Building ~~(Dual and Ice Condenser)~~ E

BASES

BACKGROUND

The shield building is a concrete structure that surrounds the steel containment vessel. Between the containment vessel and the shield building inner wall is an annular space that collects containment leakage that may occur following a loss of coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the steel containment vessel.

The Shield Building Air Cleanup System (SBACS) establishes a negative pressure in the annulus between the shield building and the steel containment vessel. Filters in the system then control the release of radioactive contaminants to the environment. The shield building is required to be OPERABLE to ensure retention of containment leakage and proper operation of the SBACS.

APPLICABLE SAFETY ANALYSES

The design basis for shield building OPERABILITY is a LOCA. Maintaining shield building OPERABILITY ensures that the release of radioactive materials from the containment atmosphere is restricted to those leakage paths and associated leakage rates assumed in the accident analysis. This restriction, in conjunction with the operation of the SBACS, will limit the site boundary radiation doses to within limits.

The shield building satisfies Criterion 3 of the NRC Policy Statement.

LCO

Shield building OPERABILITY must be maintained to ensure proper operation of the SBACS and to limit radioactive leakage from the containment to those paths and leakage rates assumed in the accident analysis.

APPLICABILITY

Maintaining shield building OPERABILITY prevents leakage of radioactive material from the shield building. Radioactive material may enter the shield building from the containment

(continued)

BASES (continued)

following a LOCA. Therefore, shield building OPERABILITY is required in MODES 1, 2, 3, and 4 when a steam line break, LOCA, or rod ejection accident could release radioactive material to the containment atmosphere.

ACTIONS

A.1

In the event shield building OPERABILITY is not maintained, shield building OPERABILITY must be restored within 24 hours.

Twenty-four hours is a reasonable Completion Time considering the limited leakage design of containment and the low probability of a DBA occurring during this time period.

B.1 and B.2

If shield building ~~OPERABILITY~~ cannot be restored to OPERABLE status within ~~in~~ the required Completion Time ~~time period~~, the ~~unit~~ plant must be ~~placed in~~ brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to ~~This is done by placing the unit in~~ at least MODE 3 within 6 hours and to ~~in~~ MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required ~~MODES~~ plant conditions from full power in an orderly manner and without challenging ~~unit~~ plant systems.

SURVEILLANCE REQUIREMENTS 1

~~SR 3.6.19.1~~

~~Verifying that shield building annulus pressure is within limits ensures that operation remains within the limits assumed in the containment analysis. The 24 hour Frequency of this SR was developed considering operating experience related to shield building annulus pressure variations and pressure instrument drift during the applicable MODES.~~

SR 3.6.~~19.2~~ 15.1

Maintaining shield building OPERABILITY requires maintaining each door in the access opening closed, except when the

(continued)

BASES (continued)

2 access opening is being used for normal transient entry and exit ~~(then at least one door must remain closed)~~. The 31-day Frequency of this SR is based on engineering judgment and is considered adequate in view of the other indications of door status that are available to the operator.

SR 3.6.~~19~~15.~~3~~2 E

This SR would give advance indication of gross deterioration of the concrete structural integrity of the shield building. The Frequency of this SR is the same as that of SR 3.6.1.1. The verification is done during shutdown and as part of Type A leakage tests associated with SR 3.6.1.1.

REFERENCES None.

~~(continued)~~

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 The MSSVs shall be OPERABLE as specified in Table 3.7.1-1 and Table 3.7.1-2.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required MSSV(s) inoperable. A	A.1	Verify by administrative means at least [two] MSSVs per steam generator are OPERABLE.	4 hours
	AND		
	A.2 1	Reduce power to less than or equal to the applicable % RTP listed in Table 3.7.1-1.	4 hours 1
B. Required Action and associated Completion Time not met. OR One or more SGs with less than [two] MSSVs OPERABLE A	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1 A	-------------------NOTE------------------- Only required to be performed ~~prior to entry into~~ in MODES 1 and 2. -- Verify each required MSSV lift setpoint per Table 3.7.1-2 in accordance with the Inservice Testing Program.	In accordance with the Inservice Testing Program

Table 3.7.1-1 (page 1 of 1)
OPERABLE Main Steam Safety Valves versus
Applicable Power in Percent of RATED THERMAL POWER

MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	APPLICABLE POWER (% RTP)
5	≤100 ≥ ~~[87]~~ 80
4	≤ 80 < ~~[87]~~ 80 , ≥ [65]
3	≤ 60 < ~~[65]~~ 60 , ≥ [43]
2	≤ 40 < ~~[43]~~ 40

1
A

Table 3.7.1-2 (page 1 of 1)
Main Steam Safety Valve Lift Settings

VALVE NUMBER				LIFT SETTING (psig ± [3]%)
STEAM GENERATOR				
#1	#2	[#3]	[#4]	
1-522	1-517	1-512	1-527	[1224]
1-523	1-518	1-513	1-528	[1215]
1-524	1-519	1-514	1-529	[1205]
1-525	1-520	1-515	1-530	[1195]
1-526	1-521	1-516	1-531	[1185]

1

4

5

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Five MSSVs are located on ~~each~~ main steam header, outside containment, upstream of the main steam isolation valves, as described in the FSAR, Section ~~[~~10.3.~~1]~~2 (Ref. 1). The MSSV capacity criteria is 105% ~~110%~~ of rated steam flow at 110% of the steam generator design pressure. This meets the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-2 in the accompanying LCO, so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine-reactor trip.

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2 and its purpose is to limit the secondary system pressure to ≤ 110% of design pressure when passing 100% of design steam flow. This design basis is sufficient to cope with any anticipated operating occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in the FSAR, Section ~~[15.2]~~ 15.2 and 15.4 (Ref. 3). Of these, the full power turbine trip without steam dump is the limiting AOO. This event also terminates normal feedwater flow to the steam generators.

The transient response for turbine trip without a direct reactor trip presents no hazard to the integrity of the RCS

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

or the Main Steam System. If a minimum reactivity feedback is assumed, the reactor is tripped on high pressurizer pressure. In this case, the pressurizer safety valves open, and RCS pressure remains below 110% of the design value. The MSSVs also open to limit the secondary steam pressure.

If maximum reactivity feedback is assumed, the reactor is tripped on overtemperature ΔT. The departure from nucleate boiling ratio increases throughout the transient, and never drops below its initial value. Pressurizer relief valves and MSSVs are activated and prevent overpressurization in the primary and secondary systems. The MSSVs are assumed to have two active and one passive failure modes. The active failure modes are spurious opening, and failure to reclose once opened. The passive failure mode is failure to open upon demand.

The MSSVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

The accident analysis requires four MSSVs per steam generator to provide overpressure protection for Design Basis transients occurring at 102% RTP. An MSSV will be considered inoperable if it fails to open on demand. The LCO requires that five MSSVs be OPERABLE in compliance with Reference 2, even though this is not a requirement of the DBA analysis. This is because operation with less than the full number of MSSVs requires limitations on allowable THERMAL POWER (to meet ASME Code requirements). These limitations are according to Table 3.7.1-1 in the accompanying LCO, and Required Action A.2.

The OPERABILITY of the MSSVs is defined as the ability to open within the setpoint tolerances, relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

The lift settings, according to Table 3.7.1-2 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

(continued)

BASES

LCO (continued)

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB.

APPLICABILITY

In MODE 1 above 40 ~~43%~~ % RTP, the number of MSSVs per steam generator required to be OPERABLE must be according to Table 3.7.1-1 in the accompanying LCO. Below 40 ~~43%~~ % RTP in MODE 1, 2, and 3, only two MSSVs per steam generator are required to be OPERABLE. 1

In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

The Actions table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1

A

With one or more MSSVs inoperable, ~~verify by administrative means that at least [two] required MSSVs per steam generator are OPERABLE. This action may be satisfied by examining logs or other information to determine whether the MSSVs are out of service for maintenance or other reasons. It does not mean that it is necessary to perform the SRs needed to demonstrate OPERABILITY of the MSSVs. The 4-hour Completion Time, which is the same as that for restoring an MSSV to OPERABLE status, is reasonable for examining information sources, such as maintenance logs, to determine if two MSSVs are OPERABLE. The Completion Time is based on the low probability of an event occurring during this time period that would require activation of the MSSVs.~~

~~A.2~~

~~An alternative to restoring the inoperable MSSV(s) to OPERABLE status is to~~ reduce power so that the available MSSV relieving capacity meets Reference 2 requirements for the applicable THERMAL POWER.

(continued)

BASES

A

ACTIONS

A.1 (continued)

Operation with less than all five MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is proportionally limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator. For example, if one MSSV is inoperable in one steam generator, the relief capacity of that steam generator is reduced by approximately 20%. To offset this reduction in relief capacity, energy transfer to that steam generator must be similarly reduced by at least 20%. This is accomplished by reducing THERMAL POWER by at least 20%, which conservatively limits the energy transfer to all steam generators to approximately 80% of total capacity, consistent with the relief capacity of the most limiting steam generator.

For each steam generator, at a specified pressure, the fractional relief capacity (FRC) of each MSSV is determined as follows:

$$FRC = \frac{A}{B}$$

where:

A = the relief capacity of the MSSV; and

B = the total relief capacity of all the MSSVs of the steam generator

The FRC is the relief capacity necessary to address operation with reduced THERMAL POWER.

The reduced THERMAL POWER levels in the LCO prevent operation at power levels greater than the relief capacity of the remaining MSSVs. The reduced THERMAL POWER is determined as follows:

$$RP = \left[1 - \left(N_1 \times FRC_1 + N_2 \times FRC_2 + \cdots + N_5 \times FRC_5\right)\right] \times 100\%$$

(continued)

BASES

A

ACTIONS

A.2 (continued)

1

where:

RP = Reduced THERMAL POWER for the most limiting steam generator expressed as a percent of RTP;

$N_1, N_2, \ldots, N_5$ represent the status of the MSSV 1, 2, ..., 5

= 0 if the MSSV is OPERABLE,
= 1 if the MSSV is inoperable;

$FRC_1, FRC_2, \ldots, FRC_5$ = the relief capacity of the MSSV 1, 2, ..., 5 as defined above.

A

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

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

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

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

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

The ANSI/ASME standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in-situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

INSERT

3

1. ~~FSAR, Section [10.3.1].~~

2. ~~American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, Section III, Article NC-7000, Class 2 Components.~~

3. ~~FSAR, Section [15.2].~~

4. ~~American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, Section XI, Article IWV-3500.~~

5. ~~ANSI/ASME OM-1-1987.~~

INSERT

1. Watts Bar FSAR, Section 10.3, "Main Steam Supply System."

2. American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, Section III, Article NC-7000, "Overpressure Protection," Class 2 Components.

3. Watts Bar FSAR, Section 15.2, Condition II - Faults of Moderate Frequency," and Section 15.4, "Condition IV - Limiting Faults."

4. American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, Section XI, Article IWV-3500, "Inservice Test: Category C Valves."

5. ANSI/ASME OM-1-1987, "Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices."

JUSTIFICATIONS FOR DEVIATIONS FROM STS 3.7.1

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

1. Change to correct error(s) in the STS.

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

3. Watts Bar prefers to use the expanded Reference information and format.

4. Change to add plant specific requested information.

5. Change to reflect Watts Bar specific parameter value(s).

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

LCO 3.7.2 [Four] MSIVs shall be OPERABLE.

A

APPLICABILITY: MODE 1,
MODES 2 and 3, except when all MSIVs are closed and deactivated with any MSIV open

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MSIV inoperable in MODE 1.	A.1 Restore MSIV to OPERABLE status.	[8] hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 2. AND B.2 Close inoperable MSIV.	6 hours 6 hours
C. ---------NOTE--------- Separate Condition entry is allowed for each MSIV. ---------------------- One or more MSIVs inoperable in MODE 2 or 3.	C.1 Close inoperable MSIV. AND C.2 Verify inoperable MSIV(s) is closed.	[8] hours Once per 7 days

A

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition B e or C not met. φ A	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	-------------------NOTE-------------------- Only required to be performed ~~prior to~~ in ~~entry into~~ MODE 1 and 2. --- Verify ~~MSIV~~ closure time of each MSIV is ≤ [4.6] 5.0 4 seconds on an actual or simulated actuation signal.	A In accordance with the [Inservice Testing Program, or 18 months]

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a high energy line break (HELB). MSIV closure terminates flow break from the unaffected (intact) steam generators. (A)

One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are downstream from the main steam safety valves (MSSVs) and auxiliary feedwater (AFW) pump turbine steam supply, to prevent MSSV and AFW isolation from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the others, and isolates the turbine, Steam ~~Bypass~~ Dump System, and other auxiliary steam supplies from the steam generators. (3)

The MSIVs close on a main steam line isolation signal generated by either low steam ~~generator~~ line pressure, [illegible]ive steam [illegible]ure rate (below P-11) or high-high containment pressure. The MSIVs fail closed on loss of control or actuation power. (3) (3)

Each MSIV has an MSIV bypass valve. Although these bypass valves are normally closed, they receive the same emergency closure signal as do their associated MSIVs. The MSIVs may also be actuated manually.

A description of the MSIVs is found in the FSAR, Section ~~[~~10.3~~]~~ (Ref. 1).

APPLICABLE SAFETY ANALYSES (2) (2)

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, discussed in the FSAR, Section ~~[~~6.2~~]~~ (Ref. 2). It is also affected by the accident analysis of the SLB events presented in the FSAR, Section ~~[15.1.5]~~ 15.4.2.1 (Ref. 3). The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand).

The limiting case for the containment analysis is the SLB inside containment, with a loss of offsite power following turbine trip, and failure of the MSIV on the affected steam

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

generator to close. At lower powers, the steam generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the containment. Due to reverse flow and failure of the MSIV to close, the additional mass and energy in the steam headers downstream from the other MSIV contributes to the total release. With the most reactive rod cluster control assembly assumed stuck in the fully withdrawn position, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the boric acid injection delivered by the Emergency Core Cooling System.

The accident analysis compares several different SLB events against different acceptance criteria. The large SLB outside containment upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The large SLB inside containment at hot zero power is the limiting case for a post trip return to power. The analysis includes scenarios with offsite power available, and with a loss of offsite power following turbine trip. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System cooldown. With a loss of offsite power, the response of mitigating systems is delayed. Significant single failures considered include failure of an MSIV to close.

The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:

a. An HELB inside containment. In order to maximize the mass and energy release into containment, the analysis assumes that the MSIV in the affected steam generator remains open. For this accident scenario, steam is discharged into containment from all steam generators until the remaining MSIVs close. After MSIV closure, steam is discharged into containment only from the affected steam generator and from the residual steam in the main steam header downstream of the closed MSIVs in the unaffected loops. Closure of the MSIVs isolates the break from the unaffected steam generators.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. A break outside of containment and upstream from the MSIVs is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs isolates the break, and limits the blowdown to a single steam generator.

c. A break downstream of the MSIVs will be isolated by the closure of the MSIVs.

d. Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator from the intact steam generators. In addition to minimizing radiological releases, this enables the operator to maintain the pressure of the steam generator with the ruptured tube below the MSSV setpoints, a necessary step toward isolating the flow through the rupture.

e. The MSIVs are also utilized during other events such as a feedwater line break. This event is less limiting so far as MSIV OPERABILITY is concerned.

The MSIVs satisfy Criterion 3 of the NRC Policy Statement.

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9 9 2

LCO

This LCO requires that [four] MSIVs in the steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation actuation signal.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 limits (Ref. 4) or the NRC staff-approved licensing basis.

APPLICABILITY

A

The MSIVs must be OPERABLE in MODE 1 and in MODES 2 and 3 ~~with any MSIV open~~, when there is significant mass and energy in the RCS and steam generators~~; therefore, the MSIVs must be OPERABLE or closed~~. When the MSIVs are closed, they are already performing the safety function.

(continued)

BASES

APPLICABILITY (continued)

In MODE 4, normally most of the MSIVs are closed, and the steam generator energy is low.

In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high-energy secondary-system pipe breaks in these MODES.

ACTIONS

A.1

With one MSIV inoperable in MODE 1, action must be taken to restore OPERABLE status within (8) hours. Some repairs to the MSIV can be made with the unit hot. The (8)-hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the MSIVs. ②

72

The (8)-hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from other containment isolation valves in that the closed system provides an additional means for containment isolation. ②

A

A

B.1 and B.2

the unit must be placed in a MODE in which the LCO does not apply. To acheive this status,

② If the MSIV cannot be restored to OPERABLE status within (8) hours, the unit must be placed into MODE 2 within 6 hours ~~and the inoperable MSIV must be closed within the next 6 hours~~. The Completion Times are reasonable, based on operating experience, to reach MODE 2 and to close the MSIVs in an orderly manner and without challenging unit systems. A

A

and Condition C would be entered.

C.1 and C.2

Condition C is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODES 2 and 3, the inoperable MSIVs may either be restored to OPERABLE status or closed. When closed, the MSIVs are

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

already in the position required by the assumptions in the safety analysis.

Ⓐ ⑤ The [8]-hour Completion Time is consistent with that allowed in Condition A~~, and conservative considering the reduced energy in the steam generators in MODES 2 and 3~~.

For inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, the inoperable MSIVs must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The ~~12-hour~~ 7-day Completion Time is reasonable, based on engineering judgment, in view of MSIV status indications available in the control room, and other administrative controls, to ensure that these valves are in the closed position. Ⓐ

D.1 and D.2

If the MSIVs cannot be restored to OPERABLE status, or are not closed within the associated Completion Time the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems. Ⓐ

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1 ③

This SR verifies that MSIV closure time is ≤ ~~[4.6]~~ 5.0 seconds on an actual or simulated actuation signal. The MSIV closure time is assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The MSIVs should not be tested at power, since even a part-stroke exercise increases the risk of a valve closure when the unit is generating power. As the MSIVs are not tested at power, they are exempt from the ASME Code,

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1 (continued)

Section XI (Ref. 5) requirements during operation in MODE 1 or 2.

(2) The Frequency is in accordance with the [Inservice Testing Program or [18] months]. The [18]-month Frequency for valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the (2) surveillance when performed at the [18]-month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

This test is conducted in MODE 3 with the unit at operating temperature and pressure, as discussed in Reference 5 exercising requirements. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

REFERENCES

(1)

1. Watts Bar FSAR, Section [10.3], "Main Steam Supply System."
2. Watts Bar FSAR, Section [6.2], "Containment Systems."
3. Watts Bar FSAR, Section ~~[15.1.5]~~ 15.4.2.1, "Major Rupture of a Main Steam Line."
4. 10 CFR 100.11.
5. American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, Section XI, Inservice Inspection, Article IWV-3400, "Inservice Tests – Category A and B Valves."

JUSTIFICATIONS FOR DEVIATIONS FROM STS 3.7.2

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

1. Change to specify Watts Bar specific Reference information and format.

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

3. Change to reflect Watts Bar specific information or values.

4. Change to correct error(s) in the STS.

5. Watts Bar considers this statement to be an error in the STS. Westinghouse has been notified and agreed that this change is appropriate and that the standard is incorrect.

3.7 PLANT SYSTEMS

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

LCO 3.7.3 [Four] MFIVs, [four] MFRVs, [and associated bypass valves] shall be OPERABLE. (1)

APPLICABILITY: MODES 1, MODE 2, and 3 ~~with any~~ ; except when MFIV, MFRV, [or associated bypass valve] ~~open and not isolated~~ is closed and deactivated. (A) (1) (2)

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more MFIV(s) inoperable.	A.1 Close or isolate MFIV(s). (A) AND A.2 Verify MFIV(s) is closed or isolated.	[72] hours (1) Once per 7 days
B. One or more MFRV(s) inoperable.	B.1 Close or isolate MFRV(s). (A) AND B.2 Verify MFRV(s) is closed or isolated.	[72] hours (1) Once per 7 days

(continued)

ACTIONS (continued)

Restore bypass valve to OPERABLE status.

CONDITION	REQUIRED ACTION	COMPLETION TIME
(A) (3)(4) C. One or more MFIV or MFRV bypass valve(s) inoperable.	C.1 Close or isolate bypass valve(s). 6 AND C.2 Verify (A) bypass valve(s) is closed or isolated.	[72] hours Once per 7 days (7)
(7) (A) D. ~~Two valves~~ One MFIV and MFRV in the same flow path inoperable. ~~for one or more flow paths.~~	D.1 ~~Close one valve or otherwise~~ Isolate affected flow path. (A)	8 hours
(7) F ~~E~~. Required Action and associated Completion Time not met.	F ~~E~~.1 Be in MODE 3. AND F ~~E~~.2 Be in MODE 4.	6 hours 12 hours
E. One MFIV and MFRV bypass ~~line~~ valve in the same flow path inoperable.	E.1 Restore one MFIV or MFRV bypass valve to OPERABLE status.	8 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	(A) ~~-----NOTE----- Only required to be performed prior to entry into MODE 2.~~ (1) Verify the closure time of each MFIV, MFRV, ~~[6]~~ and associated bypass valve ~~[s]~~ is ≤ ~~[7]~~ [6.5] seconds on an actual or simulated actuation signal.	In accordance with the ~~[~~Inservice Testing Program ~~,~~ or 18 months ~~]~~ (A)

B 3.7 PLANT SYSTEMS

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

BASES

BACKGROUND

The MFIVs isolate main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break (HELB). The safety related function of the MFRVs is to provide the second isolation of MFW flow to the secondary side of the steam generators following an HELB. Closure of the MFIVs and associated bypass valves or MFRVs and associated bypass valves terminates flow to the steam generators, terminating the event for feedwater line breaks (FWLBs) occurring upstream of the MFIVs or MFRVs. The consequences of events occurring in the main steam lines or in the MFW lines downstream from the MFIVs will be mitigated by their closure. Closure of the MFIVs and associated bypass valves, or MFRVs and associated bypass valves, effectively terminates the addition of normal feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FWLBs inside containment, and reducing the cooldown effects for SLBs. 3

5 The MFIVs and associated bypass valves~~, or MFRVs and associated bypass valves,~~ isolate the non-safety-related portions from the safety-related portions of the system. In the event of a secondary side pipe rupture inside containment, the valves limit the quantity of high energy fluid that enters containment through the break, and provide a pressure boundary for the controlled addition of auxiliary feedwater (AFW) to the intact loops.

One MFIV ~~and associated bypass valve~~ and 16-inch one MFRV ~~and its associated bypass valve~~ are located on each MFW line~~, outside but close to containment~~. ~~The MFIVs and MFRVs are located upstream of the~~ The AFW injection point is located upstream of the bypass MFIV, so that AFW may be supplied to the steam generators following a bypass MFIV or MFRV closure. The piping volume between the check valve and ~~from these valves to~~ the steam generators must be accounted for in calculating mass and energy releases, and refilled prior to AFW reaching the steam generator following either an SLB or FWLB. 4

4 One bypass MFRV and one bypass MFIV are located on a smaller 6-inch and tempering flow startup feedwater line. Both the MFIV and bypass MFIV are located in the main steam valve vault close to containment.

The MFIVs and associated bypass valves, and MFRVs and associated bypass valves close on receipt of a T_{avg} - Low coincident with reactor trip (P-4), Safety injection signal, or steam generator water 5

(continued)

BASES

except for the bypass MFIV which has no handswitch.

BACKGROUND (continued)

level — high-high signal. They may also be Closed ~~actuated~~ manually. In addition to the MFIVs and associated bypass valves, and the MFRVs and associated bypass valves, a check valve ~~inside containment is available. The check valve isolates the feedwater line, penetrating containment and ensures that the consequences of events do not exceed the capacity of the containment heat removal systems.~~ (5)

is located just outside Containment in the main steam valve vault. The Check valve terminates flow from the SG for breaks upstream of the check valve. (5) (1)

A description of the MFIVs and MFRVs is found in the FSAR, Section [10.4.7] (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the MFIVs and MFRVs and associated bypass valves is established by the analyses for the large SLB. It is also influenced by the accident analysis for the large FWLB. Closure of the MFIVs and associated bypass valves, or MFRVs and associated bypass valves, may also be relied on to terminate an SLB for core response analysis and excess feedwater event ~~upon the receipt of a steam generator water level — high-high signal or a main steam isolation signal on high steam generator level~~ the safety injection signal initiated by a steamline break inside or outside containment. (5) (8) (5)

Failure of an ~~MFIV, MFRV,~~ MFIV, MFRV, or the associated bypass valves, in a single flow path to close following an SLB or FWLB, can result in additional mass and energy being delivered to the steam generators, contributing to cooldown. This failure also results in additional mass and energy releases following an SLB or FWLB event.

The MFIVs and MFRVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

This LCO ensures that ~~the MFIVs and the MFRVs and their associated bypass valves~~ The MFIVs and bypass MFIVs will isolate main feedwater flow to the steam generators, following an FWLB or ~~main steam line break.~~ SLB. ~~These valves~~ will also isolate the non-safety-related portions from the safety-related portions of the system. (5)

This LCO requires that ~~[four]~~ MFIVs and associated bypass valves, and ~~[four]~~ MFRVs ~~[and associated bypass valves]~~, ~~each of the feedwater lines~~ be OPERABLE. The MFIVs and MFRVs and the associated bypass valves are considered (1) (A) (7)

(continued)

BASES

LCO (continued)

OPERABLE when isolation times are within limits and they close on an isolation actuation signal.

(A)

(5) Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. If a main steam feedwater isolation signal on high -high steam generator level is relied on to terminate an excess feedwater flow event, failure to meet the LCO may result in the introduction of water into the main steam lines.

APPLICABILITY

The MFIVs and MFRVs and the associated bypass valves must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and steam generators. This ensures that, in the event of an HELB, a single failure cannot result in the blowdown of more than one steam generator. In MODES 1, 2, and 3, the MFIVs and MFRVs and the associated bypass valves are required to be OPERABLE, except when closed and deactivated, to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed and deactivated or isolated by a closed manual valve, they are already performing their safety function. (2)

In MODES 4, 5, and 6 steam generator energy is low. Therefore, the MFIVs and MFRVs and the associated bypass valves are normally closed since MFW is not required. (A)

ACTIONS

The Actions table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1 and A.2

With one MFIV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within (1) 72 8 hours. When these valves are closed or isolated, they are performing their required safety function.

(1) The 72 8-hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

low probability of an event occurring during this time period that would require isolation of the main feedwater flow paths. The (72) hour Completion Time is reasonable, based on operating experience.

Inoperable MFIVs, that are closed or isolated, must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7-day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

B.1 and B.2

With one MFRV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within (72) hours. When these valves are closed or isolated, they are performing their required safety function.

The (72)-hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the main feedwater flow paths. The (72)-hour Completion Time is reasonable, based on operating experience.

Inoperable MFRVs, that are closed or isolated, must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7-day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls to ensure that the valves are closed or isolated.

(continued)

The inoperable valve should not be closed and isolated for long time periods since the 6-inch bypass line provides a small tempering flow to the upper SG nozzle. This limits the temperature difference between the SG and condensate storage tank fluid which would be supplied by the AFW System. The 6-inch line may be isolated for short periods of time to support Calorimetric Flow Measurements.

BASES

ACTIONS (continued)

C.1 ~~and C.2~~

(3)(4) With one ~~associated~~ MFIV or MFRV bypass valve in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status ~~or to close or isolate inoperable affected valves~~ within (1)72 hours. ~~When these valves are closed or isolated, they are performing their required safety function.~~ (7)

(1) The (1)72-hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the main feedwater flow paths. The (1)72-hour Completion Time is reasonable, based on operating experience.

~~(3)(4)~~ (7) ~~Inoperable associated MFIV or MFRV bypass valves, that are closed or isolated, must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7-day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.~~

D.1

(7) With ~~two~~ an MFIV and MFRV inoperable ~~valves~~ in the same flow path, there may be no redundant system to operate automatically and perform the required safety function. Although the containment can be isolated with the failure of two valves in parallel in the same flow path, the double failure can be an indication of a common mode failure in the valves of this flow path, and as such, is treated the same as a loss of the isolation capability of this flow path. Under these conditions, affected valves in each flow path must be restored to OPERABLE status, ~~or closed~~, or the affected flow path isolated within 8 hours. This action returns the system to the condition where at least one valve in each flow path is performing the required safety function. The 8-hour Completion Time is reasonable, based on operating experience, to complete the actions required to close the MFIV, or MFRV, or otherwise isolate the affected flow path. (A)

(7) INSERT

(continued)

INSERT

E.1

With two bypass valves in the same flow path inoperable, there may be no redundant system to operate automatically and perform the required safety function. Under these conditions, at least one valve in the flow path must be restored to OPERABLE status within 8 hours. The Completion Time of 8 hours is consistent with Condition D.

BASES

ACTIONS (continued)

F ~~E~~.1 and F ~~E~~.2

If the MFIV(s) and MFRV(s) and the associated bypass valve(s) cannot be restored to OPERABLE status, or closed or isolated within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience to reach the required unit conditions from full-power conditions in an orderly manner and without challenging unit systems.

7 the MFIV(s) or MFRV(s)

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFIV, MFRV, and associated bypass valves is ≤ [7] on an actual or simulated actuation signal. The MFIV and MFRV closure times are assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. These valves should not be tested at power since even a part-stroke exercise increases the risk of a valve closure with the unit generating power. As these valves are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 2) requirements during operation in MODES 1 and 2. 5

6.5 seconds

2 This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing in MODE ~~2~~ 3 in order to establish conditions consistent with those under which the acceptance criterion was generated. A

1 The Frequency for this SR is in accordance with the [Inservice Testing Program or [18] months]. The [18]-month Frequency for valve closure is based on the refueling cycle. Operating experience has shown that these components usually
1 pass the SR when performed at the [18]-month Frequency.

REFERENCES 6

1. FSAR, Section [10.4.7]., "Condensate and Feedwater Systems"
2. American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, Section XI.

JUSTIFICATIONS FOR DEVIATIONS FROM STS 3.7.3

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

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

2. Change to correct error(s) in the STS.

3. Change to specify Watts Bar specific terminology for clarity.

4. The Watts Bar design has two feed lines per SG, a normal and a bypass. The bypass line is used for startup to approximately 22% load and enters the SG through a separate upper feed nozzle. The AFW line enters the bypass line downstream of a check valve just downstream of the bypass isolation valve. The main feed line enters the SG through a lower nozzle. Both feed lines have a regulator and isolation valve. These changes to the STS are necessary to reflect this specific plant design.

5. Change required to specify plant specific parameter values or information.

6. Watts Bar prefers to use the expanded reference information and format.

7. Conditions have been modified to preclude isolation of a bypass line. The bypass lines for the WBN SG must maintain a small tempering flow to cool the upper nozzle. Therefore, if the cause of the inoperability cannot be corrected, a shutdown would be required.

8. Change to delete redundancy to previous statements.

3.7 PLANT SYSTEMS

3.7.4 Atmospheric Dump Valves (ADVs)

LCO 3.7.4 4 [Three] ADV lines shall be OPERABLE. 1

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required ADV line inoperable.	------------NOTE------------ LCO 3.0.4 is not applicable. ---------------------------- A.1 Restore required ADV line to OPERABLE status.	 7 days
B. Two or more required ADV lines inoperable.	B.1 Restore one ADV line to OPERABLE status.	24 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4 without reliance upon steam generator for heat removal.	6 hours [18] hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify one complete cycle of each ADV.	[18] months ①
[SR 3.7.4.2	Verify one complete cycle of each ADV block valve.	[18] months ①

B 3.7 PLANT SYSTEMS

B 3.7.4 Atmospheric Dump Valves (ADVs)

BASES

BACKGROUND

(2) (1) The ADVs provide a method for cooling the unit to residual heat removal (RHR) entry conditions should the preferred heat sink via the Steam Dump ~~Bypass~~ System to the condenser not be available, as discussed in the FSAR, Section [10.3] (Ref. 1). This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the condensate storage tank (CST). (3) The ADVs may also be ~~required~~ used to meet the design cooldown rate during a normal cooldown ~~when steam~~ if ~~pressure drops too low for maintenance of a vacuum in~~ the condenser is not available to permit use of the Steam Dump System.

(1) One ADV line for each of the [four] steam generators is provided. Each ADV line consists of one ADV and an associated block valve.

The ADVs are provided with upstream block valves to permit their being tested at power, and to provide an alternate means of isolation. The ADVs are equipped with pneumatic controllers to permit control of the cooldown rate.

(4) The ADVs are ~~usually~~ provided with a pressurized air ~~gas~~ supply from the auxiliary air compressors ~~of bottled nitrogen~~ that, on a loss of pressure in ~~that~~ the normal instrument air supply, automatically supplies backup air ~~nitrogen~~ to operate the ADVs. ~~The nitrogen supply is sized to provide the sufficient pressurized gas to operate the ADVs for the time required for Reactor Coolant System cooldown to RHR entry conditions.~~

A description of the ADVs is found in Reference 1. The ADVs are OPERABLE with ~~only~~ a dc-power source and control air available. In addition, handwheels are provided for local manual operation. (4)

APPLICABLE SAFETY ANALYSES

(4) The design basis of the ADVs is established by the capability to cool the unit to RHR entry conditions. The maximum design rate of ~~[75]~~ 50 °F per hour is applicable for two steam generators, each with one ADV. This rate is adequate to cool the unit to RHR entry conditions ~~with only one steam~~ (5)

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

(5) s ~~generator and one ADV,~~ utilizing the cooling water supply available in the CST.

In the accident analysis presented in Reference 1, the ADVs are assumed to be used by the operator to cool down the unit to RHR entry conditions for accidents accompanied by a loss of offsite power. Prior to operator actions to cool down the unit, the ADVs and main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the steam generator pressure below the design value. For the recovery from a steam generator tube rupture (SGTR) event, the operator is also required to perform a limited cooldown to establish adequate subcooling as a necessary step to terminate the primary-to-secondary break flow into the ruptured steam generator. The time required to terminate the primary-to-secondary break flow for a SGTR is more critical than the time required to cool down to RHR conditions for this event and also for other accidents. Thus, the SGTR is the limiting event for the ADVs. (4) ~~The~~ Three ~~number of~~ ADVs are required to be OPERABLE to satisfy the SGTR accident analysis requirements. ~~depends upon the number of unit loops and consideration of~~ This considers any single failure assumptions regarding the failure of one ADV to open on demand.

The ADVs are equipped with block valves in the event an ADV spuriously fails to open or fails to close during use.

The ADVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

(1) (1) ~~[~~Three~~]~~ ADV lines are required to be OPERABLE. One ADV line is required from each of ~~[~~three~~]~~ steam generators to ensure that at least one ADV line is available to conduct a unit cooldown following an SGTR, in which one steam generator becomes unavailable, accompanied by a single, active failure of a second ADV line on an unaffected steam generator. The block valves must be OPERABLE to isolate a failed open ADV line. (5) ~~A closed block valve does not render it or its ADV line inoperable if operator action time to open the block valve is supported in the accident analysis.~~

Failure to meet the LCO can result in the inability to cool the unit to RHR entry conditions following an event in which

(continued)

BASES

LCO (continued)

the condenser is unavailable for use with the Steam Bypass Dump System. 2

An ADV is considered OPERABLE when it is capable of providing controlled relief of the main steam flow and capable of fully opening and closing on demand.

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the ADVs are required to be OPERABLE.

In MODE 5 or 6, an SGTR is not a credible event.

ACTIONS

A.1

With one required ADV line inoperable, action must be taken to restore OPERABLE status within 7 days. The 7-day Completion Time allows for the redundant capability afforded by the remaining OPERABLE ADV lines, a non-safety-grade backup in the Steam Bypass Dump System, and MSSVs. Required Action A.1 is modified by a Note indicating that LCO 3.0.4 does not apply. 2

B.1

With two or more ADV lines inoperable, action must be taken to restore all but one ADV line to OPERABLE status. Since the block valve can be closed to isolate an ADV, some repairs may be possible with the unit at power. The 24-hour Completion Time is reasonable to repair inoperable ADV lines, based on the availability of the Steam Bypass Dump System and MSSVs, and the low probability of an event occurring during this period that would require the ADV lines. 2

C.1 and C.2

If the ADV lines cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

6

MODE 3 within 6 hours, and in MODE 4, without reliance upon steam generator for heat removal, within [18] hours. The allowed Completion Times are reasonable, based on operating experience to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. 1

A

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

either remotely or locally

To perform a controlled cooldown of the RCS, the ADVs must be able to be opened and throttled through their full range. This SR ensures that the ADVs are tested through a full-control cycle at least once per fuel cycle. Performance of inservice testing or use of an ADV during a unit cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the [18]-month Frequency. The Frequency is acceptable from a reliability standpoint. 1

1

[SR 3.7.4.2

The function of the block valve is to isolate a failed open ADV. Cycling the block valve both closed and open demonstrates its capability to perform this function. Performance of inservice testing or use of the block valve during unit cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the [18]-month Frequency. The Frequency is acceptable from a reliability standpoint.]

REFERENCES 7

1. Watts Bar FSAR, Section [10.3], "Main Steam Supply System".

JUSTIFICATIONS FOR DEVIATIONS FROM STS 3.7.4

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

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

2. Change to specify Watts Bar specific terminology for clarity.

3. Watts Bar believes it highly unlikely that the ADVs would be required for normal cooldown on a routine basis and has made a change to denote that the ADVs may be used <u>if</u> condenser vacuum is lost. As this statement is currently written, it implies that this situation is considered routine.

4. Change to specify Watts Bar specific parameter values or information.

5. Change to delete information that does not apply to Watts Bar.

6. Change to correct error(s) in the STS.

7. Change to specify Watts Bar specific Reference information and format.

3.7 PLANT SYSTEMS

3.7.5 Auxiliary Feedwater (AFW) System

LCO 3.7.5 1 [Three] AFW trains shall be OPERABLE.

1

-----------------------------NOTE-----------------------------
Only one AFW train, which includes a motor-driven pump, is required to be OPERABLE in MODE 4.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
[A. One steam supply to turbine-driven AFW pump inoperable.	A.1 Restore steam supply to OPERABLE status.	7 days AND 10 days from from discovery of failure to meet ~~this~~ the LCO] A
B. One AFW train inoperable in MODE 1, 2, or 3 [for reasons other than Condition A]. 1	B.1 Restore AFW train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet ~~this~~ the LCO A

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
1 C. Required Action and associated Completion Time for Condition A [or B] not met. OR Two AFW trains inoperable in MODE 1, 2, or 3.	C.1 Be in MODE 3. AND C.2 ~~-----NOTE----- Only required if one residual heat removal loop is OPERABLE. ----------~~ Be in MODE 4.	6 hours A [18] hours 1
1 D. [Three] AFW trains inoperable in MODE 1, 2, or 3.	------------NOTE------------ LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status. ---------------------------- D.1 Initiate action to restore one AFW train to OPERABLE status.	Immediately
E. Required AFW train inoperable in MODE 4.	E.1 Initiate Action restore AFW train to OPERABLE status. A ~~OR~~ ~~E.2 Verify by administrative means two loops are OPERABLE for removal of decay heat in accordance with LCO 3.4.6, "RCS Loops – MODE 4."~~	Immediately ~~Immediately~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1 1	Verify each AFW manual, power-operated, and automatic valve in each water flow path, [and in both steam supply flow paths to the steam turbine-driven pump,] that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.5.2 1 2 A	~~-----NOTE----- For the turbine-driven AFW pump, only required to be performed [24 hours] after reaching [ ] psig in the steam generator.~~ Not until 24 hours ≥ ~~1020~~ 1092 Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	[31] days on a STAGGERED TEST BASIS 1
SR 3.7.5.3	Verify each AFW automatic valve actuates to the correct position on an actual or simulated actuation signal when in MODES 1, 2, and 3 A	[18] months 1
SR 3.7.5.4 1 2 A	~~-----NOTE----- For the turbine-driven AFW pump, only required to be performed [24 hours] after reaching [ ] psig in the steam generator.~~ ~~1020~~ 1092 Verify each AFW pump starts automatically on an actual or simulated actuation signal when in MODES 1, 2, and 3 A	[18] months 1

(continued)

-------NOTE-------
Not required to be performed for the turbine-driven AFW pump until 24 hours after ≥ [1092] psig in the steam generator.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.5.5 1	Verify proper alignment of the required AFW flow paths by verifying flow from the condensate storage tank to each steam generator.	Prior to entering MODE 2, whenever unit has been in MODE 5 or 6 for > 30 days e

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply. The AFW pumps take suction through separate and independent suction lines from the condensate storage tank (CST) (LCO 3.7.6, "Condensate Storage Tank (CST)") and pump to the steam generator secondary side via separate and independent
7 8 connections to the main feedwater (MFW) piping. outside bypass line
containment. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)") or atmospheric dump valves (LCO 3.7.4, "Atmospheric Dump Valves (ADVs)"). If the main condenser is available, steam may be released via the steam
2 dump ~~bypass~~ valves and recirculated to the CST.

410 gpm

1 The AFW System consists of [two] motor-driven AFW pumps and steam turbine-driven pump configured into [three] trains. Each motor-driven pump provides ~~[100%]~~ of AFW flow, ~~capacity~~, 2
and the turbine-driven pump provides ~~[200%] of the required capacity~~ to the steam generators, as assumed in the accident analysis. The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system. Each motor-driven AFW pump is powered from an independent Class 1E power supply and feeds [two] steam
3 generators, ~~although each pump has the capability to be realigned from the control room to feed other steam generators.~~ The steam turbine-driven AFW pump receives
2 one of steam from two main steam lines upstream of the main steam isolation valves. Each of the steam feed lines will supply 100% of the requirements of the turbine-driven AFW pump.

720 gpm

2 The AFW System is capable of supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions; however, the main feedwater system will normally perform these functions.

The turbine-driven AFW pump supplies a common header capable of feeding all steam generators ~~with dc-powered control valves actuated to the appropriate steam generator by the~~
3 ~~Engineered Safety Feature Actuation System (ESFAS)~~. One

(continued)

BASES

BACKGROUND (continued)

pump at full flow is sufficient to remove decay heat and cool the unit to residual heat removal (RHR) entry conditions. Thus, the requirement for diversity in motive power sources for the AFW System is met.

2

The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the lowest setpoint of the MSSVs. Subsequently, the AFW System supplies sufficient water to cool the unit to RHR entry conditions, with steam released through the ADVs.

The AFW System actuates automatically on steam generator water level — low-low by the ESFAS (LCO 3.3.2, "Engineered Safety Feature Activation System (ESFAS)"). The system also actuates on loss of offsite power, safety injection, and trip of ~~all~~ both turbine-driven MFW pumps. 2

The AFW System is discussed in the FSAR, Section [10.4.9] (Ref. 1).

APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event with loss of normal feedwater.

The design basis of the AFW System is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest steam generator safety valve set-pressure plus ~~3%~~ 1% for setpoint tolerance, and 3% for accumulation. 2

In addition, the AFW System must supply enough makeup water to replace steam generator secondary inventory lost as the unit cools to MODE 4 conditions. Sufficient AFW flow must also be available to account for flow losses such as pump recirculation and line breaks.

The limiting Design Basis Accidents (DBAs) and transients for the AFW System are as follows:

a. Feedwater Line Break (FWLB); and

b. Loss of main feedwater.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

In addition, the minimum available AFW flow and system characteristics are serious considerations in the analysis of a small break loss-of-coolant accident (LOCA).

3 The AFW System design is such that it can perform its function following an FWLB between the main feedwater isolation valves and containment, combined with a loss of offsite power following turbine trip, and a single active failure of the steam turbine-driven AFW pump. In such a case, the ESFAS logic may not detect the affected steam generator if the backflow check valve to the affected MFW header worked properly. One motor-driven AFW pump would deliver to the broken MFW header ~~at the pump runout flow~~ until the problem was detected, and flow terminated by the operator. 9

within 10 minutes

Sufficient flow would be delivered to the intact steam generator by the redundant AFW pump.

The ESFAS automatically actuates the AFW turbine-driven pump and associated power operated valves and controls when required to ensure an adequate feedwater supply to the steam generators during loss of power. DC-power operated valves are ~~provided for~~ each AFW line to control the AFW flow to ~~each steam~~ generator.

INSERT A

4 The AFW System satisfies the requirements of Criterion 3 of the NRC Policy Statement.

LCO

1 This LCO provides assurance that the AFW System will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary. Three independent AFW pumps in three diverse trains are required to be OPERABLE to ensure the availability of residual heat removal capability for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering two of the pumps from independent emergency buses. The third AFW pump is powered by a different means, a steam-driven turbine supplied with steam from a source that is not isolated by closure of the MSIVs.

10 ~~The AFW System is configured into three trains.~~ The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the steam generators are OPERABLE. This requires that the two

(continued)

INSERT A

Each motor-driven auxiliary feedwater pump (one Train A and one Train B) supplies flow paths to two steam generators. Each flow path contains an automatic air-operated level control valve (LCV). The LCVs have the same train designation as the associated pump and are provided trained air. The turbine-driven auxiliary feedwater pump supplies flow paths to all four steam generators. Each of these flow paths contains an automatic air-operated LCV, two of which are designated as Train A, receive A-train air and provide flow to the same steam generators that are supplied by the B-train motor-driven auxiliary feedwater pump. The remaining two LCVs are designated as Train B, receive B-train air, and provide flow to the same steam generators that are supplied by the A-train motor-driven pump. This design provides the required redundancy to ensure that at least two steam generators receive the necessary flow assuming any single failure. It can be seen from the description provided above that the loss of a single train of air (A or B) will not prevent the auxiliary feedwater system from performing its intended safety function and is no more severe than the loss of a single auxiliary feedwater pump. Therefore, the loss of a single train of auxiliary air only affects the capability of a single motor-driven auxiliary feedwater pump because the turbine-driven pump is still capable of providing flow to the two steam generators that are separated from the other motor-driven pump.

BASES

LCO (continued)

motor-driven AFW pumps be OPERABLE in two diverse paths, each supplying AFW to separate steam generators. The turbine-driven AFW pump is required to be OPERABLE with redundant steam supplies from each of two main steam lines upstream of the MSIVs, and shall be capable of supplying AFW to any of the steam generators. The piping, valves, instrumentation, and controls in the required flow paths also are required to be OPERABLE. 1 1

The LCO is modified by a Note indicating that one AFW train, which includes a motor-driven pump, is required to be OPERABLE in MODE 4. This is because of the reduced heat removal requirements and short period of time in MODE 4 during which the AFW is required and the insufficient steam available in MODE 4 to power the turbine-driven AFW pump.

APPLICABILITY

In MODES 1, 2, and 3, the AFW System is required to be OPERABLE in the event that it is called upon to function when the main feedwater is lost. In addition, the AFW System is required to supply enough makeup water to replace the steam generator secondary inventory, lost as the unit cools to MODE 4 conditions.

In MODE 4 the AFW System may be used for heat removal via the steam generators.

In MODE 5 or 6, the steam generators are not normally used for heat removal, and the AFW System is not required.

(continued)

BASES (continued)

ACTIONS

A.1

If one of the two steam supplies to the turbine-driven AFW train is inoperable, action must be taken to restore OPERABLE status within 7 days. The 7-day Completion Time is reasonable, based on the following reasons:

1

a. The redundant OPERABLE steam supply to the turbine-driven AFW pump;

b. The availability of redundant OPERABLE motor-driven AFW pump; and

5

5

c. The low probability of an event occurring that requires the inoperable steam supply to the turbine-driven AFW pump.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any single continuous occurrence of failing to meet this LCO.

The 10-day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 7 days and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

B.1

1

With one of the required AFW trains (pump or flow path) inoperable in MODE 1, 2, or 3 for reasons other than Condition A, action must be taken to restore OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines to the turbine-driven AFW pump. The 72-hour Completion Time is reasonable, based on redundant capabilities afforded by the AFW System, time needed for repairs, and the low probability of a DBA occurring during this time period.

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any

(continued)

BASES

ACTIONS

B.1 (continued)

combination of Conditions to be inoperable during any single continuous occurrence of failing to meet this LCO.

The 10-day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 72 hours and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

C.1 and C.2

1 When Required Action A.1 [or B.1] cannot be completed within the required Completion Time, or if two AFW trains are inoperable in MODE 1, 2, or 3, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within
1 6 hours, and in MODE 4 within [18] hours.

Required Action C.2 is modified by a Note indicating that entry into MODE 4 is only required if one RHR loop is OPERABLE. The Note also is intended to convey the suspension of further action to reach MODE 4 if, while in Required Action C.2, all RHR loops become inoperable. 2 The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full-power conditions in an orderly manner and without challenging unit systems.

In MODE 4 with two AFW trains inoperable, operation is allowed to continue because only one motor-driven pump AFW train is required in accordance with the Note that modifies the LCO. Although not required, the unit may continue to cool down and initiate RHR.

D.1

1 If all [three] AFW trains are inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with non-safety-related

(continued)

BASES

ACTIONS

D.1 (continued)

equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one AFW train to OPERABLE status.

Required Action D.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one AFW train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the unit into a less safe condition.

E.1 and E.2 A required

In MODE 4, either the RCPs or the RHR loops can be used to provide forced circulation. This is addressed in LCO 3.4.6, "RCS Loops — MODE 4." With one AFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status or to immediately verify, by administrative means the OPERABILITY of a second train. The immediate Completion Time is consistent with LCO 3.4.6.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power-operated, and automatic valves in the AFW System water and steam supply flow paths, provides assurance that the proper flow paths will exist for AFW operation. This SR 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 locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.2

2 lowest set plus 1% setpoint tolerance and 3% accumulation

This SR verifies that the AFW pumps develop sufficient discharge pressure to deliver the required flow at the ~~full open~~ pressure of the MSSVs. Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. Periodically comparing the reference differential pressure developed at this reduced flow, detects trends that might be indicative of incipient failure. Performance of inservice testing discussed in the ASME Code, Section XI (Ref. 2) (only required at 3-month intervals) satisfies this requirement. (1) The [31]-day Frequency on a STAGGERED TEST BASIS results in testing each pump once every 3 months, as required by Reference 2.

(1) This SR is modified by a Note indicating that the SR be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.

SR 3.7.5.3

(A) in MODES 1, 2, and 3. In MODE 4 the required AFW train is already aligned and operating, therefore, this SR is not required

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. (1) The [18]-month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. (1) The [18]-month Frequency is acceptable based on operating experience and the design reliability of the equipment.

SR 3.7.5.4

(A) in MODES 1, 2, and 3. In MODE 4, the required pump is already operating and the auto-start function is not required

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an ESFAS by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal. (1) The [18]-month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.4 (continued)

because of the potential for an unplanned transient if the surveillance were performed with the reactor at power.

1 5

This SR is modified by a Note indicating that the SR ~~to~~ be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.

SR 3.7.5.5

This SR verifies that the AFW is properly aligned by verifying the flow paths from the CST to each steam generator prior to entering MODE 2 after more than 30 days in MODE 5 or 6. OPERABILITY of AFW flow paths must be verified before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgment, and other administrative controls ensuring that flow ~~paths~~ paths remain OPERABLE. To further ensure AFW System alignment, flow-path OPERABILITY is verified, following extended outages to determine no misalignment of valves has occurred. This SR ensures that the flow path from the CST to the steam generators is properly aligned. ~~(This SR is not required by those units that use AFW for normal startup and shutdown.)~~

1 5

3

REFERENCES

6

1. Watts Bar FSAR, Section 10.4.9, "Auxiliary Feedwater System".

2. American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, Section XI, Inservice Inspection, Article IWV-3400, "Inservice Tests – Category A and B Valves".

JUSTIFICATIONS FOR DEVIATIONS FROM STS 3.7.5

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

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

2. Change to reflect Watts Bar specific information or parameter values.

3. Change to delete information that does not apply to Watts Bar.

4. This discussion has been added to discuss the affect on operability by the loss of control air to a single train of AFW and describe valve controls. This discussion is consistent with that in the Bases of the Sequoyah Nuclear Plant Technical Specifications.

5. Change to correct error(s) in the STS.

6. Change to specify Watts Bar specific Reference information and format.

7. The AFW piping connect to two loops inside containment and the other two loops connect outside containment.

8. Editorial change. LOCA assumptions and FWLB are all important considerations in the design of AFW Systems.

9. The AFW System is equipped with pressure control valves to prevent the pump from reaching runout flow.

10. Change to remove redundancy to earlier statement.

3.7 PLANT SYSTEMS

3.7.6 Condensate Storage Tank (CST)

LCO 3.7.6 The CST level shall be ≥ [~~110,000 gal~~]. 210,000 gal. (1)

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. CST level not within limit.	A.1	Verify by administrative means OPERABILITY of ERCW backup water supply.	4 hours AND (2) Once per 12 hours thereafter
	AND		
	A.2	Restore CST level to within limit.	7 days
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	~~-----NOTE----- Only required if one residual heat removal loop is OPERABLE. ---------~~ (A) Be in MODE 4, without reliance on steam generator for heat removal.	[~~1~~]18[~~]~~ hours (3)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify the CST level is ≥ [~~110,000~~ 219000 gal]. (1)	12 hours

B 3.7 PLANT SYSTEMS

B 3.7.6 Condensate Storage Tank (CST)

BASES

BACKGROUND 4

The CST provides a preferred ~~safety grade~~ source of water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS). The CST provides a passive flow of water, by gravity, to the Auxiliary Feedwater (AFW) System (LCO 3.7.5, "Auxiliary Feedwater (AFW) System"). The steam produced is released to the atmosphere by the main steam safety valves or the atmospheric dump valves. The AFW pumps operate with a continuous recirculation to the CST.

When the main steam isolation valves are open, the preferred dump means of heat removal is to discharge steam to the condenser by the non-safety-grade path of the steam ~~bypass~~ valves. The condensed steam is returned to the CST by ~~the condensate transfer pump~~ Condenser level control valves. This has the advantage of conserving condensate while minimizing releases to the environment. 2 2

Because the CST is ~~a principal component in removing residual heat from the RCS, it is~~ not designed to withstand earthquakes and other natural phenomena, including missiles that might be generated by natural phenomena. ~~The CST is designed to Seismic Category I to ensure availability of the feedwater supply.~~ Feedwater is also available from ~~alternate sources.~~ the (ERCW) system as the safety grade water source. 4 4

Essential Raw Cooling Water

3 A description of the CST is found in the FSAR, Section 9.2.6 (Ref. 1).

APPLICABLE SAFETY ANALYSES 4

The CST provides the preferred cooling water to remove decay heat and to cool down the unit following all events in the accident analysis as discussed in the FSAR, Chapters [6] and [15] (Refs. 2 and 3, respectively). However, the ERCW system provides the safety grade water source to meet a DBA should the CST become unavailable. For anticipated operational occurrences and accidents that do not affect the OPERABILITY of the steam generators, the analysis assumption is generally 2 hours ~~30 minutes~~ at MODE 3, steaming through the MSSVs, followed by a cooldown to residual heat removal (RHR) entry conditions at the design cooldown rate. 4 1

The limiting event for the condensate volume is the large feedwater line break coincident with a loss of offsite

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

power. Single failures that also affect this event include the following:

a. Failure of the diesel generator powering the motor-driven AFW pump to the unaffected steam generator (requiring additional steam to drive the remaining AFW pump turbine); and

b. Failure of the steam-driven AFW pump (requiring a longer time for cooldown using only one motor-driven AFW pump).

These are not usually the limiting failures in terms of consequences for these events. (1)

A nonlimiting event considered in CST inventory determinations is a break in either the main feedwater bypass line or AFW line near where the two join. This break has the potential for dumping condensate until terminated by

(5) operator action~~, since the Emergency Feedwater Actuation System would not detect a difference in pressure between the steam generators for this break location~~. This loss of condensate inventory is partially compensated for by the retention of steam generator inventory.

(4) The CST does not satisfy any ~~satisfies~~ Criterion 3 of the NRC Policy Statement.

LCO (4)

[As the preferred water source to ~~To~~ satisfy accident analysis assumptions, the CST must contain sufficient cooling water to remove decay heat for

(1) 2 hours ~~[30 minutes]~~ following a reactor trip from 102% RTP, and then to cooldown the RCS to RHR entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. In doing this, it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during cooldown, as well as account for any losses from the steam-driven AFW pump turbine, or before isolating AFW to a broken line.

(1) 210,000 The CST level required is equivalent to a usable volume of ≥ [~~110,000 gallons~~], which is based on holding the unit in

(3) MODE 3 for 12 hours, followed by a cooldown to RHR entry conditions at ~~[75]~~ 50 °F/hour. This basis is established in Reference 4 and exceeds the volume required by the accident analysis. (1)

(continued)

BASES

LCO (continued)

The OPERABILITY of the CST is determined by maintaining the tank level at or above the minimum required level.

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the CST is required to be OPERABLE.

In MODE 5 or 6, the CST is not required because the AFW System is not required.

ACTIONS

A.1 and A.2

6

If the CST level is not within limits, the OPERABILITY of the backup supply should be verified by administrative means within 4 hours and once every 12 hours thereafter. OPERABILITY of the backup feedwater supply must include verification that the flow paths from the backup water supply to the AFW pumps are OPERABLE, and that the backup supply has the required volume of water available. The CST must be restored to OPERABLE status within 7 days, because the backup supply may be performing this function in addition to its normal functions. The 4-hour Completion Time is reasonable, based on operating experience to verify the OPERABILITY of the backup water supply. The 7-day Completion Time is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event occurring during this time period requiring the CST.

B.1 and B.2

3

A

If the CST cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on the steam generator for heat removal, within 18 hours. Required Action B.2 is modified by a Note indicating that entry into MODE 4 is only required if one or more RHR loops are OPERABLE. The Note also is intended to convey the suspension of further action to reach MODE 4 if, while in Required Action B.2, all RHR loops became inoperable. The

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full-power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

S

This SR verifies that the CST contains the required volume of cooling water. ~~(The required CST volume may be single value or a function of RCS conditions.)~~ The 12-hour Frequency is based on operating experience and the need for operator awareness of unit evolutions that may affect the CST inventory between checks. Also, the 12-hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal deviations in the CST level.

REFERENCES

7

1. Watts Bar FSAR, Section [9.2.6], "Condensate Storage Facilities."
2. Watts Bar FSAR, Chapter [6], "Engineered Safety Features."
3. Watts Bar FSAR, Chapter [15], "Accident Analysis."
4. TVA Calculation HCG-LCS-043085, "Minimum CST Water Level Required to Support the AFW System."

JUSTIFICATIONS FOR DEVIATIONS FROM STS 3.7.6

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

1. Change to reflect Watts Bar specific information or parameter values.

2. Change to specify Watts Bar specific terminology for clarity.

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

4. Multiple changes to this specification bases are required to reflect that the CST is not safety-related at Watts Bar. The CST provides a preferred water source, however the ERCW system (LCO 3.7.8) is the safety grade water supply.

5. Change to delete information that does not apply to Watts Bar.

6. The backup water supply is the ERCW system which pumps from the Tennessee River which would be a more or less infinite volume.

7. Watts Bar prefers to use the expanded reference information and format.

3.7 PLANT SYSTEMS

3.7.7 Component Cooling ~~Water (CCW)~~ System (CCS) (1)

LCO 3.7.7 Two CC~~W~~S trains shall be OPERABLE. (1)

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
(1) A. One CC~~W~~S train inoperable.	------------NOTE------------ Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops – MODE 4," for residual heat removal loops made inoperable by CC~~W~~S. ---------------------------- A.1 Restore CC~~W~~S train to OPERABLE status.	(A) (1) (1) 72 hours
B. Required Action and associated Completion Time of Condition A not met. ~~OR~~ (1) ~~Two CC~~W~~S trains inoperable.~~	B.1 Be in MODE 3. ~~AND~~ ~~B.2 Be in MODE 4.~~ AND	6 hours ~~12 hours~~ (A)
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.~~3~~ 2 ~~--------NOTE-------- Only required if CCW S has sufficient cooling capability to reach and maintain MODE 5.~~ ① (A) Be in MODE 5.	36 hours ~~from discovery of CCW S capability adequate to reach and maintain MODE 5~~ ①

NOTE
Isolation of flow to individual components does not render the CCS system inoperable (A)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
① SR 3.7.7.1	Verify each CCS manual, power-operated, and automatic valve in the flow path servicing safety-related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
① SR 3.7.7.2	Verify each CCS automatic valve in the flow path actuates to the correct position on an actual or simulated actuation signal.	~~[18]~~ 24 months ②
① SR 3.7.7.3	Verify each CCS pump starts automatically on an actual or simulated actuation signal.	~~[18]~~ 24 months ②

INSERT B ③

INSERT B

SR 3.7.7.4	Verify that the alternate feeder breaker to the C-S pump is open.	7 days

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling ~~Water (CCW)~~ System (CCS) 1

BASES

BACKGROUND 1

The CC~~W~~S ~~System~~ provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CC~~W~~S ~~System~~ also provides this function for various nonessential components, as well as the spent fuel storage pool. The CC~~W~~S ~~System~~ serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the ~~Service~~ Essential Raw Cooling Water System, and thus to the environment. 1 1

INSERT C 1

~~A typical CCW System~~ CCS ~~is arranged as two independent, full-capacity cooling loops, and has isolatable nonsafety related components. Each safety related train includes a full capacity pump, surge tank, heat exchanger, piping, valves, and instrumentation.~~ Each safety related train is powered from a separate bus. An open surge tank in the system provides pump trip protective functions to ensure that sufficient net positive suction head is available. The pump in each train is automatically started on receipt of a safety injection (SI) signal, and all nonessential components are isolated.

Additional information on the design and operation of the system, along with a list of the components served, is presented in the FSAR, Section [9.2.2] (Ref. 1). The principal safety-related function of the ~~CCW System~~ CCS ~~function~~ is the removal of decay heat from the reactor via the Residual Heat Removal (RHR) System. This may be during a normal or post-accident cooldown and shutdown. 2 1

APPLICABLE SAFETY ANALYSES

The design basis of the ~~CCW System~~ CCS is for one ~~CCW~~ CCS train to remove the post loss-of-coolant accident (LOCA) heat load from the containment sump during the recirculation phase, with a maximum CC~~W~~S temperature of [120]°F (Ref. 2). The Emergency Core Cooling System (ECCS) LOCA and containment OPERABILITY LOCA, each, model the maximum and minimum performance of the ~~CCW System~~ CCS, respectively. The normal temperature of the CC~~W~~S is ~~[80]~~ 95°F and, during unit cooldown to MODE 5 (T_{cold} < [200]°F), a maximum temperature of [95]°F ~~110~~ 120 is 1 1 2 1 2

(continued)

INSERT C

The CCS is arranged as two independent full-capacity cooling trains. Separate Train A equipment is provided in each unit, whereas Train B is shared by both units. Train A in unit 1 is served by CCS Hx A and CCS pump 1A-A. Pump 1B-B, which is actually Train B equipment, is also normally aligned to the Train A header in unit 1. However, pump 1B-B can be realigned to Train B on loss of Train A.

Similarly, Train A in unit 2 is served by CCS Hx B and CCS pump 2A-A with support from pump 2B-B.

Train B in both units is served by CCS Hx C. Normally, only CCS pump C-S is aligned to the Train B headers since few nonessential, normally-operating loads are assigned to Train B. However, pumps 1B-B and 2B-B can be realigned to the Train B headers on a loss of the C-S pump.

BASES

The CCS design based on these values, bounds the post-accident conditions such that the sump fluid will not increase in temperature after alignment of the RHR heat exchangers

APPLICABLE SAFETY ANALYSES (continued)

4 assumed. ~~This prevents the containment sump fluid from increasing in temperature~~ during the recirculation phase following a LOCA, and provides a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System (RCS) by the ~~SI~~ ECCS pumps.

1 The ~~CCW System~~ CCS is designed to perform its function with a single failure of any active component, assuming a loss of offsite power.

1 2 1 The ~~CCW System~~ CCS also functions to cool the unit from RHR entry conditions (T_{cold} < ~~[~~350~~]~~°F), to MODE 5 (T_{cold} < ~~[~~200~~]~~°F) during normal and post-accident operations. The time required to cool from ~~[~~350~~]~~ to ~~[~~200~~]~~°F is a function of the number of ~~CCW~~ CCS and RHR trains operating. 1 One ~~CCW~~ CCS train is sufficient to remove decay heat during subsequent operations with T_{cold} < ~~[~~200~~]~~°F. This assumes a 2 1 maximum ~~service water~~ ERCW temperature of ~~[95]~~ 85°F occurring 2 simultaneously with the maximum heat loads on the system.

1 The ~~CCW System~~ CCS satisfies Criterion 3 of the NRC Policy Statement.

LCO

1 The ~~CCW~~ CCS trains are independent of each other to the degree that each has separate controls and power supplies, and the operation of one does not depend on the other. 1 In the event of a DBA, one ~~CCW~~ CCS train is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two trains of ~~CCW~~ CCS must be OPERABLE. 1 At least one ~~CCW~~ CCS train will operate assuming the worst-case single active failure occurs coincident with a loss of offsite power.

1 A ~~CCW~~ CCS train is considered OPERABLE when:

a. The pump and associated surge tank are OPERABLE; and

b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety-related function are OPERABLE.

1 The isolation of ~~CCW~~ CCS from other components or systems not required for safety may render those components or system

(continued)

BASES

LCO (continued) (1) inoperable, but does not affect the OPERABILITY of the CC~~W~~S System.

APPLICABILITY (1) In MODES 1, 2, 3, and 4, the ~~CCW System~~ CCS is a normally operating system, which must be prepared to perform its post-accident safety functions, primarily RCS heat removal, which is achieved by cooling the RHR heat exchanger.

(1) In MODE 5 or 6, the OPERABILITY requirements of the ~~CCW System~~ CCS are determined by the systems it supports.

ACTIONS

A.1

Required Action A.1 and Required Actions is modified by a Note indicating that the applicable Conditions of LCO 3.4.6, "RCS Loops – MODE 4," be entered if an inoperable CC~~W~~S train results in an inoperable RHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components. (1)

(1) If one CC~~W~~S train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE CC~~W~~S train is adequate to perform the heat removal function. The 72-hour Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this period.

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

(1) If the CC~~W~~S train cannot be restored to OPERABLE status within the associated Completion Time, ~~or two CCWS trains are inoperable,~~ (A) the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, ~~in MODE 4 within 12 hours,~~ (A) (5) and in MODE 5 within 36 hours. ~~Required Action B.3 is modified by a Note indicating that completion of the Action is required only if CCWS has sufficient cooling capability to reach and maintain MODE 5.~~ (A) (1)

~~With both trains inoperable, immediate action must be taken to restore at least one train to OPERABLE status. In this~~

(continued)

BASES

ACTIONS

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

~~case, there is no heat sink for the RHR System, thus one CCW~~ S ~~train must be restored to OPERABLE status and the unit should be maintained in MODE 4, where decay heat can be removed by the steam generators.~~ 1

~~With both trains inoperable, flexibility (and abnormal operating procedures) are left to the operator to manage the situation. This allows remaining in MODE 4 with an alternate means of heat removal. This action allows total loss of function without entry into MODE 5 in accordance with LCO 3.0.3, which may not be possible with two CCW~~ S ~~trains inoperable. When a CCW~~ S ~~train is OPERABLE, the unit should then be placed in MODE 5; otherwise the unit should be maintained in MODE 4 until one CCW~~ S ~~train has sufficient cooling capability to reach and maintain MODE 5. In this case, LCO 3.0.3 is not applicable, since the unit cannot be brought to MODE 5 without at least one CCW~~ S ~~train OPERABLE. Adequate heat removal can be maintained using the steam generators and natural circulation.~~ 1 1

A

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full-power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

Verifying the correct alignment for manual, power-operated, and automatic valves in the CC~~W~~S flow path provides assurance that the proper flow paths exist for CC~~W~~S operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. 1

A

¶ This SR is modified by a Note indicating that the isolation of the CCS flow to individual components may render these components inoperable, but does not affect the OPERABILITY of the CCS.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1 (continued)

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

SR 3.7.7.2

This SR verifies proper automatic operation of the CC~~W~~S valves on an actual or simulated actuation signal. The CC~~W~~S System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. The ~~[~~18~~]~~-month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed on the ~~[~~18~~]~~-month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint. 1

2

2

SR 3.7.7.3

This SR verifies proper automatic operation of the CC~~W~~S pumps on an actual or simulated actuation signal. The ~~CCW System~~ CCS is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. The ~~[~~18~~]~~-month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed on the ~~[~~18~~]~~-month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint. 1

2

2

3 INSERT D

REFERENCES 6

1. Watts Bar FSAR, Section ~~[~~9.2.2~~]~~, "Component Cooling System"
2. FSAR, Section [6.2].

INSERT D

SR 3.7.7.4

This SR verifies that the C-S pump is powered from the normal power source when it is aligned for OPERABLE status. Verification of the correct power alignment ensures that the two CCS trains remain independent. The 7-day Frequency is based on engineering judgment, is consistent with procedural controls governing breaker operation, and ensures correct breaker position.

JUSTIFICATIONS FOR DEVIATIONS FROM STS 3.7.7

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

1. Change to specify Watts Bar specific terminology for clarity.

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

3. Added surveillance and bases to verify alternate feeder breaker position for the C-S pump to address SER 8.3.1.7 issue. This was noted in TVA letter dated June 21, 1985 in lieu of having alarms for alternate feeder breakers.

4. Change to correct error in the STS. Correct terminology is ECCS pumps. Many Westinghouse plants have specific intermediate head pumps called SI pumps. This statement is clearly referring to any ECCS pump, not just the SI pump.

5. Correct editorial error in the STS.

6. Watts Bar prefers to use the expanded reference information and format.

3.7 PLANT SYSTEMS

3.7.8 Essential Raw Cooling ~~Service~~ Water System ~~(SWS)~~ (ERCW) 1

LCO 3.7.8 Two ERCW ~~SWS~~ trains shall be OPERABLE. 1

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ERCW ~~SWS~~ train inoperable. 1 1 1	------------NOTES------------ 1. Enter applicable conditions of LCO 3.8.1, "AC Sources – Operating," for emergency diesel generator made inoperable by ~~SWS.~~ ERCW 2. Enter applicable conditions of LCO 3.4.6, "RCS Loops – MODE 4," for residual heat removal loops made inoperable by ~~SWS.~~ ERCW ----------------------------------- A.1 Restore ERCW ~~SWS~~ train to OPERABLE status.	and Required Actions A 72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. ~~OR~~ 1 ~~Two ERCW ~~SWS~~ trains inoperable~~	B.1 Be in MODE 3. ~~AND~~ ~~B.2 Be in MODE 4.~~ e AND B.3 ~~--------NOTE--------- Only required if ~~SWS~~ ERCW has sufficient cooling capability to reach and maintain MODE 5. ---------------------~~ e Be in MODE 5.	6 hours ~~12 hours~~ A 36 hours ~~from discovery of ~~SWS~~ ERCW capacity adequate to reach and maintain MODE 5~~ e A

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.8.1 1	-------------------NOTE------------------- Isolation of flow to individual components does not render the ~~SWS~~ ERCW inoperable. -- Verify each ~~SWS~~ ERCW manual, power-operated, and automatic valve in the flow path servicing safety-related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
1 SR 3.7.8.2	Verify each ~~SWS~~ ERCW automatic valve in the flow path actuates to the correct position on an actual or simulated actuation signal.	[18] months 2
1 SR 3.7.8.3	Verify each ~~SWS~~ ERCW pump starts automatically on an actual or simulated actuation signal.	[18] months 2

B 3.7 PLANT SYSTEMS

B 3.7.8 ~~Service Water System (SWS)~~ Essential Raw Cooling Water (ERCW) System (1)

BASES

BACKGROUND (1)

The ~~SWS~~ ERCW System provides a heat sink for the removal of process and operating heat from safety-related components during a Design Basis Accident (DBA) or transient. During normal operation, and a normal shutdown, the ~~SWS~~ ERCW System also provides this function for various safety related and non-safety-related components. The safety-related function is covered by this LCO.

INSERT E (1)

~~The SWS consists of two separate, 100%-capacity, safety-related, cooling water trains. Each train consists of two 100%-capacity pumps, one component cooling water (CCW) heat exchanger, piping, valving, instrumentation, and two cyclone separators. The pumps and valves are remote and manually aligned, except in the unlikely event of a loss-of-coolant accident (LOCA). The pumps aligned to the critical loops are automatically started upon receipt of a safety injection (SI) signal, and all essential valves are aligned to their post-accident positions. The SWS also provides emergency makeup to the spent fuel pool and CCW System [and is the backup water supply to the Auxiliary Feedwater System].~~

(1) Additional information about the design and operation of the ~~SWS~~ ERCW system, along with a list of the components served, is presented in the FSAR, Section ~~[9.2.1]~~ 9.2.1 (Ref. 1). The principal safety-related function of the ~~SWS~~ ERCW System is the removal of decay heat from the reactor via the ~~CCW System~~ CCS. (2) (1)

APPLICABLE SAFETY ANALYSES

The design basis of the ~~SWS~~ ERCW System is for one ~~SWS~~ ERCW train, in conjunction with the ~~CCW System~~ CCS and a 100% capacity ~~containment cooling system~~ Containment Spray System and residual heat removal system, to remove core decay heat following a design basis LOCA as discussed in the FSAR, Section ~~[6.2]~~ 9.2.1 (Ref. ~~2~~ 1). This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA and provides for a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System by the ~~ECCS~~ ECCS pumps. The ~~SWS~~ ERCW System is designed to perform its function with a single (1) (2) (3)

(continued)

INSERT E

The shared ERCW system consists of eight 50% ERCW pumps, four traveling water screens, four screen wash pumps, four strainers, associated piping, valves, and instrumentation.

Water for the ERCW system enters two separate sump areas of the pumping station through four traveling water screens, two for each sump. Four ERCW pumping units, all on the same plant train, take suction from one of the sumps, and four more on the opposite plant train take suction from the other sump. One set of pumps and associated equipment is designated Train A, and the other Train B. These trains are redundant and are normally maintained separate and independent of each other. Each set of four pumps discharges into a common manifold, from which two separate headers (1A and 2A for Train A, and 1B and 2B for Train B) each with its own automatic backwashing strainer, supply water to the various system users. Two pumps per train are adequate to supply worst case conditions. Two pumps per train are aligned to receive power from different diesel generators. Operator designated pumps and valves are remote and manually aligned, except in the unlikely event of a loss-of-coolant accident (LOCA). The pumps are automatically started upon receipt of a safety injection (SI) signal, and all essential valves are aligned to their post-accident positions. The ERCW system also provides emergency makeup to the ~~Spent Fuel Pool and~~ ~~CCS System~~ and is the backup water supply to the Auxiliary Feedwater System.

Component Cooling System (CCS)

BASES

APPLICABLE SAFETY ANALYSES (continued)

failure of any active component, assuming the loss of offsite power.

1 The ~~SWS~~ ERCW system, in conjunction with the ~~CCW System~~ CCS, also cools the unit from residual heat removal, as discussed in the FSAR, Section ~~[5.4.7]~~ 5.5.7 (Ref. ~~3~~ 2) entry conditions to MODE 5 during normal and post-accident operations. The time required for this evolution is a function of the number of ~~CCW~~ CCS and RHR System trains that are operating. One ~~SWS~~ ERCW train is sufficient to remove decay heat during subsequent operations in MODES 5 and 6. This assumes a maximum ~~SWS~~ ERCW temperature of ~~[95]~~ 85°F occurring simultaneously with maximum heat loads on the system. 2 1

1 The ~~SWS~~ ERCW system satisfies Criterion 3 of the NRC Policy Statement.

LCO

1 Two ~~SWS~~ ERCW trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post-accident heat loads, assuming that the worst-case, single active failure occurs coincident with the loss of offsite power.

1 An ~~SWS~~ ERCW train is considered OPERABLE during MODES 1, 2, 3, and 4 when:

1
a. ~~The pump is~~ Two pumps are OPERABLE; and

b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety-related function are OPERABLE.

APPLICABILITY

1 In MODES 1, 2, 3, and 4, the ~~SWS~~ ERCW system is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the ~~SWS~~ ERCW system and required to be OPERABLE in these MODES.

1 In MODES 5 and 6, the OPERABILITY requirements of the ~~SWS~~ ERCW system are determined by the systems it supports.

(continued)

BASES (continued)

ACTIONS

A.1

ERCW ERCW

① If one ~~SWS~~ train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE ~~SWS~~ train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE ~~SWS~~ train could result in loss of ~~SWS~~ function. Required Action A.1 is modified by two Notes. The first Note indicates that the applicable Conditions of LCO 3.8.1, "AC Sources - Operating," be entered if an inoperable ~~SWS~~ train results in an inoperable emergency diesel generator. The second Note indicates that the applicable Conditions of LCO 3.4.6, "RCS Loops - MODE 4," be entered if an inoperable ~~SWS~~ train results in an inoperable ~~decay~~ heat removal train. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components. The 72-hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this time period.

① ERCW system ERCW ①

and Required Actions

Ⓐ

residual ①

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

ERCW ERCW Ⓐ

If the ~~SWS~~ train cannot be restored to OPERABLE status within the associated Completion Time, ~~or two SWS trains are inoperable,~~ the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, ~~in MODE 4 within 12 hours,~~ and in MODE 5 within 36 hours. ~~Required Action B.3 is modified by a Note precluding the requirement to place the unit in MODE 5, unless the SWS has sufficient cooling capability to support all heat loads necessary to reach and maintain MODE 5. In this case, the unit should be maintained in MODE 4 until this capability is restored.~~

Ⓐ ERCW System ①

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full-power conditions in an orderly manner and without challenging unit systems.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR is modified by a Note indicating that the isolation of the SWS components or systems may render those components inoperable, but does not affect the OPERABILITY of the SWS.

ERCW system

Verifying the correct alignment for manual, power-operated, and automatic valves in the ~~SWS~~ flow path provides assurance that the proper flow paths exist for ~~SWS~~ operation. This SR 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 SR 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 SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. 1

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

SR 3.7.8.2

This SR verifies proper automatic operation of the ~~SWS~~ ERCW system valves on an actual or simulated actuation signal. The ~~SWS~~ is a normally operating system that cannot be fully actuated as part of normal testing. The ~~[~~18~~]~~-month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed on the ~~[~~18~~]~~-month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint. 1 2 2

SR 3.7.8.3

ERCW

This SR verifies proper automatic operation of the ~~SWS~~ pumps on an actual or simulated actuation signal. The ~~SWS~~ ERCW System is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. The ~~[~~18~~]~~-month Frequency is based on the need to perform this 1 2

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.3 (continued)

Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed on the [18]-month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

2

REFERENCES

4

1. Watts Bar FSAR, Section [9.2.1], "Essential Raw Cooling Water."

~~2. FSAR, Section [6.2].~~

2 ~~3~~. Watts Bar FSAR, Section ~~[5.4.7]~~ 5.5.7, "Residual Heat Removal System."

JUSTIFICATIONS FOR DEVIATIONS FROM STS 3.7.8

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

1. Change to specify Watts Bar specific terminology for clarity.

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

3. Change to correct error in the STS. Correct terminology is ECCS pumps. Many Westinghouse plants have specific intermediate head pumps called SI pumps. This statement is clearly referring to any ECCS pump, not just the SI pump.

4. Watts Bar prefers to use the expanded reference information and format.

3.7 PLANT SYSTEMS

3.7.9 Ultimate Heat Sink (UHS)

LCO 3.7.9 The UHS shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
1 ~~[A. One or more cooling tower with one cooling tower fan inoperable.~~	~~A.1 Restore fan(s) to OPERABLE status.~~	~~7 days]~~
~~B. Required Action and associated Completion Time of Condition A not met.~~ ~~OR~~ A. UHS inoperable ~~[for reasons other than Condition A]~~. 1	A ~~B~~.1 Be in MODE 3. ~~AND~~ A ~~B.2 Be in MODE 4.~~ AND A ~~B.3~~ 2 ~~-----NOTE----- Only required if UHS has sufficient cooling capability to reach and maintain MODE 5. -----~~ Be in MODE 5.	6 hours ~~[12] hours~~ 2 A A e 36 hours ~~from discovery of adequate UHS capability adequate to reach and maintain MODE 5~~ A e 3

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
4	[SR 3.7.9.1 Verify water level of UHS is ≥ [562] ft [mean sea level].	[24] hours]
2	SR 3.7.9.~~2~~ 1 Verify average water temperature of UHS is ≤ [~~90~~]°F. 85	24 hours
1	[SR 3.7.9.3 Verify the operation of each cooling tower fan for ≥ [15] minutes.	31 days]

B 3.7 PLANT SYSTEMS

B 3.7.9 Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The UHS provides a heat sink for processing and operating heat from safety related components during a transient or accident, as well as during normal operation. This is done by utilizing the ~~Service Water System (SWS)~~ and the Component Cooling ~~Water (CCW)~~ System.

5

Essential Raw Cooling Water (ERCW) System

(CCS)

~~The UHS has been defined~~ as that complex of water sources, including necessary retaining structures (e.g., a pond with its dam, or a river with its dam), and the canals or conduits connecting the sources with, but not including, the cooling water system intake structures as discussed in the FSAR, Section [9.2.5] (Ref. 1). If cooling towers or portions thereof are required to accomplish the UHS safety functions, they should meet the same requirements as the sink. The two principal functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident.

INSERT F

1

A variety of complexes is used to meet the requirements for an UHS. A lake or an ocean may qualify as a single source. If the complex includes a water source contained by a structure, it is likely that a second source will be required.

The basic performance requirements are that a 30-day supply of water be available, and that the design basis temperatures of safety-related equipment are not exceeded. Basins of cooling towers generally include less than a 30-day supply of water, typically 7 days or less. A 30-day supply would be dependent on other source(s) and makeup system(s) for replenishing the source in the cooling tower basin. For smaller basin sources, which may be as small as a 1-day supply, the systems for replenishing the basin and the backup source(s) become of sufficient importance that the makeup system itself may be required to meet the same design criteria as an engineered safety feature (e.g., single failure considerations), and multiple makeup water sources may be required.

(continued)

INSERT F

The UHS is defined as the Tennessee River, including the TVA controlled dams upstream of the intake structure, Chicamauga Dam (the nearest downstream dam), and the plant intake channel, not including the intake structure, as discussed in FSAR Section 9.2.5 (Ref. 1). The maximum UHS temperature of 85 °F ensures adequate heat load removal capacity for a minimum of 30 days after reactor shutdown or a shutdown following an accident, including a Loss of Coolant Accident (LOCA).

BASES

BACKGROUND (continued)

Additional information on the design and operation of the system, along with a list of components served, can be found in Reference 1.

6

~~If the UHS does not meet its design limits of water temperature, water level, or number of OPERABLE cooling tower fans, the UHS may not have sufficient capacity to bring the unit to a safe, controlled shutdown during a Design Basis Accident (DBA) from full power, but may be able to support unit operation at a reduced power level.~~

APPLICABLE SAFETY ANALYSES

1

The UHS is the sink for heat removed from the reactor core following all accidents and anticipated operational occurrences in which the unit is cooled down and placed on residual heat removal (RHR) operation. ~~For units that use UHS as the normal heat sink for condenser cooling via the Circulating Water System, unit operation at full power is its maximum heat load.~~ Its maximum post-accident heat load occurs approximately 20 minutes after a design basis loss-of-coolant accident (LOCA). Near this time, the unit switches from injection to recirculation and the containment cooling systems and RHR are required to remove the core decay heat.

The operating limits are based on conservative heat transfer analyses for the worst-case LOCA. Reference 1 provides the details of the assumptions used in the analysis, which include worst-expected meteorological conditions, conservative uncertainties when calculating decay heat, and worst-case single active failure (e.g., single failure of a man-made structure). The UHS is designed in accordance with Regulatory Guide 1.27 (Ref. 2), which requires a 30-day supply of cooling water in the UHS.

The UHS satisfies Criterion 3 of the NRC Policy Statement.

LCO

4

ERCW system

The UHS is required to be OPERABLE and is considered OPERABLE if it contains ~~a sufficient volume of~~ water at or below the maximum temperature that would allow the ~~SWS~~ to operate for at least 30 days following the design basis LOCA without the loss of net positive suction head (NPSH), and without exceeding the maximum design temperature of the equipment served by the ~~SWS~~. To meet this condition, the 5 5

ERCW system.

(continued)

BASES

LCO (continued) 2 4

UHS temperature should not exceed 85 ~~[90°F]~~, and the level ~~should not fall below [562 ft mean sea level] during normal unit operation.~~

APPLICABILITY

In MODES 1, 2, 3, and 4, the UHS is required to support the OPERABILITY of the equipment serviced by the UHS and required to be OPERABLE in these MODES.

In MODE 5 or 6, the OPERABILITY requirements of the UHS are determined by the systems it supports.

ACTIONS

1

~~[A.1~~

~~If one or more cooling towers have one fan inoperable (i.e., up to one fan per cooling tower inoperable), action must be taken to restore the inoperable cooling tower fan(s) to OPERABLE status within [ ] days.~~

~~The 7-day Completion Time is reasonable based on the low probability of an accident occurring during the 7 days that one cooling tower fan is inoperable (in one or more cooling towers), the number of available systems, and the required to reasonably complete the Required Action.]~~

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

1

~~[If the cooling tower fan cannot be restored to OPERABLE status within the associated Completion Time, or]~~ if the UHS is inoperable ~~for reasons other than Condition A~~, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, ~~in MODE 4 within [12] hours,~~ and in MODE 5 within 36 hours. ~~Required Action B.3 is modified by a Note precluding the requirement to place the unit in MODE 5, unless the UHS has sufficient cooling capability to support all heat loads necessary to reach and maintain MODE 5. In this case, the unit should be maintained in MODE 4 until this capability is restored.~~ A 2 1 e

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions

(continued)

BASES

ACTIONS ① A~~B~~.1, A~~B~~.2 2, and A~~B~~.~~3~~ 2 (continued)

from full-power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS ④

~~SR 3.7.9.1~~

~~This SR verifies that adequate long-term (30-day) cooling can be maintained. The specified level also ensures that sufficient NPSH is available to operate the SWS pumps. The [24]-hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the UHS water level is ≥ [562] ft [mean sea level].~~

② e [SR 3.7.9.~~2~~ 1

ERCW system ⑤

This SR verifies that the ~~SWS~~ is available to cool the CC~~W~~ S System to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a DBA. The 24-hour Frequency is based on industry operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the average water temperature of the UHS is ≤ ~~[90]~~ 85°F~~]~~.] e

④

~~SR 3.7.9.3~~

~~Operating each cooling tower fan for ≥ [15] minutes ensures that all fans are OPERABLE and that all associated controls are functioning properly. It also ensures that fan or motor failure, or excessive vibration, can be detected for corrective action. The 31-day Frequency is based on operating experience, the known reliability of the fan units, the redundancy available, and the low probability of significant degradation of the UHS cooling tower fans occurring between surveillances.~~

(continued)

BASES (continued)

REFERENCES 7	1. Watts Bar FSAR, Section 9.2.5, "Ultimate Heat Sink."
	2. Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants", Revision 1, March 1974.

JUSTIFICATIONS FOR DEVIATIONS FROM STS 3.7.9

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

1. The UHS for Watts Bar is the Tennessee River. Watts Bar does not use cooling tower fans. Extensive changes have been made to delete information not applicable to Watts Bar with our natural draft cooling towers and UHS.

2. Deleted brackets that identify plant specific information.

3. Delete the word "adequate" since it is redundant to "capable" and is inconsistent with similar completion times in LCO 3.7.7 and 3.7.8.

4. Watts Bar does not consider the loss of the UHS to be a credible event as discussed in FSAR section 9.2.5.

5. Change to reflect plant specific terminology.

6. Deleted this paragraph since it is confusing. If the UHS is inoperable, a shutdown is required by the action statements. No continued low power operation is permitted as implied by this paragraph. We believe this to be an error in the STS.

7. Watts Bar prefers to use the expanded reference information and format.

3.7 PLANT SYSTEMS

3.7.10 Control Room Emergency ~~Filtration~~ Ventilation System (CRE~~F~~VS) 1

LCO 3.7.10 Two CRE~~F~~VS trains shall be OPERABLE. 1

APPLICABILITY: MODES 1, 2, 3, 4, ~~[5, and 6,]~~ and
During movement of irradiated fuel assemblies. 2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CRE~~F~~VS train inoperable. 1	A.1 Restore CRE~~F~~VS train to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
2 C. Required Action and associated Completion Time of Condition A not met ~~[in MODE 5 or 6, or]~~ during movement of irradiated fuel assemblies.	~~-----NOTE----- Place in toxic gas protection mode if auto-swapover to toxic gas protection mode is inoperable. ----------~~ C.1 Place OPERABLE CRE~~F~~VS train in emergency mode. OR	3 Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2.1 Suspend CORE ALTERATIONS. AND C.2.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately
1 D. Two CREFS trains inoperable in MODE 1, 2, 3, or 4.	D.1 Enter LCO 3.0.3.	Immediately
1 2 E. Two CREFS trains inoperable ~~[in MODE 5 or 6, or]~~ during movement of irradiated fuel assemblies.	E.1 Suspend CORE ALTERATIONS. AND E.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.10.1 4 Verify the operation of each CREFS train for ~~[≥ 10 continuous hours with the heaters operating or (for systems without heaters)]~~ ≥ 15 minutes ~~]~~.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.10.2	Perform required CREFS filter testing in accordance with the [Ventilation Filter Testing Program (VFTP)].	In accordance with [VFTP]
SR 3.7.10.3	Verify each CREFS train actuates on an actual or simulated actuation signal(s).	[18] months
SR 3.7.10.4	Verify one CREFS train can maintain a positive pressure of ≥ [0.125] inches water gauge, relative to the ~~adjacent/[turbine building]~~ outside atmosphere during the pressurization mode of operation at a makeup flow rate of ≤ [~~3000~~] 325 cfm and a recirculation flow rate of 3675 ± 10% cfm.	[18] months on a STAGGERED TEST BASIS

B 3.7 PLANT SYSTEMS

B 3.7.10 Control Room Emergency ~~Filtration~~ Ventilation System (CREVS) 1

BASES

BACKGROUND 1 The CREVS provides a protected environment from which operators can control the unit following an uncontrolled 3 release of radioactivity~~(s), chemicals, or toxic gas(s)~~.

The CREFS consists of two independent, redundant trains that recirculate and filter the control room air. Each train 7 consists of a ~~prefilter or demister, a~~ high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system. ~~as well as demisters to remove water droplets from the air stream.~~ 7 ~~A second bank of HEPA filters follows the adsorber section to collect carbon fines, and provide backup in case of failure of the main HEPA filter bank.~~

1 Actuation of the CREVS occurs automatically upon receipt of a safety injection signal on either unit or upon indication of high radiation in the outside air supply.

The CREVS is an emergency system, parts of which ~~may~~ also operate during normal unit operations. ~~In the standby mode of operation.~~ ~~Upon receipt of the actuating signal(s), normal air supply to the control room is isolated, and the stream of ventilation air is recirculated through the system filter trains. The prefilters or demisters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers. Continuous operation of each train for at least 10 hours per month, with the heaters on, reduces moisture buildup on the HEPA filters and adsorbers. Both the demister and heater are important to the effectiveness of the charcoal adsorbers.~~ 7

~~Actuation of the CREVS places the system in either of two separate states (emergency radiation state, toxic gas isolation state) of the emergency mode of operation, depending on the initiation signal.~~ 7 Actuation of the system to the ~~emergency radiation state of the~~ emergency mode of operation, closes the unfiltered outside air intake and unfiltered exhaust dampers, and aligns the system for recirculation of the control room air through the redundant trains of 1 air handling ~~heating and air conditioning~~ units, with a portion of the stream of air directed through HEPA and the charcoal filters. The emergency mode 7 ~~radiation state~~ also initiates pressurization and filtered ventilation of the air supply to the control room.

(continued)

BASES

BACKGROUND (continued)

7 ~~Outside air is filtered, diluted with building air from the electrical equipment and cable spreading rooms, and added to the air being recirculated from the control room.~~ Pressurization of the control room prevents infiltration of unfiltered air from the surrounding areas of the building. 3 ~~The actions taken in the toxic-gas isolation state are the same, except that the signal switches control room ventilation to an isolation alignment to prevent outside air from entering the control room.~~

3 ~~The air entering the control room is continuously monitored by radiation and toxic-gas detectors. One detector output above the setpoint will cause actuation of the emergency radiation state or toxic-gas isolation state, as required. The actions of the toxic-gas isolation state are more restrictive, and will override the actions of the emergency radiation state.~~

5 A single train will pressurize the control room to about [0.125] inches water gauge, and provide an air exchange rate in excess of [25]% per hour. The CREFS operation in maintaining the control room habitable is discussed in the FSAR, Section [6.4] (Ref. 1).

Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across the other filter train. Normally open isolation dampers are arranged in series pairs so that the failure of one damper to shut will not result in a breach of isolation. The CREFS is designed in accordance with Seismic Category I requirements.

1 The CREFS is designed to maintain the control room environment for 30 days of continuous occupancy after a Design Basis Accident (DBA) without exceeding a 5-rem-whole-body dose.

APPLICABLE SAFETY ANALYSES

1 The CREFS components are arranged in redundant, safety-related ventilation trains. The location of components and ducting within the control room envelope ensures an adequate supply of filtered air to all areas requiring access. The CREFS provides airborne radiological protection for the control room operators, as demonstrated by the control room accident dose analyses for the most limiting design basis

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

loss-of-coolant accident (LOCA), fission-product release presented in the FSAR, ~~Chapter [15]~~ Section 15.5.3 (Ref. 2). 5

The analysis of toxic-gas releases demonstrates that the toxicity limits are not exceeded in the control room following a toxic chemical release, as presented in Reference 1.

1 The worst-case single active failure of a component of the CRES, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

The CRES satisfies Criterion 3 of the NRC Policy Statement.

LCO

1 Two independent and redundant CRES trains are required to be OPERABLE to ensure that at least one is available assuming a single failure disables the other train. Total system failure could result in exceeding a dose of 5 rem to the control room operator in the event of a large radioactive release.

1 The CRES is considered OPERABLE when the individual components necessary to limit operator exposure are OPERABLE in both trains. A CRES train is OPERABLE when the associated:

a. Fan is OPERABLE;

b. HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and

7 c. ~~Heater, demister,~~ ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

APPLICABILITY

In MODES 1, 2, 3, 4, ~~[5 and 6,]~~ and during movement of irradiated fuel assemblies, CRES must be OPERABLE to control operator exposure during and following a DBA. 2 1

(continued)

BASES

APPLICABILITY (continued)

In [MODE 5 or 6], the CREFS is required to cope with the release from the rupture of an outside waste-gas tank. (2)

During movement of irradiated fuel assemblies, the CREFS must be OPERABLE to cope with the release from a fuel-handling accident. (1)

ACTIONS

A.1

When one CREFS train is inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREFS train is adequate to perform the control room protection function. However, the overall reliability is reduced because a single failure in the OPERABLE CREFS train could result in loss of CREFS function. The 7-day Completion Time is based on the low probability of a DBA occurring during this time period, and that the remaining train can provide the required capability. (1) (1)

B.1 and B.2

In MODE 1, 2, 3, or 4, if the inoperable CREFS train cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full-power conditions in an orderly manner and without challenging unit systems. (1)

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

(2) (A) (8) ~~[In MODE 5 or 6, or]~~ during movement of irradiated fuel assemblies, if the inoperable CREFS train cannot be restored to OPERABLE status~~, and Required Action A.1 cannot be completed~~ within the required Completion Time, action must be taken to immediately place the OPERABLE CREFS train in the emergency mode. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected.

(continued)

BASES

ACTIONS

C.1, C.2.1, and C.2.2 (continued)

An alternative to Required Action C.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

3

~~Required Action C.1 is modified by a Note indicating to place the system in the toxic-gas protection mode if auto-swapover to toxic-gas protection mode is inoperable.~~

D.1 V V

1

If both CREOS trains are inoperable in MODE 1, 2, 3, or 4, the CREOS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

E.1 and E.2

which 9 A

2

~~[In MODE 5 or 6, or]~~ during movement of irradiated fuel assemblies with two CREOS trains inoperable, action must be taken immediately to suspend activities could result in a release of radioactivity that might enter the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

g A

~~This SR verifies that a train in a standby mode of operation starts on demand and continues to operate.~~ Standby systems should be checked periodically to ensure that they ~~start and~~ e function properly. As the environment and normal operating

A

9 Conditions on this system are not too

severe, testing each train once every month provides an adequate check of this system. Monthly heater operations dry out any moisture accumulated in the charcoal from humidity in the ambient air. ~~[Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be operated~~ 4

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1 (continued)

~~for ≥ 15 minutes to demonstrate the function of the system.~~ The 31-day Frequency is based on the reliability of the equipment and the two-train redundancy availability.

SR 3.7.10.2

This SR verifies that the required CREFS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. The CREFS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 3). The [VFTP] includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the [VFTP].

SR 3.7.10.3

This SR verifies that each CREFS train starts and operates on an actual or simulated actuation signal. The Frequency of [18] months is specified in Regulatory Guide 1.52 (Ref. 3).

SR 3.7.10.4

This SR verifies the integrity of the control room enclosure, and the assumed inleakage rates of the potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning ~~of the CREFS~~. During the emergency mode of operation, the CREFS is designed to pressurize the control room ≥ [0.125] inches water gauge positive pressure with respect to ~~adjacent areas~~ the outside atmosphere in order to prevent unfiltered inleakage. The CREFS is designed to maintain this positive pressure with one train at a makeup flow rate of ~~[3000] cfm~~ 325 cfm and a recirculation flow rate of 3675 ± 10% cfm. The Frequency of [18] months on a STAGGERED TEST BASIS is consistent with the guidance provided in NUREG-0800 (Ref. 4).

(continued)

BASES (continued)

REFERENCES

(10)

1. Watts Bar FSAR, Section ~~[6.4].~~ 6.4, "Habitability Systems."
2. FSAR, ~~Chapter [15].~~ Section 15.5.3, "~~Posta~~ Environmental Consequences of a Postulated Loss of Coolant Accident."
3. Regulatory Guide 1.52, Rev. 02.
4. NUREG-0800, Standard Review Plan, Section 6.4, "Control Room Habitability System", Rev. 2, July 1981.

JUSTIFICATIONS FOR DEVIATIONS FROM STS 3.7.10

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

1. Change to reflect plant specific terminology/design.

2. The CREVS is not required to cope with the release of a Waste Gas Decay Tank rupture.

3. Watts Bar no longer uses toxic gas detectors or the toxic gas protection mode as discussed in FSAR 6.4.3.

4. The CREVS air cleanup units do not have heaters.

5. Change to delete brackets that specify plant specific information and parameters.

6. The recirculation flow rate and makeup flow rate are both required to be verified to ensure proper system operation.

7. Change to remove information not applicable to Watts Bar design.

8. Watts Bar considers this phrase to be an error in the STS since it is redundant to the first phrases. "During movement of...if the CREVS cannot be restored..." is equivalent to Required Action A.1 not met. This is also inconsistent with the similar paragraph in B.1 and B.2.

9. Change to correct editorial error in the STS.

10. Watts Bar prefers to use the expanded reference information and format.

3.7 PLANT SYSTEMS

3.7.11 Control Room Emergency Air Temperature Control System (CREATCS)

LCO 3.7.11 Two CREATCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, ~~[5, and 6,]~~ and
During movement of irradiated fuel assemblies. 1

ACTIONS

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

(continued)

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.11.1 Verify each CREATCS train has the capability to remove the assumed heat load.	[18] months (2)

B 3.7 PLANT SYSTEMS

B 3.7.11 Control Room Emergency Air Temperature Control System (CREATCS)

BASES

BACKGROUND

The CREATCS provides temperature control for the control room following isolation of the control room.

The CREATCS consists of ~~two~~ 9 independent and redundant trains that provide cooling ~~and heating~~ of recirculated control room air. Each train consists of ~~heating coils, cooling coils~~ an air handling unit (AHU), water chiller, chilled-water pump, and associated piping, ductwork, instrumentation, and controls to provide for control room temperature control. The CREATCS is a subsystem providing air temperature control for the control room. (3)

The CREATCS is an emergency system, parts of which may also operate during normal unit operations. A single train will provide the required temperature control to maintain the control room between ~~[70] and [85]~~ 60 and 104°F. The CREATCS operation in maintaining the control room temperature is discussed in the FSAR, Section ~~[6.4]~~ 9.4.1 (Ref. 1). (2)

APPLICABLE SAFETY ANALYSES

The design basis of the CREATCS is to maintain the control room temperature for 30 days of continuous occupancy.

The CREATCS components are arranged in redundant, safety-related trains. During emergency operation, the CREATCS maintains the temperature between ~~[70] and [85]~~ 60 and 104°F. A single-active failure of a component of the CREATCS, with a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for control room temperature control. The CREATCS is designed in accordance with Seismic Category I requirements. The CREATCS is capable of removing sensible and latent-heat loads from the control room, which include consideration of equipment heat loads and personnel occupancy requirements, to ensure equipment OPERABILITY. (2)

The CREATCS satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

Two independent and redundant trains of the CREATCS are required to be OPERABLE to ensure that at least one is available, assuming a single failure disabling the other train. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident.

Chillers, AHUs

3

The CREATCS is considered to be OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in both trains. These components include the ~~heating and cooling coils~~ and associated temperature control instrumentation. In addition, the CREATCS must be operable to the extent that air circulation can be maintained.

APPLICABILITY

1

In MODES 1, 2, 3, 4, ~~[5, and 6]~~ and during movement of irradiated fuel assemblies, the CREATCS must be OPERABLE to ensure that the control room temperature will not exceed equipment operational requirements following isolation of the control room.

1

~~[In MODE 5 or 6,] CREATCS may not be required for those facilities that do not require automatic control room isolation.~~

ACTIONS

A.1

With one CREATCS train inoperable, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CREATCS train is adequate to maintain the control room temperature within limits. However, the overall reliability is reduced because a single failure in the OPERABLE CREATCS train could result in loss of CREATCS function. The 30-day Completion Time is based on the low probability of an event requiring control room isolation, the consideration that the remaining train can provide the required protection, and that alternate safety- or non-safety-related cooling means are available.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

In MODE 1, 2, 3, or 4, if the inoperable CREATCS train cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes the risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full-power conditions in an orderly manner and without challenging unit systems.

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

1 ~~[In MODE 5 or 6, or]~~ ~~d~~During movement of irradiated fuel, if the inoperable CREATCS train cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CREATCS train must be placed in operation immediately. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that active failures will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

D.1

If both CREATCS trains are inoperable in MODE 1, 2, 3, or 4 the control room CREATCS may not be capable of performing its intended function. Therefore, LCO 3.0.3 must be entered immediately.

E.1 and E.2

1 ~~[In MODE 5 or 6, or]~~ ~~d~~During movement of irradiated fuel assemblies, with two CREATCS trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition

(continued)

BASES

ACTIONS

E.1 and E.2 (continued)

that minimizes risk. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

This SR verifies that the heat removal capability of the system is sufficient to remove the assumed heat load in the control room. This SR consists of a combination of testing and calculations. The [18]-month Frequency is appropriate since significant degradation of the CREATCS is slow and is not expected over this time period. 2

REFERENCES

1. FSAR, Section ~~[6.4].~~ 9.4.1, "Control Room Area Ventilation System." 5

This is accomplished by verifying that the system has not degraded. The only measurable parameters ~~are~~ that could degrade undetected ~~by~~ during normal operation is the system air flow and chilled water flow rate. Verification of these two flow rates will provide assurance that the heat removal capacity of the system is still adequate. 4

JUSTIFICATIONS FOR DEVIATIONS FROM STS 3.7.11

1. There are no accidents which require the CREATCS system in MODES 5 or 6.
2. Change to delete brackets that specify plant specific information and parameters.
3. Change to reflect plant specific terminology/design.
4. Watts Bar has proposed this paragraph as a way of meeting the intent of the SR. The SR as currently written provides little guidance on what constitutes an acceptable verification of heat removal capability.
5. Watts Bar prefers to use the expanded reference information and format.

3.7 PLANT SYSTEMS

3.7.12 Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS)

LCO 3.7.12 Two ECCS PREACS trains shall be OPERABLE.

1

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ECCS PREACS train inoperable.	A.1 Restore ECCS PREACS train to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.12.1 Verify the operations of each ECCS PREACS train for [≥ 10 continuous hours with the heaters operating or (for systems without heaters) ≥ 15 minutes].	31 days

(continued)

(1)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.7.12.2	Perform required ECCS PREACS filter testing in accordance with the [Ventilation Filter Testing Program (VFTP)].	In accordance with the [VFTP]
SR 3.7.12.3	Verify each ECCS PREACS train actuates on an actual or simulated actuation signal.	[18] months
SR 3.7.12.4	Verify one ECCS PREACS train can maintain a pressure ≤ [-0.125] inches water gauge relative to atmospheric pressure during the [post-accident] mode of operation at a flow rate of ≤ [3000] cfm.	[18] months on a STAGGERED TEST BASIS
[SR 3.7.12.5	Verify each ECCS PREACS filter bypass damper can be closed.	[18] months]

B 3.7 PLANT SYSTEMS

B 3.7.12 Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS)

1

BASES

BACKGROUND

The ECCS PREACS filters air from the area of the active ECCS components during the recirculation phase of a loss-of-coolant accident (LOCA). The ECCS PREACS, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the ECCS pump room area and the lower reaches of the auxiliary building.

The ECCS PREACS consists of two independent and redundant trains. Each train consists of a heater, a prefilter or demister, a high efficiency particulate air (HEPA) filter, an activated charcoal absorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system, as well as demisters functioning to reduce the relative humidity of the air stream. A second bank of HEPA filters follows the absorber section to collect carbon fines and provide backup in case the main HEPA filter bank fails. The downstream HEPA filter is not credited in the accident analysis, but serves to collect charcoal fines, and to back up the upstream HEPA filter should it develop a leak. The system initiates filtered ventilation of the pump room following receipt of a safety injection (SI) signal.

The ECCS PREACS is a standby system, aligned to bypass the system HEPA filters and charcoal absorbers. During emergency operations, the ECCS PREACS dampers are realigned, and fans are started to begin filtration. Upon receipt of the actuating Engineered Safety Feature Actuation System signal(s), normal air discharges from the ECCS pump room isolate, and the stream of ventilation air discharges through the system filter trains. The prefilters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal absorbers.

The ECCS PREACS is discussed in the FSAR, Sections [6.5.1], [9.4.5], and [15.6.5] (Refs. 1, 2, and 3, respectively) since it may be used for normal, as well as post-accident, atmospheric cleanup functions. The primary purpose of the

(continued)

BASES

BACKGROUND (continued)

heaters is to maintain the relative humidity at an acceptable level, consistent with iodine removal efficiencies per Regulatory Guide 1.52 (Ref. 4).

1

APPLICABLE SAFETY ANALYSES

The design basis of the ECCS PREACS is established by the large-break LOCA. The system evaluation assumes a passive failure of the ECCS outside containment, such as a SI pump seal failure, during the recirculation mode. In such a case, the system limits radioactive release to within the 10 CFR 100 limits (Ref. 5), or the NRC staff-approved licensing basis (e.g., a specified fraction of Reference 5 limits). The analysis of the effects and consequences of a large-break LOCA is presented in Reference 3. The ECCS PREACS also actuates following a small-break LOCA, in those cases where the ECCS goes into the recirculation mode of long-term cooling, to clean up releases of smaller leaks, such as from valve stem packing.

Two types of system failures are considered in the accident analysis: complete loss of function, and excessive LEAKAGE. Either type of failure may result in a lower efficiency of removal for any gaseous and particulate activity released to the ECCS pump rooms following a LOCA.

The ECCS PREACS satisfies Criterion 3 of the NRC Policy Statement.

LCO

Two independent and redundant trains of the ECCS PREACS are required to be OPERABLE to ensure that at least one is available, assuming that a single failure disables the other train coincident with loss of offsite power. Total system failure could result in the atmospheric release from the ECCS pump room exceeding 10 CFR 100 limits in the event of a Design Basis Accident (DBA).

ECCS PREACS is considered OPERABLE when the individual components necessary to maintain the ECCS pump room filtration are OPERABLE in both trains.

(continued)

BASES

LCO (continued)

An ECCS PREACS train is considered OPERABLE when its associated:

a. Fan is OPERABLE;

b. HEPA filter and charcoal absorber are not excessively restricting flow, and are capable of performing their filtration functions;

c. Heater, demister, ductwork, valves, and dampers are OPERABLE and air circulation can be maintained; and

1

APPLICABILITY

In MODES 1, 2, 3, and 4, the ECCS PREACS is required to be OPERABLE consistent with the OPERABILITY requirements of the ECCS.

In MODE 5 or 6, the ECCS PREACS is not required to be OPERABLE since the ECCS is not required to be OPERABLE.

ACTIONS

A.1

With one ECCS PREACS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this time, the remaining OPERABLE train is adequate to perform the ECCS PREACS function.

The 7-day Completion Time is appropriate because the risk contribution is less than that for the ECCS (72-hour Completion Time), and this system is not a direct support system for the ECCS. The 7-day Completion Time is based on the low probability of a DBA occurring during this time period, and that the remaining train can provide the required capability.

Concurrent failure of two ECCS PREACS trains would result in the loss of functional capability; therefore, LCO 3.0.3 must be entered immediately.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the ECCS PREACS train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full-power conditions in an orderly manner and without challenging unit systems.

1

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

This SR verifies that a train in a standby mode starts on demand and continues to operate. Standby systems should be checked periodically to ensure that they start and function properly. As the environment and normal operating conditions on this system are not severe, testing each train once a month provides an adequate check on this system. Monthly heater operations dry out any moisture that may have accumulated in the charcoal from humidity in the ambient air. [Systems with heaters must be operated ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.] The 31-day Frequency is based on the known reliability of equipment and the two-train redundancy available.

SR 3.7.12.2

This SR verifies that the required ECCS PREACS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. The ECCS PREACS filter tests are in accordance with Reference 4. The [VFTP] includes testing HEPA filter performance, charcoal absorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test Frequencies and additional information are discussed in detail in the [VFTP].

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.3

This SR verifies that each ECCS PREACS train starts and operates on an actual or simulated actuation signal. The [18]-month Frequency is consistent with that specified in Reference 4.

SR 3.7.12.4

This SR verifies the integrity of the ECCS pump room enclosure. The ability of the ECCS pump room to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify proper functioning of the ECCS PREACS. During the [post-accident] mode of operation, the ECCS PREACS is designed to maintain a slight negative pressure in the ECCS pump room, with respect to adjacent areas, to prevent unfiltered LEAKAGE. The ECCS PREACS is designed to maintain a $\leq$ [-0.125] inches water gauge relative to atmospheric pressure at a flow rate of [3000] cfm from the ECCS pump room. The Frequency of [18] months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 6).

This test is conducted with the tests for filter penetration; thus, an [18]-month Frequency on a STAGGERED TEST BASIS is consistent with that specified in Reference 4.

SR 3.7.12.5

Operating the ECCS PREACS bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the ECCS PREACS bypass damper is verified if it can be closed. An [18]-month Frequency is consistent with that specified in Reference 4.

1

REFERENCES

1. FSAR, Section [6.5.1].
2. FSAR, Section [9.4.5].
3. FSAR, Section [15.6.5].

(continued)

1

BASES

REFERENCES (continued)

4. Regulatory Guide 1.52 (Rev. 2).
5. 10 CFR 100.11.
6. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.

JUSTIFICATIONS FOR DEVIATIONS FROM STS 3.7.12

1. Change to delete entire specification since it does not apply to Watts Bar. The ABGTS (Watts Bar LCO 3.7.12) performs this function as well as the functions for STS LCOs 3.7.13 and 3.7.14.

B 3.7 PLANT SYSTEMS

B 3.7.~~13~~ 12 Auxiliary Building Gas Treatment System ~~Fuel Building Air Cleanup System (FBACS)~~ 1 2

BASES

BACKGROUND 1 The ABGTS ~~FBACS~~ filters airborne radioactive ~~ABGTS~~ particulates from the area of the fuel pool following a fuel handling accident ~~or~~ and from the area of active unit ECCS components and unit penetration rooms following a loss-of-coolant accident (LOCA). ~~The FBACS, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the fuel pool area.~~ 6

The ABGTS ~~FBACS~~ consists of two independent and redundant trains. Each train consists of a heater, a prefilter ~~or demister~~ moisture separator, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system, ~~as well as demisters, functioning to reduce the relative humidity of the airstream~~. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case the main HEPA filter bank fails. The downstream HEPA filter is not credited in the analysis~~, but serves to collect charcoal fines, and to back up the upstream HEPA filter should it develop a leak~~. The system initiates filtered ventilation of the ~~fuel handling building~~ Auxiliary Building Secondary Containment Enclosure (ABSCE) following receipt of a Phase A Containment Isolation signal or a high radiation signal ~~from~~ from the spent fuel pool area. 1 5 1

1 The ABGTS ~~FBACS~~ is a standby system, ~~parts of which may also be operated~~ not in use during normal plant operations. ~~Upon receipt of the actuating signal, normal air discharges from the building, the fuel handling building is isolated, and the stream of ventilation air discharges through the system filter trains.~~ During emergency operations, the ABSCE dampers are realigned and ABGTS fans are started to begin filtration. Air is exhausted from the ~~ESF~~ Unit 1 ECCS pump rooms, unit penetration rooms, and fuel handling area through the filter trains. 6 The prefilters or ~~demisters~~ moisture separators remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers. 5

The ABGTS ~~FBACS~~ is discussed in the FSAR, Sections ~~[6.5.1]~~ 15 and 6.2.3, 9.4.2 ~~[9.4.5]~~, and ~~[15.7.4]~~ (Refs. 1, 2, ~~and 3~~ 3, and 4, respectively)~~, because it may be used for normal, as well as post-accident, atmospheric cleanup functions~~. 3 6

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

1 The ~~FBACS~~ ABGTS design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel-handling accident. The analysis of the fuel-handling accident, given in Reference 3, assumes that all fuel rods in an assembly are damaged. The analysis of the LOCA assumes that radioactive materials leaked from the Emergency Core Cooling System (ECCS) are filtered and adsorbed by the 1 ABGTS ~~FBACS~~. The DBA analysis of the fuel-handling accident assumes that only one train of the ~~FBACS~~ ABGTS is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the one remaining train of this filtration system. 1 The amount of fission products available for release from the ~~fuel-handling building~~ ABSCE is determined for a fuel-handling accident and for a LOCA. These assumptions and the analysis follow the guidance provided in Regulatory Guides 1.25 (Ref. ~~5~~4) and 1.4 (Ref. 6). 6

1 The ~~FBACS~~ ABGTS satisfies Criterion 3 of the NRC Policy Statement.

LCO

Two independent and redundant trains of the ~~FBACS~~ ABGTS are required to be OPERABLE to ensure that at least one train is available, assuming a single failure that disables the other train, coincident with a loss of offsite power. Total system failure could result in the atmospheric release from 1 the ~~fuel-handling building~~ ABSCE exceeding the 10 CFR 100 limits (Ref. ~~7~~5) in the event of a fuel-handling accident or LOCA. 6

1 The ~~FBACS~~ ABGTS is considered OPERABLE when the individual components necessary to control exposure in the fuel-handling building are OPERABLE in both trains. An 1 ~~FBACS~~ ABGTS train is considered OPERABLE when its associated:

a. Fan is OPERABLE;

b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration function; and

1 c. Heater, ~~demister~~ moisture separator, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

7 A INSERT G

(continued)

INSERT G

APPLICABILITY	In MODE 1, 2, 3, or 4, the ABGTS is required to be OPERABLE to provide fission-product removal associated with ECCS leaks due to a LOCA and leakage from containment and annulus. In MODE 5 or 6, the ABGTS is not required to be OPERABLE since the ECCS is not required to be OPERABLE. During movement of irradiated fuel in the fuel handling area, the ABGTS is required to be OPERABLE to alleviate the consequences of a fuel-handling accident.

BASES (continued)

ACTIONS

A.1

(1) With one ~~FBACS~~ ABGTS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this period, the remaining OPERABLE train is adequate to perform the ABGTS ~~FBACS~~ function. The 7-day Completion Time is based on the risk from an event occurring requiring the inoperable ~~FBACS~~ ABGTS train, and the remaining ~~FBACS~~ ABGTS train providing the required protection.

B.1 and B.2

(1) (3) In MODE 1, 2, 3, or 4, when Required Action A.1 unit cannot be completed within the associated Completion Time, or when both ABGTS ~~FBACS~~ trains are inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 3 within 6 hours, and in MODE 5 within 36 hours. The Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full-power conditions in an orderly manner and without challenging unit systems.

C.1 and C.2

(1) When Required Action A.1 cannot be completed within the required Completion Time, during movement of irradiated fuel assemblies in the ~~fuel building~~ fuel handling area, the OPERABLE ~~FBACS~~ ABGTS train must be started immediately or fuel movement suspended. This action ensures that the remaining train is OPERABLE, that no undetected failures preventing system operation will occur, and that any active failure will be readily detected.

If the system is not placed in operation, this action requires suspension of fuel movement, which precludes a fuel-handling accident. This does not preclude the movement of fuel assemblies to a safe position.

D.1

(1) When two trains of the ~~FBACS~~ ABGTS are inoperable during movement of irradiated fuel assemblies in the fuel ~~building~~ handling area, action must be taken to place the unit in a condition in which the LCO does not apply. Action must be taken immediately to

(continued)

BASES

ACTIONS

D.1 (continued)

suspend movement of irradiated fuel assemblies in the fuel ~~building~~ handling area (1). This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.~~13~~12.1 (2)

~~This SR verifies that a train in a standby mode starts on demand and continues to operate.~~ (A) Standby systems should be checked periodically to ensure that they start and function properly. As the environmental and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system.

Monthly heater operation ~~drys~~ dries (7) out any moisture accumulated in the charcoal from humidity in the ambient air. ~~[Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.]~~ (5) The 31-day Frequency is based on the known reliability of the equipment and the two-train redundancy available.

SR 3.7.~~13~~12.2 (2)

This SR verifies that the required ABGTS ~~FBACS~~ testing is performed in accordance with the ~~[~~Ventilation Filter Testing Program (VFTP~~)~~]. The ABGTS ~~FBACS~~ filter tests are in accordance with Regulatory Guide 1.52 (Ref. ~~6~~8). The ~~[~~VFTP~~]~~ includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the ~~[~~VFTP~~]~~. (1) (3)

SR 3.7.~~13~~12.3 (2)

(3) This SR verifies that each ABGTS (1) ~~FBACS~~ train starts and operates on an actual or simulated actuation signal. The ~~[~~18~~]~~-month Frequency is consistent with Reference ~~6~~ 8.

(continued)

BASES

ACTIONS (continued)

A.3

Due to the potential of having diluted the boron concentration of the reactor coolant, SR 3.9.1.1 (verification of boron concentration) must be performed whenever Condition A is entered to demonstrate that the required boron concentration exists. The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

These valves are to be secured closed to isolate possible dilution paths. The likelihood of a significant reduction in the boron concentration during MODE 6 operations is remote due to the large mass of borated water in the refueling cavity and the fact that all unborated water sources are isolated, precluding a dilution. The boron concentration is checked every 72 hours during MODE 6 under SR 3.9.1.1. This surveillance demonstrates that the valves are closed through a system walkdown. The 31-day Frequency is based on engineering judgment and is considered reasonable in view of other administrative controls that will ensure that the valve opening is an unlikely possibility.

REFERENCES

1

1. FSAR, Section ~~[~~15.2.4~~]~~, "Uncontrolled Boron Dilution."
2. NUREG-0800, Standard Review Plan, Section 15.4.6, "Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the RCS."

BASES

SURVEILLANCE REQUIREMENTS (continued) (1)

12 (2)
SR 3.7.~~13~~.4

This SR verifies the integrity of the ~~fuel building enclosure~~ ABSCE. The ability of the ~~fuel building~~ ABSCE to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the ~~FBACS~~ ABGTS. During the ~~[~~post-accident~~]~~ mode of operation, the ~~FBACS~~ ABGTS is designed to maintain a slight negative pressure in the ~~fuel building~~ ABSCE, to prevent unfiltered LEAKAGE. The ~~FBACS~~ ABGTS is designed to maintain a ~~≤ [-0.125]~~ negative pressure between -0.25 and -0.5 inches water gauge in the ABSCE with respect to atmospheric pressure at a ~~flow rate of [20,000] cfm to the fuel building~~ nominal flow rate of 9000 ± 10% cfm while maintaining a vacuum relief rate ≥ 2000 CFM. The Frequency of ~~[~~18~~]~~ months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. ~~7~~ 9). (1) (1) (1) (3)

This test is conducted with the tests for filter penetration; thus, an ~~[~~18~~]~~-month Frequency (on a STAGGERED TEST BASIS) is consistent with Reference ~~6~~ 8.

(4) (5)

~~SR 3.7.13.5~~

~~[Operating the FBACS filter bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the FBACS filter bypass damper is verified if it can be closed. An [18]-month Frequency is consistent with Reference 6.]~~

REFERENCES (8)

Watts Bar
1. FSAR, Section ~~[~~6.5.1~~]~~, "Engineered Safety Features ~~Ventilation~~ (ESF) Filter Systems."
2. FSAR, Section ~~[~~9.4.~~5]~~ 2, "Fuel Handling Area Ventilation System."
3. FSAR, ~~Section [15.7.4]~~ chapter 15, "Accident Analysis."
4. FSAR, Section 6.2.3, "Secondary Containment Functional Design."
5 ~~4~~. Regulatory Guide 1.25.
6. Regulatory Guide 1.4
7 ~~6~~ ~~5~~. 10 CFR 100.
8 ~~7~~ ~~6~~. Regulatory Guide 1.52 (Rev. 2).
9 ~~8~~ ~~7~~. NUREG-0800, Standard Review Plan, Section 6.5.1, Rev. 2, July 1981.

JUSTIFICATIONS FOR DEVIATIONS FROM STS 3.7.13

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

1. Change to reflect plant specific terminology/design.

2. Renumbered LCO to account for deletion of STS LCO 3.7.12.

3. Change to delete brackets that specify plant specific information and parameters.

4. The exhaust flow rate and vacuum relief flow rate are both required to be verified to ensure proper system operation.

5. Change to remove information not applicable to Watts Bar design.

6. The ABGTS performs the functions for STS LCOs 3.7.12, 3.7.13 and 3.7.14. The ABGTS does not provide for environmental control of temperature and humidity and does not function during normal operation. The relevant information from the other STS bases has been inserted into this bases.

7. Change to correct error in the STS. Westinghouse has been notified of the error and has notified the NRC chapter coordinator.

8. Watts Bar prefers to use the expanded reference information and format.

1

3.7 PLANT SYSTEMS

3.7.14 Penetration Room Exhaust Air Cleanup System (PREACS)

LCO 3.7.14 Two PREACS trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One PREACS train inoperable.	A.1 Restore PREACS train to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.14.1	Verify the operation of each PREACS train for [≥ 10 continuous hours with heaters operating or (for systems without heaters) ≥ 15 minutes].	31 days

(continued)

1

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.7.14.2	Perform required PREACS filter testing in accordance with the [Ventilation Filter Testing Program (VFTP)].	In accordance with the [VFTP]
[SR 3.7.14.3	Verify each PREACS train actuates on an actual or simulated actuation signal.	[18] months]
[SR 3.7.14.4	Verify one PREACS train can maintain a pressure ≤ [-0.125] inches water gauge relative to atmospheric pressure during the [post-accident] mode of operation at a flow rate of ≤ [3000] cfm.	[18] months on a STAGGERED TEST BASIS]
[SR 3.7.14.5	Verify each PREACS filter bypass damper can be closed.	[18] months]

(F)

B 3.7 PLANT SYSTEMS

B 3.7.14 Penetration Room Exhaust Air Cleanup System (PREACS)

BASES

BACKGROUND

The PREACS filters air from the penetration area between containment and the auxiliary building.

The PREACS consists of two independent and redundant trains. Each train consists of a heater, a prefilter or demister, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation, as well as demisters, functioning to reduce the relative humidity of the airstream, also form part of the system. A second bank of HEPA filters, which follows the adsorber section, collects carbon fines and provides backup in case of failure of the main HEPA filter bank. The downstream HEPA filter, although not credited in the accident analysis, collects charcoal fines and serves as a backup should the upstream HEPA filter develop a leak. The system initiates filtered ventilation following receipt of a safety injection (SI) signal.

The PREACS is a standby system, parts of which may also operate during normal unit operations. During emergency operations, the PREACS dampers are realigned and fans are started to initiate filtration. Upon receipt of the actuating signal(s), normal air discharges from the penetration room, the penetration room is isolated, and the stream of ventilation air discharges through the system filter trains. The prefilters remove any large particles in the air, as well as any entrained water droplets, to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The PREACS is discussed in the FSAR, Sections [6.5.1], [9.4.5], and [15.6.5] (Refs. 1, 2, and 3, respectively) since it may be used for normal, as well as post-accident, atmospheric cleanup functions. Heaters may be included for moisture removal on systems operating in high humidity conditions. The primary purpose of the heaters is to maintain the relative humidity at an acceptable level consistent with iodine removal efficiencies per Regulatory Guide 1.52 (Ref. 4).

(continued)

1

BASES (continued)

APPLICABLE SAFETY ANALYSES

The PREACS design basis is established by the large-break loss-of-coolant accident (LOCA). The system evaluation assumes a passive failure outside containment, such as valve packing leakage during a Design Basis Accident (DBA). In such a case, the system restricts the radioactive release to within the 10 CFR 100 limits (Ref. 4), or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits). The analysis of the effects and consequences of a large-break LOCA are presented in Reference 3.

Two types of system failures are considered in the accident analysis: a complete loss of function, and excessive LEAKAGE. Either type of failure may result in less efficient removal of any gaseous or particulate material released to the penetration room following a LOCA.

The PREACS satisfies Criterion 3 of the NRC Policy Statement.

LCO

Two independent and redundant trains of the PREACS are required to be OPERABLE to ensure that at least one train is available, assuming there is a single failure disabling the other train coincident with a loss of offsite power.

The PREACS is considered OPERABLE when the individual components necessary to control radioactive releases are OPERABLE in both trains. A PREACS train is considered OPERABLE when its associated:

a. Fan is OPERABLE;

b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Heater, demister, ductwork, valves, and dampers are OPERABLE and air circulation can be maintained.

APPLICABILITY

In MODES 1, 2, 3, and 4, the PREACS is required to be OPERABLE, consistent with the OPERABILITY requirements of the Emergency Core Cooling System (ECCS).

(continued)

BASES

1

APPLICABILITY (continued)

In MODE 5 or 6, the PREACS is not required to be OPERABLE since the ECCS is not required to be OPERABLE.

ACTIONS

A.1

With one PREACS train inoperable, the action must be taken to restore OPERABLE status within 7 days. During this period, the remaining OPERABLE train is adequate to perform the PREACS function. The 7-day Completion Time is appropriate because the risk contribution of the PREACS is less than that of the ECCS (72-hour Completion Time), and this system is not a direct support system for the ECCS. The 7-day Completion Time is based on the low probability of a DBA occurring during this period, and the remaining train providing the required capability.

B.1 and B.2

If the inoperable train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full-power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR verifies that a train in a standby mode starts on demand and continues to operate. Standby systems should be checked periodically to ensure that they start and function properly. As the environmental and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system. Monthly heater operation dries out any moisture that may have accumulated in the charcoal as a result of humidity in the ambient air. [Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥ 15 minutes to

(continued)

BASES

1

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1 (continued)

demonstrate the function of the system.] The 31-day Frequency is based on the known reliability of equipment and the two train redundancy available.

SR 3.7.14.2

This SR verifies that the required PREACS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. The PREACS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 5). The [VFTP] includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the [VFTP].

SR 3.7.14.3

This SR verifies that each PREACS starts and operates on an actual or simulated actuation signal. The [18]-month Frequency is consistent with that specified in Reference 5.

SR 3.7.14.4

This SR verifies the integrity of the penetration room enclosure. The ability of the penetration room to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify proper function of PREACS. During the [post-accident] mode of operation, the PREACS is designed to maintain a ≤ [-0.125] inches water gauge relative to atmospheric pressure at a flow rate of [3000] cfm in the penetration room, with respect to adjacent areas, to prevent unfiltered LEAKAGE. The Frequency of [18] months is consistent with the guidance provided in NUREG-0800 (Ref. 6).

(continued)

BASES

1

SURVEILLANCE REQUIREMENTS

SR 3.7.14.4 (continued)

The minimum system flow rate maintains a slight negative pressure in the penetration room area, and provides sufficient air velocity to transport particulate contaminants, assuming only one filter train is operating. The number of filter elements is selected to limit the flow rate through any individual element to about [3000] cfm. This may vary based on filter housing geometry. The maximum limit ensures that the flow through, and pressure drop across, each filter element are not excessive.

The number and depth of the adsorber elements ensure that, at the maximum flow rate, the residence time of the air stream in the charcoal bed achieves the desired adsorption rate. At least a [0.125]-second residence time is necessary for an assumed [99]% efficiency.

The filters have a certain pressure drop at the design flow rate when clean. The magnitude of the pressure drop indicates acceptable performance, and is based on manufacturers' recommendations for the filter and adsorber elements at the design flow rate. An increase in pressure drop, or a decrease in flow indicates that the filter is being loaded or that there are other problems with the system.

This test is conducted along with the tests for filter penetration; thus, the [18]-month Frequency is consistent with that specified in Reference 5.

SR 3.7.14.5

It is necessary to operate the PREACS filter bypass damper to ensure that the system functions properly. The OPERABILITY of the PREACS filter bypass damper is verified if it can be closed. An [18]-month Frequency is consistent with that specified in Reference 5.

REFERENCES

1. FSAR, Section [6.5.1].

2. FSAR, Section [9.4.5].

(continued)

1

BASES

REFERENCES (continued)

3. FSAR, Section [15.6.5].

4. 10 CFR 100.

5. Regulatory Guide 1.52, Rev. 02.

6. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.

JUSTIFICATIONS FOR DEVIATIONS FROM STS 3.7.14

1. Change to delete entire specification since it does not apply to Watts Bar. The ABGTS (Watts Bar LCO 3.7.12) performs this function as well as the functions for STS LCOs 3.7.12 and 3.7.13.

3.7 PLANT SYSTEMS

3.7.~~15~~ 13 Fuel Storage Pool Water Level

1

LCO 3.7.~~15~~ 13 The fuel storage pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify the fuel storage pool water level is ≥ 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	7 days

B 3.7 PLANT SYSTEMS

B 3.7.~~15~~ 13 Fuel Storage Pool Water Level 1

BASES

BACKGROUND

The minimum water level in the fuel storage pool meets the assumptions of iodine-decontamination factors following a fuel-handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

2

A general description of the fuel storage pool design is given in the FSAR, Section ~~[~~9.1.2~~]~~, Reference 1. A description of the Spent Fuel Pool Cooling and Cleanup System is given in the FSAR, Section ~~[~~9.1.3~~]~~, Reference 2. The assumptions of the fuel-handling accident are given in the FSAR, Section ~~[15.7.4]~~ 15.4.5, Reference 3.

APPLICABLE SAFETY ANALYSES

The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.25 (Ref. 4). The resultant 2-hour thyroid dose per person at the exclusion area boundary is a small fraction of the 10 CFR 100 (Ref. 5) limits.

According to Reference 4, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel-handling accident. With 23 ft of water, the assumptions of Reference 4 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks, however, there may be < ~~23 ft of water above the~~ top of the fuel bundle and the surface, indicated by the width of the bundle. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rows fail from a hypothetical maximum drop. A

The fuel storage pool water level satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

The fuel storage pool water level is required to be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel-handling accident analysis (Ref. 3). As such, it is the minimum required for fuel storage and movement within the fuel storage pool.

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the fuel storage pool, since the potential for a release of fission products exists.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the fuel storage pool water level is lower than the required level, the movement of irradiated fuel assemblies in the fuel storage pool is immediately suspended to a safe position. This action effectively precludes the occurrence of a fuel-handling accident. This does not preclude movement of a fuel assembly to a safe position.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.~~15~~13.1 1

This SR verifies sufficient fuel pool storage water is available in the event of a fuel-handling accident. The water level in the fuel storage pool must be checked periodically. The 7-day Frequency is appropriate because

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.~~15~~13.1 (continued) 1

the volume in the pool is normally stable. Water level changes are controlled by plant procedures and are acceptable based on operating experience.

During refueling operations, the level in the fuel storage pool is in equilibrium with the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.6.1.

REFERENCES

3

~~1. FSAR, Section [9.1.2].~~

~~2. FSAR, Section [9.1.3].~~

~~3. FSAR, Section [15.7.4].~~

~~4. Regulatory Guide 1.25, [Rev. 00].~~

~~5. 10 CFR 100.11.~~

1. Watts Bar FSAR ~~(Unit 1)~~, Section 9.1.2, "Spent Fuel Storage."

2. Watts Bar FSAR ~~(Unit 1)~~, Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System."

3. Watts Bar FSAR ~~(Unit 1)~~, Section 15.4.5, "Fuel Handling Accident."

4. Regulatory Guide 1.25, March 1972 ~~[Rev. 00]~~, "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."

5. Title 10, Code of Federal Regulations, Part 100.11,"Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

JUSTIFICATIONS FOR DEVIATIONS FROM STS 3.7.15

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

1. Renumbered LCO to account for deletion of STS LCO 3.7.12.

2. Change to delete brackets that specify plant specific information and parameters.

3. Watts Bar prefers to use the expanded reference information and format.

1

3.7 PLANT SYSTEMS

3.7.~~16~~ 14 Fuel Storage Pool Boron Concentration

LCO 3.7.~~16~~ 14 The fuel storage pool boron concentration shall be ≥ [2300] ppm.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool boron concentration not within limit.	-----------NOTE------------- LCO 3.0.3 is not applicable. -----------------------------	
	A.1 Suspend movement of fuel assemblies in the fuel storage pool.	Immediately
	AND	
	A.2.1 Initiate action to restore fuel storage pool boron concentration to within limit.	Immediately
	OR	
	A.2.2 Verify a by administrative means [Region 2] fuel storage pool verification has been performed since the last movement of fuel assemblies in the fuel storage pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.~~16~~ 14.1	Verify the fuel storage pool boron concentration is within limit.	7 days

(1)

B 3.7 PLANT SYSTEMS

B 3.7.~~16~~ 14 Fuel Storage Pool Boron Concentration

BASES

BACKGROUND

In the Maximum Density Rack (MDR) [(Refs. 1 and 2)] design, the spent fuel storage pool is divided into two separate and distinct regions which, for the purpose of criticality considerations, are considered as separate pools. [Region 1], with [336] storage positions, is designed to accommodate new fuel with a maximum enrichment of [4.65] wt% U-235, or spent fuel regardless of the discharge fuel burnup. [Region 2], with [2670] storage positions, is designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within the acceptable domain according to Figure [3.7.~~17~~ 15-1], in the accompanying LCO. Fuel assemblies not meeting the criteria of Figure [3.7.~~17~~ 15-1] shall be stored in accordance with paragraph 4.3.1.1 in Section 4.3, "Fuel Storage."

The water in the spent fuel storage pool normally contains soluble boron which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 3) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is associated with the movement of fuel from [Region 1 to Region 2], and accidental misloading of a fuel assembly in [Region 2]. This could potentially increase the criticality of [Region 2]. To mitigate these postulated criticality-related accidents, boron is dissolved in the pool water. Safe operation of the MDR with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with LCO 3.7.~~17~~ 15, "Spent Fuel Assembly Storage." Prior to movement of an assembly, it is necessary to perform SR 3.7.~~16~~ 14.1.

(continued)

BASES (continued)

1

APPLICABLE SAFETY ANALYSES

Most accident conditions do not result in an increase in the activity of either of the two regions. Examples of these accident conditions are the loss of cooling (reactivity increase with decreasing water density) and the dropping of a fuel assembly on the top of the rack. However, accidents can be postulated, which could increase the reactivity. This increase in reactivity is unacceptable with unborated water in the storage pool. Thus, for these accident occurrences, the presence of soluble boron in the storage pool prevents criticality in both regions. The postulated accidents are basically of two types. A fuel assembly could be incorrectly transferred from [Region 1 to Region 2] (e.g., an unirradiated fuel assembly or an insufficiently depleted fuel assembly). The second type of postulated accidents is associated with a fuel assembly which is dropped adjacent to the fully loaded [Region 2] storage rack. This could have a small positive reactivity effect on [Region 2]. However, the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by either one of the two postulated accident scenarios. The accident analyses is provided in the FSAR, Section [15.7.4], Reference 4.

The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of the NRC Policy Statement.

LCO

The fuel storage pool boron concentration is required to be ≥ [2300] ppm. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in Reference 4. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the fuel storage pool.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel storage pool, until a complete spent fuel storage pool verification has been performed following the last movement of fuel assemblies in the spent fuel storage pool. This LCO does not apply following the verification, since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in

(continued)

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BASES

APPLICABILITY (continued)

progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

ACTIONS

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

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. An acceptable alternative is to verify by administrative means that the fuel storage pool verification has been performed since the last movement of fuel assemblies in the fuel storage pool. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.

If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.~~16~~ 14.1

This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 7-day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.

(continued)

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BASES (continued)

REFERENCES

1. [Callaway FSAR, Appendix 9.1A, "The Maximum Density Rack (MDR) Design Concept."]

2. [Description and Evaluation for Proposed Changes to Facility Operating Licenses DPR-39 and DPR-48 (Zion Power Station).]

3. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

4. FSAR, Section [15.7.4].

JUSTIFICATIONS FOR DEVIATIONS FROM STS 3.7.16

1. Watts Bar does not utilize the maximum density rack design concept.

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3.7 PLANT SYSTEMS

3.7.~~17~~ 15 Spent Fuel Assembly Storage

LCO 3.7.~~17~~ 15 The combination of initial enrichment and [discharge fuel] burnup of each spent fuel assembly stored in [Region 2] shall be within the Acceptable [Burnup Domain] of Figure 3.7.~~17~~ 15-1.

APPLICABILITY: Whenever any fuel assembly is stored in [Region 2] of the spent fuel storage pool.

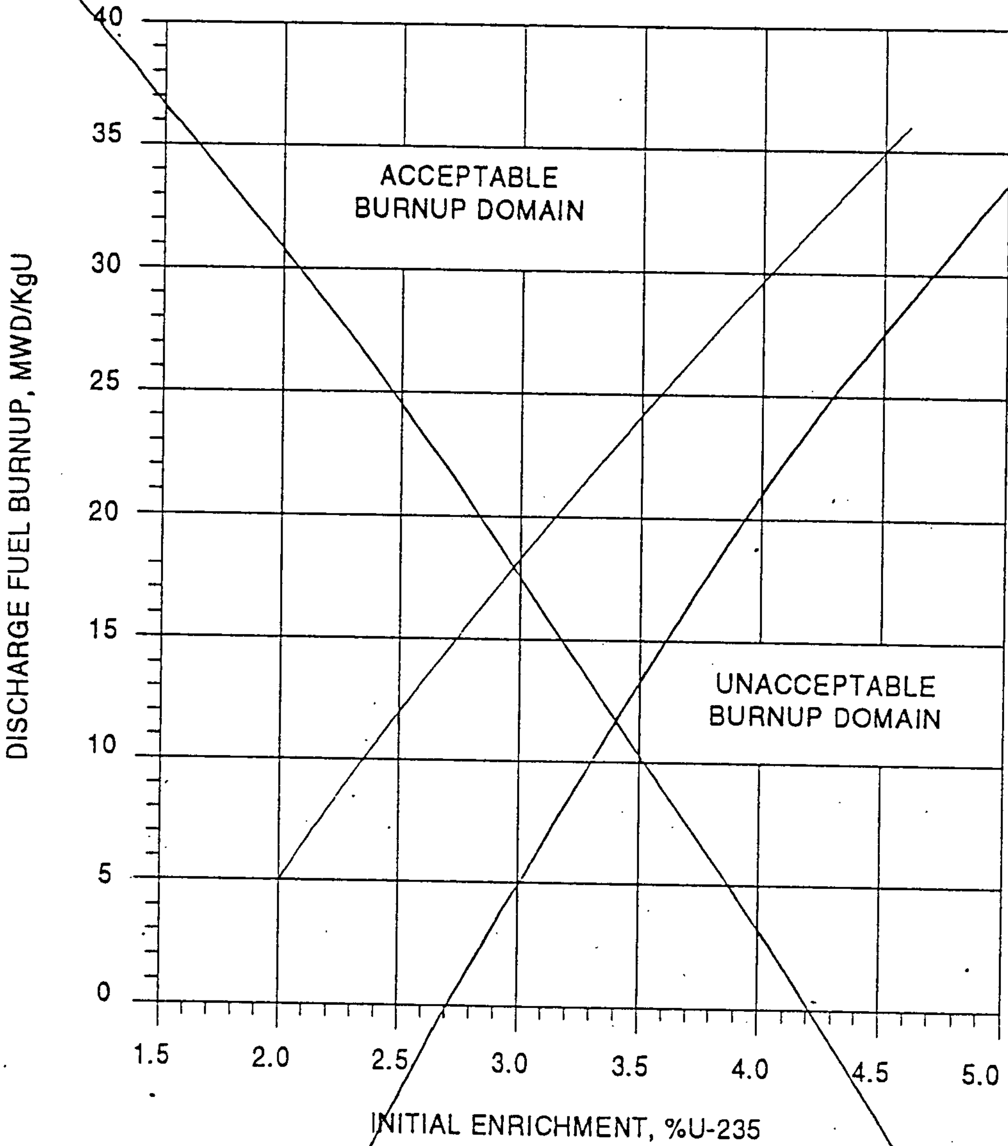
ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of this LCO not met.	------------NOTE------------ LCO 3.0.3 is not applicable. ---------------------------- A.1 Initiate action to move the noncomplying fuel assembly from [Region 2].	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.~~17~~ 15.1	Verify by administrative means the initial enrichment and [discharge fuel] burnup of the fuel assembly is in accordance with Figure 3.7.~~17~~ 15-1.	Prior to storing the fuel assembly in [Region 2]

①



Not to be used for Operation.
For illustration purposes only.

Figure 3.7.~~17~~15-1 (page 1 of 1)
Fuel Assembly Burnup Limits in Region 2

1

B 3.7 PLANT SYSTEMS

B 3.7.~~17~~ 15 Spent Fuel Assembly Storage

BASES

BACKGROUND

In the Maximum Density Rack (MDR) [(Refs. 1 and 2)] design, the spent fuel storage pool is divided into two separate and distinct regions which, for the purpose of criticality considerations, are considered as separate pools. [Region 1], with [336] storage positions, is designed to accommodate new fuel with a maximum enrichment of [4.65] wt% U-235, or spent fuel regardless of the discharge fuel burnup. [Region 2], with [2670] storage positions, is designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within the acceptable domain according to Figure 3.7.~~17~~-1, in the accompanying LCO. Fuel assemblies not meeting the criteria of Figure [3.7.~~17~~ 15-1] shall be stored in accordance with paragraph 4.3.1.1 in Section 4.3, "Fuel Storage." 15

The water in the spent fuel storage pool normally contains soluble boron which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 3) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe, accident scenario is associated with the movement of fuel from [Region 1 to Region 2], and accidental misloading of a fuel assembly in [Region 2]. This could potentially increase the criticality of [Region 2]. To mitigate these postulated criticality-related accidents, boron is dissolved in the pool water. Safe operation of the MDR with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with the accompanying LCO. Prior to movement of an assembly, it is necessary to perform LCO 3.7.~~16~~ 14, "Fuel Storage Pool Boron Concentration," and perform SR 3.7.~~16~~ 14.1.

(continued)

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BASES (continued)

APPLICABLE SAFETY ANALYSES

The hypothetical accidents can only take place during or as a result of the movement of an assembly (Ref. 4). For these accident occurrences, the presence of soluble boron in the spent fuel storage pool (controlled by LCO 3.7.~~16~~) prevents criticality in both regions. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents may be limited to a small fraction of the total operating time. During the remaining time period with no potential for accidents, the operation may be under the auspices of the accompanying LCO. 14

The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of the NRC Policy Statement.

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, in accordance with Figure 3.7.~~17~~-1, in the accompanying LCO, ensures the k_{eff} of the spent fuel storage pool will always remain 0.95, assuming the pool to be flooded with unborated water. Fuel assemblies not meeting the criteria of Figure [3.7.~~17~~-1] shall be stored in accordance with paragraph 4.3.1.1 in Section 4.3, "Fuel Storage." 15 15

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in [Region 2] of the fuel storage pool.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the configuration of fuel assemblies stored in [Region 2] the spent fuel storage pool is not in accordance with Figure 3.7.~~17~~-1, or paragraph 4.3.1.1, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figure 3.7.~~17~~-1, or paragraph 4.3.1.1. 15 15

(continued)

BASES

1

ACTIONS

A.1 (continued)

If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.~~17~~ 15.1

This SR verifies by administrative means that the initial enrichment and [discharge fuel] burnup of the fuel assembly is in accordance with Figure [3.7.~~17~~ 15.-1] in the accompanying LCO. For fuel assemblies in the unacceptable range of Figure 3.7.~~17~~ 15-1, performance of this SR will ensure compliance with paragraph 4.3.1.1.

REFERENCES

1. [Callaway FSAR, Appendix 9.1A, "The Maximum Density Rack (MDR) Design Concept."]
2. [Description and Evaluation for Proposed Changes to Facility Operating Licenses DPR-39 and DPR-48 (Zion Power Station).]
3. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).
4. FSAR, Section [15.7.4].

JUSTIFICATIONS FOR DEVIATIONS FROM STS 3.7.17

1. Watts Bar does not utilize the maximum density rack design concept.

3.7 PLANT SYSTEMS

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3.7.~~18~~14 Secondary Specific Activity

LCO 3.7.~~18~~14 The specific activity of the secondary coolant shall be ≤ [0.100] μCi/gm DOSE EQUIVALENT I-131.

2

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	A.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
1 SR 3.7.~~18~~14.1	Verify the specific activity of the secondary coolant is ≤ [0.100] μCi/gm DOSE EQUIVALENT I-131. 2	31 days

B 3.7 PLANT SYSTEMS

B 3.7.~~18~~ 14 Secondary Specific Activity (1)

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady-state conditions, the activity is primarily iodines with relatively short half-lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission-product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of (2) 0.1 [1.0] μCi/gm (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half-lives, (i.e., < 20 hours). I-131, with a half-life of 8.04 days, concentrates faster than it decays, but does not reach equilibrium because of blowdown and other losses.

With the specified activity limit, the resultant 2-hour thyroid dose to a person at the exclusion area boundary (EAB) would be about 0.58 rem if the main steam safety valves (MSSVs) open for 2 hours following a trip from full power.

Operating a unit at the allowable limits could result in a 2-hour EAB exposure of a small fraction of the 10 CFR 100 (Ref. 1) limits, or the limits established as the NRC staff-approved licensing basis.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the FSAR, Chapter [15] (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of [0.10] μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed a small fraction of the unit EAB limits (Ref. 1) for whole-body and thyroid dose rates. 2

With the loss of offsite power, the remaining steam generators are available for core-decay-heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator atmospheric dump valves (ADVs). The Auxiliary Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line, is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through the MSSVs and ADVs during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

2

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be ≤ [0.10] μCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to a small fraction of the required limit (Ref. 1).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner

(continued)

BASES

LCO (continued)

to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary-to-secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity cannot be restored to within limits within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full-power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.~~18~~/14.1 1

This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post-accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The ~~[~~31~~]~~-day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit. 2

(continued)

BASES (continued)

REFERENCES ③

1. 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance"
2. Watts Bar FSAR, Chapter ~~[15]~~ 15, "Accident Analysis."

JUSTIFICATIONS FOR DEVIATIONS FROM STS 3.7.18

1. Renumbered LCO to account for deletion of STS LCO 3.7.12, 3.7.14, 3.7.16, and 3.7.17.

2. Change to delete brackets that specify plant specific information and parameters.

3. Watts Bar prefers to use the expanded reference information and format.

NOTE: This copy of the Westinghouse Standard Technical Specifications (NUREG-1431, Draft) is based on a second round draft (May 1992) which addressed industry comments on the NUREG submitted in July 1991. The editorial review by the industry and the NRC (June 1992) resulted in additional changes which are denoted by a greek delta symbol. The Proof & Review copy of the RSTS (July 1992) was not available in time to for inclusion into the Watts Bar markup, however, there should be only minor editorial differences between the editorial review copy (June 1992) and the Proof & Review copy (July 1992) of the RSTS.

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources – Operating

LCO 3.8.1 The following AC electrical sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E ~~power~~ AC Electrical △ ~~distribution~~ ~~sub~~system~~(s)~~; and

1 b. ~~Two~~ Four Diesel Generators (DG~~)~~s capable of supplying the onsite Class 1E ~~power~~ ~~distribution~~ ~~sub~~system~~(s)~~ ~~[; and~~

2 ~~c. Automatic load sequencers for Train A and Train B].~~

3 NOTE
The C-S DG may be substituted for any of the required DGs.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ~~[required]~~ 4 offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 △ ~~(offsite circuit check)~~ for the ~~[required]~~ OPERABLE offsite circuit.	1 hour AND Once per 8 hours thereafter
	AND	
	A.2 Declare required feature(s) with no offsite power available, inoperable when ~~its/~~their △ required redundant feature~~(s)~~ ~~is/~~are inoperable.	24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature~~(s)~~
	AND	

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 Restore ~~[required]~~ offsite circuit~~s~~ to OPERABLE status.	72 hours *AND* 6 days from discovery of failure to meet ~~the~~ LCO
B. ---------NOTE--------- Required Actions B.3.1 or B.3.2 shall be completed if this Condition is entered. -------------------- ~~One [required] diesel generator (DG) inoperable.~~ 1 One or more required DG(s) in Train A inoperable. OR One or more required DG(s) in Train B inoperable.	B.1 Perform SR 3.8.1.1 ~~(offsite circuit check)~~ for the ~~[required]~~ offsite circuits.	1 hour *AND* Once per 8 hours thereafter
	AND	
	B.2 Declare required feature(s) supported by the inoperable DG(s) inoperable when ~~its/~~their required redundant feature(s) ~~is/~~are inoperable.	4 hours from discovery of Condition B concurrent with inoperability of required redundant feature(s)
	AND	
	B.3.1 Determine OPERABLE DGs are ~~is~~ not inoperable due to common cause failure.	~~[8]~~ 24 hours 5
	OR	
	B.3.2 Perform SR 3.8.1.2 ~~(DG start)~~ for OPERABLE DGs	~~[8]~~ 24 hours
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.4 Restore required DG(s) to OPERABLE status.	72 hours **AND** 6 days from discovery of failure to meet ~~the~~ LCO
C. Two ~~[required]~~ offsite circuits inoperable.	C.1 Declare required feature(s) inoperable when ~~its/~~their required redundant feature(s) ~~is/~~are inoperable. **AND** C.2 Restore one ~~required~~ offsite circuit to OPERABLE status.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s) 24 hours
D. One ~~[required]~~ offsite circuit inoperable. **AND** ~~One required DG inoperable.~~ One or more required DG(s) in Train A inoperable, OR One or more required DG(s) in Train B inoperable	------------NOTE------------ ~~When Condition E is entered with no AC power source to one train,~~ Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems-Operating." ~~for AC Distribution Subsystem,~~ ~~[illegible]~~ ------------------------------ D.1 Restore ~~required~~ offsite circuit to OPERABLE status. **OR**	~~With one AC Train de-energized as a result of Condition D,~~ 12 hours (continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (continued)	D.2	Restore required DG(s) to OPERABLE status.	12 hours
See next sheet for Condition E 1 ~~E. Two required DGs inoperable.~~	~~E.1~~	~~Restore one required DG to OPERABLE status.~~	~~2 hours~~
F. Required Actions and Associated Completion Times of Condition A, B, C, D, E or ~~X~~ not met.	F.1 AND F.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours
See next sheet for Condition G 1 ~~G. Three or more required AC sources inoperable.~~	~~G.1~~	~~Enter LCO 3.0.3.~~	~~Immediately~~
See next Sheet for Condition H 1 ~~H. ---------NOTE--------- This Condition may be deleted if the unit design is such that any sequencer failure mode will only affect the ability of the associated diesel generator to power its respective safety loads following a loss of offsite power independent of, or coincident with a design basis event. --------------------- One required load sequencer inoperable.~~	~~H.1~~	~~Restore all required automatic load sequencers to OPERABLE status.~~	~~[12] hours~~

2

E. One or more required DG(s) in Train A inoperable. AND One or more required DG(s) in Train B inoperable.	E.1 Restore required DGs in Train A to OPERABLE status. OR E.2 Restore required DGs in Train B to OPERABLE status	2 hours 2 hours
G. Two offsite circuits inoperable. AND One or more required DG(s) in Train A inoperable. OR One or more required DG(s) in Train B inoperable.	G.1 Enter LCO 3.0.3.	Immediately
H. One offsite circuit inoperable. AND One or more required DG(s) in Train A inoperable. AND One or more required DG(s) in Train B inoperable.	H.1 Enter LCO 3.0.3	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each ~~[required]~~ offsite circuit.	7 days
SR 3.8.1.2	-------------------NOTES------------------- 1. Performance of SR 3.8.1.7 satisfies this SR. (6) 2. All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to engine loading. e ~~3.~~ 2. A modified DG start involving idling and gradual acceleration to synchronous speed may be used for this SR as recommended by the manufacturer. When modified start procedures are not used, the time, voltage, and frequency tolerances of SR 3.8.1.~~8~~ 7 must be met. e Δ --- Verify each DG starts from standby conditions and achieves steady state voltage and frequency ~~within the ranges:~~ Δ ~~a.~~ ≥ [~~3740~~] TBD V and ≤ [~~4580~~] V~~;~~ and e ~~b.~~ ≥ [58.8] Hz and ≤ [61.2] Hz.	As specified ~~by~~ in Table 3.8.1-1 Δ

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.3 △	-----NOTES----- 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load and power factor range s do not invalidate this test. 3. This surveillance shall be conducted ~~on~~ only one DG at a time. 4. This SR shall be preceeded by and immediately follow without shutdown, a successful performance of SR 3.8.1.2 or SR 3.8.1.7. ----------------------------------- (A) Verify each DG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ ~~[4500]~~ 3960 kW and ≤ ~~[5000]~~ 4400 kW and power factor ≤ ~~[~~0.~~9~~8 0~~]~~.	As specified ~~by~~ in Table 3.8.1-1
SR 3.8.1.4	Verify each ~~fuel~~ skid mounted (A) day tank ~~[and engine mounted fuel tank]~~ contains ≥ ~~[ ]~~ 220 gal of fuel oil △	31 days
SR 3.8.1.5	Check for and remove accumulated water from each skid mounted (A) day tank, ~~[and engine mounted tank]~~.	~~[~~31~~]~~ days
SR 3.8.1.6	Verify ~~the~~ fuel oil △ transfer system operate(s) to ~~[automatically]~~ transfer fuel oil from storage tank(s) ~~[s]~~ to the skid-mounted (A) day tank ~~[and engine mounted tank]~~.	the 4day (A) ~~[92]~~ days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.7	(6) -------------------NOTE------------------- All DG starts may be preceded by an engine prelube period. -- Verify each DG starts from ~~standby~~ condition and achieves in ≤ [10] seconds, voltage and frequency ~~within the ranges:~~ ~~a.~~ ≥ [~~3740~~] 6555 V and ≤ [~~4580~~] 7260 V ~~and~~ ~~b.~~ ≥ [58.8] Hz and ≤ [61.2] Hz.	184 days
[SR 3.8.1.8	-------------------NOTES------------------- 1. This Surveillance shall not be performed in MODE 1 or 2. 2. Credit may be taken for unplanned events that satisfy this SR. -- Verify ~~[automatic/~~manual~~]~~ transfer of ~~AC power sources~~ each 6.9KV shutdown board power supply from the normal offsite circuit to each alternate ~~required~~ offsite circuit.	[18 months]]

(continued)

(B) each 6.9KV shutdown board power supply

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.9	-------------------NOTES------------------- 1. This Surveillance shall not be performed in MODE 1 or 2. 2. Credit may be taken for unplanned events that satisfy this SR. --- A Verify each DG operating at a power factor ≤ [0.~~9~~0] 8 rejects a load ≥ [~~1200~~] ~~600~~ 646 kW, and: A a. Following load rejection, the frequency is ≤ [~~63~~] 61.2 Hz; b. △ Within [3] seconds following load rejection, the voltage is ≥ [~~3740~~] 6555 V and ≤ [~~4580~~] 7260 V; and c. △ Within [3] seconds following load rejection, the frequency is ≥ [58.8] Hz and ≤ [61.2] Hz.	[18 months]
SR 3.8.1.10	-------------------NOTES------------------- 1. This Surveillance shall not be performed in MODE 1 or 2. 2. Credit may be taken for unplanned events that satisfy this SR. --- A Verify each DG, operating at a power factor ≤ [0.~~9~~] 8 does not trip and voltage is maintained ≤ [~~5000~~] 8280 V during and following a load rejection of ≥ [~~4500~~] 3960 kW and ≤ [~~5000~~] 4400 kW.	[18 months]

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.11	------------------NOTES------------------ ⑥ ~~1.~~ All DG starts may be preceded by an engine prelube period. 1 ~~2~~. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. 2 ~~3~~. Credit may be taken for unplanned events that satisfy this SR. --- Verify on an actual or simulated loss of offsite power signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; ~~and~~ c. DG auto-starts from standby condition and: 1. energizes permanently connected loads in ≤ [10] seconds, 2. energizes auto-connected shutdown loads through automatic load sequencer, 3. maintains steady-state voltage ≥ ~~[3740]~~ V and ≤ [~~4580~~] V, TBD 4. maintains steady-state frequency ≥ [58.8] Hz and ≤ [61.2] Hz, and 5. supplies permanently connected [and auto-connected] shutdown loads for ≥ [5] minutes.	[18 months]

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.12	------------------NOTES------------------ 6 1. All DG starts may be preceded by prelube procedures as recommended by the manufacturer. e 1 ~~2.~~ This Surveillance shall not be performed in MODE 1 or 2. e 2 ~~3.~~ Credit may be taken for unplanned events that satisfy this SR. --- Δ Engineered Safety Feature Verify on an actual or simulated ~~[ESF]~~ actuation signal each DG auto-starts from standby condition and: a. In ≤ [10] seconds after auto-start and during tests, achieves voltage ≥ [~~3740~~] 6555 V and ≤ [~~4580~~] 7260 V; b. In ≤ [10] seconds after auto-start and during tests, achieves Frequency ≥ ~~[~~58.8~~]~~ Hz and ≤ ~~[~~61.2~~]~~ Hz; c. Operates for ≥ ~~[~~5~~]~~ minutes; d. Permanently connected loads remain energized from the offsite power system; and 2 e. Emergency loads are energized ~~[or auto-connected through the automatic load sequencer]~~ to the offsite power system.	e ~~[~~18 months~~]~~

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.13	------------------NOTES------------------ 1. This Surveillance shall not be performed in MODES 1, 2, 3, or 4. 2. Credit may be taken for unplanned events that satisfy this SR. --- Verify each DG's automatic trips are bypassed on actual or simulated loss of voltage signal on the emergency bus concurrent with an actual or simulated ESF actuation signal except: a. Engine overspeed; and b. Generator differential current~~;~~. ~~c. [Low lube oil pressure];~~ 7 ~~d. [High crankcase pressure]; and~~ ~~e. [Start failure relay].~~	[18 months]

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.1.14	----------NOTES---------- S 1. Momentary transients outside the load and power factor range do not invalidate this test. 2. This Surveillance shall not be performed in MODE 1 or 2. 3. Credit may be taken for unplanned events that satisfy this SR. ----------------------------------- Verify each DG operating at a power factor ≤ [0.~~9~~ 8] operates for ≥ 24 hours: A a. For ≥ ~~[2]~~ 4 hours loaded Δ ≥ ~~[5250]~~ 4620 kW ~~≤ load~~ and ≤ ~~[5,500]~~ 4840 kW; and b. For the remaining hours of the test loaded Δ ≥ ~~[4500]~~ 3960 kW and ≤ ~~[5000]~~ 4400 kW.	[18 months]

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.15	------------------NOTE~~S~~------------------ ~~1.~~ This Surveillance shall be performed within 5 minutes of shutting down the DG after the DG has operated ≥ [2] hours loaded ≥ ~~[4500]~~ 3960 kW and ≤ ~~[5000]~~ 4400 kW ~~for [Division 1 and 2] DGs~~. Momentary transients outside of load range do not invalidate this test. 6 2. All DG starts may be preceded by an engine prelube period. -- Verify each DG starts and achieves in ≤ [10] seconds, voltage and frequency ~~a.~~ ≥ ~~[3740]~~ 6555 V and ≤ ~~[4580]~~ 7260 V ~~and~~ ~~b.~~ ≥ [58.8] Hz and ≤ [61.2] Hz.	~~[~~18 months~~]~~

A

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.16	-------------------NOTES------------------ 1. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. 2. Credit may be taken for unplanned events that satisfy this SR. --- Verify each DG: a. Synchronizes with offsite power source while loaded with emergency loads upon a simulated restoration of offsite power; b. Transfers loads to offsite power source; and c. Returns to ready-to-load operation.	[18 months]
~~SR 3.8.1.17~~	~~-------------------NOTES------------------~~ ~~1. This Surveillance shall not be performed in MODE 1, 2, 3, or 4.~~ ~~2. Credit may be taken for unplanned events that satisfy this SR.~~ ~~---~~ ~~Verify with a DG operating in test mode and connected to its bus, an actual or simulated ESF actuation signal overrides the test mode by:~~ ~~a. Returning DG to ready-to-load operation [; and~~ ~~b. Automatically energizing the emergency load from offsite power].~~	~~[18 months]~~

A

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.~~18~~ 17 (A)	------------------NOTES------------------ 1. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. 2. Credit may be taken for unplanned events that satisfy this SR. --- Verify the interval between each sequenced load block is within ~~± [10% of design interval]~~ the specified bands of FSAR Table 8.3-3, for each ~~emergency [and shutdown]~~ accident condition and non-accident condition load sequencer.	[18 months]
SR 3.8.1.~~16~~ 18 (6)	------------------NOTES------------------ 1. All DG starts may be preceded by prelube procedures as recommended by the manufacturer. 1 ~~2~~. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. 2 ~~3~~. Credit may be taken for unplanned events that satisfy this SR. --- Verify ~~that~~ on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ESF actuation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; ~~and~~ c. DG auto-starts from standby condition and: 1. energizes permanently connected loads in ≤ [10] seconds,	[18 months]

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.~~19~~ 18 (continued)	2. energizes auto-connected emergency loads through load sequencer, 3. achieves steady-state voltage ≥ [~~3740~~] TBD V and ≤ [4580] V, 4. achieves steady-state frequency ≥ [58.8] Hz and ≤ [61.2] Hz, and 5. supplies permanently connected and auto-connected emergency loads for ≥ [5] minutes.	
SR 3.8.1.~~20~~ 19 16	-------------------NOTE------------------- All DG starts may be preceded by engine prelube period. -- Verify ~~that~~ when started simultaneously from standby condition, each DG ~~each~~ achieves in ≤ [10] seconds, voltage and frequency ~~a.~~ ≥ [3744] 6555 V and ≤ [4576] 7260 V ~~and~~ ~~b.~~ ≥ [58.8] Hz and ≤ [61.2] Hz.	10 years

INSERT ADD'L SRs

8

INSERT

SURVEILLANCE		FREQUENCY
SR 3.8.1.20	Verify correct breaker alignment, indicated power availability and voltage ≥ 118V for each DG 125V DC distribution panel and associated battery charger.	7 days
SR 3.8.1.21	Verify battery terminal voltage ≥ 124 V on float charge for each DG.	7 days
SR 3.8.1.22	Verify no visible corrosion at terminals and connectors for each DG battery. <u>OR</u> 150 Verify connection resistance is ≤ ~~100~~ E-6 ohms for inter-cell connections, ≤ ~~100~~ E-6 ohms for inter-rack connections, ≤ ~~100~~ E-6 ohms for inter-tier connections, and ≤ ~~100~~ E-6 ohms for terminal connections for each DG battery.	92 days
SR 3.8.1.23	Verify cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration for each DG battery.	12 months
SR 3.8.1.24	Remove visible terminal corrosion, verify cell-to-cell and terminal connections are clean and coated with anti-corrosion material for each DG battery.	12 months
SR 3.8.1.25	150 Verify connection resistance is ≤ ~~100~~ E-6 ohms for inter-cell connections, ≤ ~~100~~ E-6 ohms for inter-rack connections, ≤ ~~100~~ E-6 ohms for inter-tier connections, and ≤ ~~100~~ E-6 ohms for terminal connections for each DG battery.	12 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.26	----------------NOTES------------------ 1. SR 3.8.1.27 may be performed in lieu of SR 3.8.1.26 once per 60 months. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. 3. Credit may be taken for unplanned events that satisfy this SR. --- Verify DG battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery-service test.	18 months

(continued)

INSERT (continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.1.27	-----------------NOTES----------------- 1. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. 2. Credit may be taken for unplanned events that satisfy this SR. ------------------------------------- Verify DG battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test.	60 months AND -----NOTE----- Only applicable when battery shows degradation or has reached 85% of the expected life -------------- 12 months OR -----NOTE----- Only applicable when battery does not show degradation and has reached 85% of the expected life and is ≥100% capacity -------------- 24 months

Table 3.8.1-1
Diesel Generator Test Schedule

NUMBER OF FAILURES IN LAST 25 VALID TESTS[a]	FREQUENCY
≤ 3	31 days
≥ 4	7 days[b] (but no less than 24 hours)

(a) Criteria for determining number of failures and valid test s shall be in accordance with Regulatory Position [C.2.1 of Regulatory Guide 1.9, Revision 3], where the number of test s and failures is determined on a per DG basis.

(b) This test frequency shall be maintained until seven consecutive failure-free starts from standby conditions and load-run tests have been performed. This is consistent with Regulatory Position [C.2.3.3], of Regulatory Guide 1.9, Revision 3]. If subsequent to the ~~seven~~ 7 failure-free tests ~~one~~ 1 or more additional failures occur such that there are again ~~four~~ 4 or more failures in the last 25 tests, the testing interval shall again be reduced as noted above and maintained until ~~seven~~ 7 consecutive failure free tests have been performed.

[Note: If Revision 3 of Regulatory Guide 1.9 is not approved, he above table will be modified to be consistent with the existing version of Regulatory Guide 1.108, GL 84-15, or other approved version.]

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources—Operating

BASES

A

BACKGROUND

Supplies electrical power to four power trains shared between the two units, with each train powered by an independent Class 1E 6.9 kV shutdown board. Power trains 1A and 2A comprise load group A, and power trains 1B and 2B comprise load group B. Two DGs associated with one load group can provide all safety related functions to mitigate a loss-of-coolant accident (LOCA) in one unit and safely shutdown the opposite unit. Each 6.9 kV shutdown board has two separate and independent offsite sources of power as well as a dedicated onsite DG source. The A and B train ESF systems each provide for the minimum safety functions necessary to shutdown the unit and maintain it in a safe shutdown condition.

~~Introduction~~ (19)

The unit ~~Class 1E AC~~ electrical power distribution system AC ~~power~~ sources consist of the offsite power sources (preferred power sources, normal and alternate(s)), and the onsite standby power sources (Train A and Train B diesel generators (DGs)). As required by 10 CFR 50, Appendix A, GDC 17, (Ref. 1), the design of the AC electrical power system provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) systems.

The onsite Class 1E AC distribution system ~~is divided into redundant load groups (trains) so that the loss of any one group will not prevent the minimum safety functions from being performed. Each train has connections to two preferred offsite power sources and a single Diesel Generator (DG).~~

Offsite power is supplied to the Insert from next page ~~unit switchyard(s) from the transmission network by [two] transmission lines. From the switchyard(s) two electrically and physically separated circuits provide AC power, through [step-down station auxiliary transformers], to the 4.16 kV ESF buses.~~ A detailed description of the offsite power network and ~~the~~ circuits to the Class 1E ~~ESF buses~~ shutdown boards is found in ~~the FSAR, Ref. 2.~~ Reference 3.

~~An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the~~ onsite Class 1E ESF ~~bus or~~ buses (i.e., 6.9 kV Shutdown Boards) A

~~Certain required plant loads are returned to service in a predetermined sequence in order to prevent overloading the transformer supplying offsite power to the onsite Class 1E Distribution System. Within [1 minute] after the initiating signal is received, all automatic and permanently connected loads needed to recover the unit or maintain it in a safe condition are returned to service via the load sequencer.~~

A single offsite circuit is capable of providing the ESF loads. Both of these circuits are required to meet the Limiting Condition for Operation.

(i.e., Watts Bar Hydro Plant switchyard) A

(continued)

Insert for pg B 3.8-1

Watts Bar 161KV transformer yard by two dedicated lines from the Watts Bar Hydro Plant switchyard. This is described in more detail in FSAR Chapter 8 (Ref. 3). From the 161KV transformer yard, two electrically and physically separated circuits provide AC power, through step-down common station service transformers, to the 6.9 KV shutdown boards. The two offsite AC electrical power sources are designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions.

WBN uses 4 DG sets for Unit 1 operation. These same DGs will be shared for Unit 2 operation. WBN also utilizes a C-S DG that can be manually aligned (electrically and mechanically) to any 6.9 kV shutdown board to replace an existing DG.

BASES (continued)

A

The onsite standby power source for each ~~4.16 kV ESF bus~~ 6.9 kV shutdown board is a dedicated ~~diesel generator.~~ DG. ~~Diesel generators (DGs) [11] and [12] are dedicated to ESF buses [11] and [12], respectively.~~ A DG starts automatically on a safety injection signal (SIS) (i.e., low pressurizer pressure or high containment pressure signals) or on an ~~[ESF bus~~ degraded voltage or ~~undervoltage~~ loss-of-voltage signal. After the ~~diesel generator~~ DG has started, it will automatically tie to its respective ~~bus~~ 6.9 kV shutdown board after offsite power is tripped as a consequence of ~~ESF bus undervoltage~~ loss-of-voltage or degraded voltage, independent of or coincident with a ~~safety injection signal~~ SIS. The DGs will also start and operate in the standby mode without tieing to the ~~ESF bus~~ 6.9 kV shutdown board on a ~~safety injection signal~~ SIS alone. Following the trip of offsite power, ~~[a sequencer/an undervoltage~~ a loss-of-voltage signal strips non-permanent loads from the ~~ESF bus~~ 6.9 kV shutdown board. When the DG is tied to the ~~ESF bus,~~ 6.9 kV shutdown board, loads are then sequentially connected to their respective ~~ESF bus~~ 6.9 kV shutdown board by the automatic ~~load~~ sequencer. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading the DG by automatic load application.

B

In the event of a loss of preferred power, the ~~ESF electrical loads~~ 6.9 kV shutdown boards are automatically connected to the ~~diesel generators~~ DGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a Design Basis Accident (DBA) such as a ~~loss of coolant accident (LOCA)~~.

Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the DG in the process. Within ~~1 minute~~ the required interval FSAR Table 8.3-3 after the initiating signal is received, all automatic and permanently connected loads needed to recover the unit or maintain it in a safe condition are returned to service. (10)

Ratings for Train ~~A and Train B~~ 1A, 1B, 2A, and 2B DGs satisfy the requirements of Regulatory Guide 1.9, (Ref. ~~3~~ 4). The continuous service rating of each ~~of the~~ DG is ~~[7,000]~~ 4400 kW with ~~[10]~~% overload permissible for up to 2 hours in any 24-hour period. The ESF loads that are powered from the ~~4.16 kV ESF buses~~ 6.9 kV shutdown boards are listed in ~~the FSAR, Ref. 2.~~ Reference 3.

A Insert 3.8.1-1

APPLICABLE SAFETY ANALYSES

The initial conditions of ~~design basis~~ DBAs and transient ~~and accident~~ analyses in the FSAR, Chapter 6 (Ref. 10), ~~"Engineering Safety Features"~~, and Chapter 15 (Ref. 8), ~~"Accident Analyses"~~, assume ESF systems are OPERABLE. The AC electrical power

(continued)

INSERT 3.8.1-1

Control power for the DGs is provided by five DG battery systems, one per DG. Each system is comprised of a battery, a battery charger, distribution center, cabling, and cable ways. The DG 125V DC control power and field-flash circuits have power supplied from their respective 125V distribution panel. The normal supply of DC current is from the associated charger. The battery provides control and field-flash power when the charger is unavailable. The charger supplies the normal DC loads, maintains the battery in a fully charged condition, and recharges (480V AC available) the battery while supplying the required loads regardless of the status of the unit. The batteries are physically and electrically independent. The battery has sufficient capacity when fully charged to supply required loads for a minimum of 30 minutes following a loss of normal power. Each battery is normally required to supply loads during the time interval between loss of normal feed to its charger and the receipt of emergency power to the charger from its respective DG.

BASES (continued)

sources 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 ~~(TS)~~ 3.2 "Power Distribution Limits," 3.4 "Reactor Coolant System," and 3.6 "Containment Systems."

The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This results in maintaining at least ~~one~~ two train of ~~the~~ onsite or one train of offsite AC sources OPERABLE during accident conditions in the event of: A 1

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

b. A worst-case single failure.

AC sources satisfy ~~— Operating satisfies~~ Criterion 3 of NRC ~~Interim~~ Policy Statement, ~~as described in References 10 and 11.~~

LCO A

Two qualified circuits between the ~~offsite transmission network~~ Watts Bar Hydro 161 KV switchyard and the onsite Class 1E Distribution System; and separate and independent DGs for each train ensure availability of the required power to shutdown the reactor and maintain it in a safe shutdown condition after an anticipated operational occurrence (AOO) or a postulated ~~Design Basis Accident (DBA)~~.

Qualified offsite circuits are ~~those which~~ are described in ~~the~~ FSAR, have been reviewed and accepted by the NRC staff ~~as~~ meeting the requirements of GDC 17 (Ref. 1), and are part of the ~~licensing~~ basis for the unit.

~~[In addition, one required automatic load sequencer per ESF bus shall be OPERABLE.]~~

Each offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the ~~ESF buses.~~ 6.9 KV Shutdown board. ~~[Offsite circuit #1 consists of Safeguards Transformer 'B', which is~~

Offsite power from the Watts Bar Hydro 161-KV switchyard to the onsite Class 1E distribution system is from two independent immediate access circuits. Each of the two circuits are fed from the switchyard through a 161-KV transmission line and 161-to-6.9-KV transformer (common station service transformers) to the onsite Class 1E distribution system. The low and medium voltage power system starts at the high-side of the common station service transformers.

(continued)

It is acceptable for a single circuit to be cross-tied between trains providing the second circuit and associated interrupting devices or protective relaying is OPERABLE. this allows loads from both trains to be supplied from a single offsite source. However, if interrupting devices or protective relaying that normally serves to provide electrical independence between the two circuits are inoperable, it is not acceptable to conclude that offsite circuits are still OPERABLE.

BASES (continued)

(13)

~~supplied from Switchyard Bus 'B', and is fed through breaker 52-3 powering the ESF transformer XNB01, which, in turn, powers the #1 ESF bus through its normal feeder breaker. Offsite circuit #2 consists of the Startup Transformer, which is normally fed from the Switchyard Bus 'A', and is fed through breaker PA 0201 powering the ESF transformer, which, in turn, powers the #2 ESF bus through its normal feed Breaker.]~~

(B) Each DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ~~ESF bus~~ 6.9 KV Shutdown board on detection of ~~bus undervoltage~~ loss-of voltage, and accepting required loads. This will be accomplished within [10] seconds. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the ~~ESF buses~~ 6.9 KV shutdown boards. These capabilities are required to be met from a variety of initial conditions such as: DG in standby ~~hot~~ with the engine hot, DG in standby ~~cold~~ at ambient conditions, and DG operating in parallel test mode. △

(3)

A Note has been added to indicate that the C-S DG may be substituted for any of the required DGs. However, the C-S DG cannot be declared OPERABLE until it is connected electrically in place of another DG, and it has satisfied applicable Surveillance Requirements.

Proper sequencing of loads, [including tripping of non-essential loads,] is a required function for DG OPERABILITY.

The AC sources in one train must be separate and independent (to the extent possible) of the AC sources in the other train. For the onsite ~~diesel generators~~ DGs, the separation and independence is complete.

For the offsite AC sources, the separation and independence is to the extent practical. A circuit may be connected to more than one ~~ESF bus~~, with fast-transfer capability to the other circuit OPERABLE, and not violate separation criteria. A circuit which is not connected to ~~an ESF bus~~, is required to have OPERABLE fast-transfer interlock mechanisms, to at least two ~~ESF buses~~, to support OPERABILITY of that circuit.

APPLICABILITY △ The AC sources ~~[and sequencers]~~ are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of ~~anticipated operational occurrences~~ (AOOs) or abnormal transients; and

(continued)

BASES (continued)

b. Adequate core cooling is provided and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

AC power requirements for MODES 5 and 6 are covered in ~~Specification~~ 3.8.2, "AC Sources – Shutdown."
△ LCO

ACTIONS

A.1

To ensure a highly reliable power source remains with one offsite circuit inoperable, it is necessary to verify the OPERABILITY of the remaining required offsite circuit on a more frequent basis. Since the Required Action only △ specifies "perform~~ance~~," a failure of SR 3.8.1.1 acceptance criteria will not result in a Required Action not met. However, if a second required circuit fails SR 3.8.1.1, the second offsite circuit is inoperable, and Condition C, for two offsite circuits inoperable, is entered.

A.2

DG

Required Action A.2, which only applies if the train cannot be powered from an offsite source, is intended to provide assurance that an event coincident with a single failure of the associated ~~diesel generator~~ DG will not result in a complete loss of safety function of critical redundant required features. The ~~specific list of~~ features are determined by encompassed ~~on~~ Required Action A.2 ~~is provided~~ in accordance with Specification 5.8.1 (SFDP). These features are ~~those which are~~ powered from the redundant AC electrical power trains. This includes motor driven auxiliary feedwater pumps. Single train systems, such as turbine driven auxiliary feedwater pump~~s~~, may not be the included. (11)

Safety Function Determination Program

The Completion Time for Required Action A.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. the train has no offsite power supplying it loads; and

(continued)

BASES (continued)

(11)

b~~2~~. ~~a~~ required feature ~~(see Specification 5.8)~~ on the other train is inoperable.

If at anytime during the existence of Condition A (one offsite circuit inoperable) a redundant required feature subsequently becomes inoperable, this Completion Time would begin to be tracked.

Discovering no offsite power to one train of the onsite Class 1E Power Distribution System ~~AND~~ coincident with one or more inoperable required support or supported features, or both, ~~inoperable~~ that are associated with the other train that has offsite power, results in starting the Completion Times for the Required Action. Twenty-four hours is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

The remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to Train ~~X~~ A and Train ~~Z~~ B of the onsite Class 1E Distribution System. The 24 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 24 hour~~s~~ Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and the low probability of a DBA occurring during this period.

Δ Required features are those that are designed with functionally redundant safety-related trains. If a plant has a required feature that has no functionally related counterpart, that feature may not be required to be included.

A.3

According to ~~Per~~ Regulatory Guide 1.93, ~~"Availability of Electric Power Sources"~~ (Ref. 4), operation may continue in Condition A for ~~a~~ period that should not exceed 72 hours. With one offsite circuit inoperable, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the unit safety systems. In this Condition, however, the remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the onsite Class 1E Distribution System.

(continued)

BASES (continued)

The completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action A.3 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, a DG is inoperable and that DG is subsequently returned OPERABLE, the LCO may already have been not met for up to 72 hours. This could lead to a total of 144 hours, since initial failure of the LCO, to restore the offsite circuit. At this time a DG could again become inoperable, the circuit restored OPERABLE, and an additional 72 hours allowed (for a total of 9 days) prior to complete restoration of the LCO. The 6 day Completion Time provides a limit on the time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The "AND" connector between "72 ~~hr~~ hours" and "6 days" ~~dictates~~ means that both Completion Times apply simultaneously, and the more restrictive must be met.

As in Required Action A.2, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed-outage-time "clock". This will result in establishing the "time zero" at the time ~~the~~ that LCO was initially not met, instead of the time Condition A was entered.

B.1

1 one or more DG(s) inoperable in Train A or Train B

To ensure a highly reliable power source remain with ~~an inoperable DG~~, it is necessary to verify the availability of the offsite circuits on a more frequent basis. Since the Required Action only specifies "perform~~ance~~," a failure of SR 3.8.1.1 acceptance criteria will not result in a Required Action not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions and Required Actions must then be entered. ~~to reflect the new unit state.~~

(continued)

BASES (continued)

B.2

are determined

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that a DG ~~diesel generator~~ is inoperable, will not result in a complete loss of safety function of critical systems. The ~~specific list of~~ redundant required features encompassed by Required Action B.2 ~~is provided~~ in accordance with Specification 5.8, ~~Safety Function Determination Program.~~ These ~~redundant~~ features ~~are those which~~ are designed with redundant safety related trains. This includes motor driven auxiliary feedwater pumps. Single train systems, such as the turbine driven auxiliary feedwater pump~~s~~, are not included. Redundant required feature failures consist of inoperable features ~~(from Specification 5.8)~~ associated with a train redundant to the train which has inoperable DG(s). (11)

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage discovery that both:

a. An inoperable DG ~~diesel generator~~ exists; and

b. A required feature ~~(see Specification 5.8)~~ on the other train ~~(Train A or B)~~ is inoperable. If at any time during the existence of this Condition (one or more DG(s) ~~diesel generator~~ inoperable) a required feature subsequently becomes inoperable, this Completion Time would begin to be tracked. (11)

(1) (S) in Train A or one or more DG(s) in Train B

Discovering one or more required DG(s) inoperable ~~AND~~ coincident with one or more required support or supported features, or both, inoperable that are associated with the OPERABLE DG(s), results in starting the Completion Time for the Required Action. Four ~~(4)~~ hours from the discovery of these events existing concurrently, is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

Required features are those that are designed with functionally redundant safety-related train. If a unit has a required feature that has no functionally redundant counterpart, that feature is not required to be included.

(continued)

BASES (continued)

In this Condition, the remaining OPERABLE DG~~[s]~~ and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. ~~Thus,~~ ~~o~~On a component basis, ~~we have lost~~ single-failure protection for the required feature's function may have been lost; however, ~~we have not lost~~ function has not been lost. The 4 hour Completion Time takes into account the OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and the low probability of a DBA occurring during this period.

B.3.1 and B.3.2

Required Action B.3.1 provides ~~an~~ allowance to avoid unnecessary testing of OPERABLE DGs ~~diesel generators~~. If it can be determined that the cause of the inoperable ~~diesel generator~~ DG(s) does not exists on the OPERABLE ~~diesel generator,~~ DG(s), SR 3.8.1.2 ~~(diesel generator start)~~ does not have to be performed. If the cause of inoperability exists on other ~~diesel generator(s)~~ DGs, the other ~~diesel generator(s)~~ DGs would be declared inoperable upon discovery and Condition E of LCO 3.8.1 ~~would be entered~~. Once the failure is repaired, and the common ~~mode~~ cause failure no longer exists, Required Action B.3.1 is satisfied. If the cause of the initial inoperable ~~diesel generator~~ cannot be confirmed ~~to~~ not to exist on the remaining ~~diesel generator(s)~~ DG(s), performance of SR 3.8.1.2 will suffice to provide assurance of continued OPERABILITY of ~~that/~~those ~~diesel generator(s)~~ DGs.

Δ

~~Per~~ According to Generic Letter 84-15 (Ref. ~~7~~ 9), ~~8~~ 24 hours is ~~a~~ reasonable to confirm that the OPERABLE ~~diesel generator(s)~~ DGs are not affected by the same problem as the inoperable ~~diesel generator~~ DG(s).

~~The Note in~~ Condition B ~~requires~~ is modified by a Note indicating that Required Action B.3.1 or B.3.2 ~~shall~~ must be completed if Condition ~~E~~ B is entered. The intent is that all DG inoperabilities must be investigated for common cause failures regardless of how long the DG inoperability persists.

1 Would be entered if the other inoperable DG(s) are not on the same train, otherwise, if the other inoperable DG(s) are on the same train, the unit remains in Condition B.

(continued)

BASES (continued)

B.4

~~Per~~ According to Regulatory Guide 1.93, (Ref. 4), operation may continue in Condition B for a period that should not exceed 72 hours.

In Condition B, the remaining OPERABLE DG~~[s]~~ES and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. The 72-hour Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action B.4 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an offsite circuit is inoperable and that circuit is subsequently returned OPERABLE, the LCO may already have been not met for up to 72 hours. This ~~could~~ lead to a total (5) of 144 hours, since initial failure ~~of~~ to meet the LCO, to restore the DG. At this time an offsite circuit could again become inoperable, the DG restored OPERABLE, and an additional 72 hours ~~allowed~~ (for a total of 9 days) allowed prior to complete restoration of the LCO. The 6 day Completion Time provides a limit on time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which ~~C~~conditions A and B are entered concurrently. The ~~AND CONNECTOR BETWEEN~~ 72 hours and 6 days ~~dictates~~ that both Completion Times (5) apply simultaneously, means and the more restrictive must be met.

As in Required Action B.2, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed time "clock." This will result in establishing the "time zero" at the time ~~the~~ that LCO was initially not met, instead of the time Condition B was entered.

C.1 and C.2

Required Action C.1, which applies when two offsite circuits are inoperable, is intended to provide assurance that an event with a coincident single failure will not result in a complete loss of redundant required safety functions. The Completion Time for this failure of redundant required

BASES (continued)

(Ref. 4)

features is reduced to 12 hours from that allowed ~~for~~ one train without offsite power (Required Action A.2). The rationale for the reduction to 12 hours is that Regulatory Guide 1.93 allows a Completion Time of 24 hours for two required offsite circuits inoperable, based upon the assumption that two complete safety trains are OPERABLE. When a concurrent redundant required feature failure exists, this assumption is not the case, and a shorter Completion Time of 12 hours is appropriate. The ~~specific list of~~ features encompassed by Required Action C.1 ~~is~~ are determined in accordance with Specification 5.8. ~~Safety Function Determination Program.~~ These features ~~are those which~~ are powered from redundant AC safety trains. This includes motor driven auxiliary feedwater pumps. Single train features, such as the turbine driven auxiliary pump~~s~~, are not included. (11)

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The Completion Time for Required Action C.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action the Completion Time only begins on discovery that both:

a. ~~1.~~ All required offsite circuits are inoperable; and

b. ~~2.~~ A required feature ~~(Specification 5.8)~~ is inoperable. (11)

If at any # time during the existence of Condition C (two offsite circuits inoperable) and a required feature becomes ~~is~~ inoperable, this Completion Time would begin to be tracked.

~~Per~~ According to Regulatory Guide 1.9~~3~~, (Ref. 4), operation may continue in Condition C for a period that should not exceed 24 hours. This ~~degradation~~ level of degradation means that the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the effects of an accident; however, the onsite AC source have not been degraded. This ~~degradation~~ level of degradation generally corresponds to a total loss of the immediately accessible offsite power sources.

Because of the normally high availability of the offsite sources, this level of degradation may appear to be more severe than other combinations of two AC sources inoperable (1) (e.g., combinations that involve an offsite circuit and one DG inoperable, or one or more DG(s) in each train inoperable). ~~that involve one or more DGs inoperable~~. However, two

(continued)

BASES (continued)

factors tend to decrease the severity of this ~~degradation~~ level. of degradation

a. The configuration of the redundant AC electrical power system that remains available is not susceptible to a single bus or switching failure; and

b. The time required to detect and restore an unavailable offsite power source is generally much less than that required to detect and restore an unavailable onsite AC source.

With both of the required offsite circuits inoperable, sufficient onsite AC sources are available to maintain the unit in a safe shutdown condition in the event of a DBA or ~~design basis~~ transient. ~~or accident~~. In fact, a simultaneous loss of offsite AC sources, a LOCA, and a worst-case single failure were postulated as a part of the design basis in the safety analysis. Thus, the 24-hour ~~completion time~~ provides a period of time to effect restoration of one of the offsite circuits commensurate with the importance of maintaining an AC electrical power system capable of meeting its design criteria.

According to ~~Per~~ Reference 4, with the available offsite AC sources, ~~two~~ less than required by the LCO, operation may continue for 24 hours. If two offsite sources are restored within 24 hours, unrestricted operation may continue. If only one offsite source is restored within 24 hours, power operation would continue in accordance with Condition A.

D.1 and D.2

Pursuant to LCO 3.0.6, are the Distribution System ~~Actions~~ Action would not be entered even if all AC sources to it were inoperable resulting in de-energization. Therefore, the Required Actions of Condition D ~~has been~~ modified by a Note to indicate that when Condition D is entered ~~with no AC sources to one train~~ the Conditions and Required Actions for LCO 3.8.9, "Distribution Systems - Operating," must be immediately entered. This allows Condition D to provide requirements for the loss of one offsite circuit and one DG without regard to whether a train is de-energized. LCO 3.8.9 ~~would~~ provides the appropriate restrictions for ~~the situation involving~~ a de-energized train. 18

(continued)

BASES (continued)

~~Per~~ According to Regulatory Guide 1.93, (Ref. 4), operation may continue in Condition ~~E~~ D for a period that should not exceed 12 hours.

In Condition D, individual redundancy is lost in both the offsite electrical power system and the onsite AC electrical power system. Since power system redundancy is provided by two diverse sources of power, however, the reliability of the power systems in this Condition may appear higher than that in Condition C (loss of both required offsite circuits). This difference in reliability is offset by the susceptibility of this power system configuration to a single bus or switching failure. The 12-hour Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and the low probability of a DBA occurring during this period.

E.1

1

With ~~train A and B DGs inoperable,~~ One or more required DG(s) in Train A inoperable simultaneous with one or more required DG(s) in Train B inoperable, there are no remaining standby AC sources. Thus, with an assumed loss of offsite electrical power, insufficient standby AC sources are available to power the minimum required ESF functions. Since the offsite electrical power system is the only source of AC power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

~~Per~~ According to Reference 4, with ~~both DGs inoperable~~, operation may continue for a period that should not exceed 2 hours.

F.1 and F.2

If the inoperable AC electrical power sources cannot be restored to OPERABLE status within the required Completion Time

Δ

The unit must be ~~placed in~~ brought to a MODE in which the LCO does not apply. ~~if the inoperable AC electrical power sources cannot be restored to OPERABLE status within the associated Completion Time. This is done by placing the unit in at~~ To achieve this status, the unit must be brought to at

(continued)

BASES (continued)

least MODE 3 within 6 hours and ~~to~~ to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, ~~related to the amount of time required~~ to reach the required ~~MODES~~ unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(12) ~~G.1 and H.1~~ (14) Cannot be guaranteed

Condition G and Condition H ~~corresponds~~ correspond to a level of degradation in which all redundancy in the AC electrical power supplies ~~has been lost~~ cannot be guaranteed. At this severely degraded level, any further losses in the AC electrical power system will cause a loss of function. Therefore, no additional time is justified for continued operation. The unit is required to commence a controlled shutdown ~~as required~~ by LCO 3.0.3.

~~H.1~~

~~Condition H is one required [automatic load sequencer] inoperable. The Required Action is to restore all required [automatic load sequencers] to OPERABLE status within the Completion Time of [12] hours.~~

~~The sequencer(s) is (are) an essential support system to [both the offsite circuit and the DG associated with a given ESF bus.] [Furthermore, the sequencer is on the primary success path for most major AC electrically powered safety systems powered from the associated ESF bus.] Therefore, loss of an [ESF bus's sequencer] affects every major ESF system in the [division]. The [12]-hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining sequencer OPERABILITY. This time period also ensures that the probability of an accident (requiring sequencer OPERABILITY) occurring during periods where the sequencer is inoperable is minimal.~~

~~This Condition is preceded by a note that allows the Condition to be deleted if the unit design is such that any sequencer failure mode will only affect the ability of the associated diesel generator to power its respective safety loads under any conditions. Implicit in this note is the concept that the Condition must be retained if any sequencer failure mode would result in the inability to start all or~~

(continued)

BASES (continued)

~~part of the safety loads when required, regardless of power availability, or would result in overloading the offsite power circuit to a safety bus during an event and thereby cause its failure. Also implicit in the Note is that the Condition is not applicable to any train which does not have a sequencer.~~

SURVEILLANCE REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with 10 CFR 50 GDC 18 (Ref. ~~10~~ 5). Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions). The SRs for demonstrating the OPERABILITY of the DGs are in accordance with the recommendations of Regulatory Guide 1.9 (Ref. 2), Regulatory Guide 1.108, (Ref. ~~7~~ 6); and Regulatory Guide 1.137, (Ref. ~~8~~ 7), as addressed in the FSAR.

Where the SRs discussed herein specify voltage and frequency tolerances, the following is applicable. The minimum steady-state output voltage of ~~[3740]~~ 6600 V is 90% of the nominal ~~4160V output voltage~~ 6900 (Ref. 11) voltage. ~~This value, which is specified in ANSI C84.1-1982, allows for voltage drop to the terminals of 4000V motors whose minimum operating voltage is specified as 90% or 3600V. It also allows for voltage drops to motors and other equipment down through the 120V level where minimum operating voltage is also usually specified as 90% of name plate rating.~~ The specified maximum steady-state output voltage of ~~[4756]~~ 7180 V is equal to the maximum operating voltage specified for 4000V motors. It ensures that for a lightly loaded distribution system, the voltage at the terminals of 4000V motors ~~will be~~ is no more than the maximum rated operating voltages. The specified minimum and maximum ~~frequency~~ frequencies of the DG is 58.8 ~~Hz~~ Hz and 61.2 ~~Hz~~ Hz respectively. ~~This is~~ These values are equal to ± 2% of the 60 ~~Hz~~ Hz nominal frequency and is derived from the recommendations given in ~~Regulatory Guide 1.9 (Ref. 2).~~ Reference 2.

20

of [TBD] is [Bases to be provided].

1970

INSERT

E

SR 3.8.1.1

This SR assures proper circuit continuity for the offsite AC electrical power supply to the onsite distribution network and availability of offsite AC electrical power. The breaker

(continued)

INSERT

The specified minimum transient value of 6555 V is 95% of the nominal bus voltage of 6900 V and is the minimum voltage required for the DG supply breaker to close ~~to~~ on the 6.9 Kv Shutdown Board. The specified maximum transient value of 7260 V is 110% of the nameplate rating of 6600 V motors.

BASES (continued)

that ∆

alignment verifies that each breaker is in its correct position to ensure distribution buses and loads are connected to their preferred power source, and independence of offsite circuits is maintained. The 7-day Frequency is adequate since breaker position is not likely to change without the operator being aware of it and its status is displayed in the control room.

∆ that appropriate

SR 3.8.1.2 and SR 3.8.1.7

∆ DBAs and

These SRs help to ensure the availability of the standby electrical power supply to mitigate ~~design basis~~ transients ~~and accidents~~ and maintain the unit in a safe shutdown condition.

(b) ~~To minimize the wear on moving parts that do not get lubricated when the engine is not running, these SRs have been modified by a Note (Note 2 for SR 3.8.1.2, Note 1 for SR 3.8.1.7) to indicate that all DG starts for these Surveillances may be preceded by an engine prelube period and followed by a warmup period prior to loading.~~

The DG engines for WBN have an oil circulation [illegible] soakback system [illegible] operates continuously to preclude the need for a prelube and warmup when a DG is started from standby.

For the purposes of ~~the~~ SR 3.8.1.2 and SR 3.8.1.7 testing, the DGs shall be started from standby conditions. Standby conditions for a DG means the diesel engine coolant and oil are being continuously circulated and temperature maintained consistent with manufacturer recommendations.

In order to reduce stress and wear on diesel engines ~~some~~ the manufacturer~~s~~ recommends a modified start in which the starting speed of DGs is limited, warmup is limited to this lower speed, and the DGs are gradually accelerated to synchronous speed prior to loading. This is the intent of ~~Note 1~~ 2 which is only applicable when such modified start procedures are recommended by the manufacturer. (A)

Start

SR 3.8.1.7 requires that, on a 184-day Frequency, the DG start from standby conditions and achieve required voltage and frequency within 10 seconds. The 10-second requirement supports the assumptions in the design basis LOCA analysis (Ref. 8).

The 10-second start requirement is not applicable to SR 3.8.1.2 (See Note 2) when a modified start procedure as described

(continued)

BASES (continued)

above is used. If a modified start is not used, 10-second start requirement of SR ~~3.8.7~~ 3.8.1.7 applies.

Since SR 3.8.1.7 ~~does~~ requires a 10-second start, it is more restrictive than SR 3.8.1.2, and it may be performed in lieu of SR 3.8.1.2. This is the intent of Note 1 of SR 3.8.1.2.

Diesel Generator Test Schedule in the accompanying LCO. △

The normal 31-day Frequency for SR 3.8.1.2 (see ~~DG test schedule,~~ Table 3.8.1-1) is consistent with Regulatory Guide 1.9 (Ref. 2). The 184-day Frequency for SR 3.8.1.7 is a reduction in cold testing consistent with Generic Letter 84-15 (Ref. ~~X~~ 9). These Frequencies provide adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing.

SR 3.8.1.3

This surveillance ~~verifies~~ that the DGs are capable of synchronizing with the offsite electrical system and accepting loads greater than or equal to the equivalent of the maximum expected accident loads. A minimum run time of 60 minutes is required to stabilize engine temperatures, while minimizing the time that the DG is connected to the offsite source.

In order to ~~assure~~ ensure that the DG is tested under load conditions that are as close to design basis conditions as possible, testing shall be performed using a power factor ~~of~~ ≤ ~~[0.9]~~ 0.8. This power factor ~~range~~ is chosen to be representative of the actual design basis inductive loading that the DG would experience. The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

(A)

The normal 31-day Frequency for this Surveillance ~~(see DG test schedule,~~ (Table 3.8.1-1) ~~is~~ consistent with Regulatory Guide 1.9 (Ref. 2).

Note 1 modifies this Surveillance to indicate that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized (Ref. ~~10~~ 9).

(continued)

BASES (continued)

Note 2 modifies this Surveillance by stating that momentary transients because of changing bus loads do not invalidate this test. Similarly, momentary Power Factor transients above the limit will not invalidate the test.

Note 3, indicates that only this Surveillance should ~~only~~ be conducted on one DG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations.

Note 4 stipulates a prerequisite requirement for performance of this SR. A successful DG start must precede this test to credit satisfactory performance.

SR 3.8.1.4

B

This SR provides verification that the level of fuel oil in each DG ~~the~~ day tank ~~[and engine mounted tank]~~ is at or above the level at which fuel oil is automatically added. The level is ~~normally~~ expressed as an equivalent volume in gallons, and is selected to ensure adequate fuel oil for a minimum of 1 ~~one~~ hour of DG ~~Diesel Generator~~ operation at full load plus 10%.

Skid-mounted

The 31-day Frequency is adequate to assure ~~ensure~~ that a sufficient supply of fuel oil is available, since low-level alarms are provided and facility operators would be aware of any large uses of fuel oil during this period.

SR 3.8.1.5

Microbiological fouling is a major cause of fuel-oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel-oil day ~~[and engine mounted]~~ tanks once every 31 days will eliminate the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it will eliminate the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water will

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BASES (continued)

minimize fouling as well as provide data regarding the watertight integrity of the fuel-oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 7). This SR is for preventative maintenance. The presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed during the performance of this surveillance.

SR 3.8.1.6

B

This Surveillance demonstrates that each required fuel-oil transfer pump operates and transfers fuel oil from its associated storage tank to its associated skid mounted day tank. This is required to support continuous operation of standby power sources. This Surveillance provides assurance that the fuel-oil transfer pump is OPERABLE, the fuel-oil piping system is intact, the fuel-delivery piping is not obstructed, and the controls and control systems for automatic fuel transfer systems are OPERABLE. ¶ ~~[~~The Frequency for this SR is 92 days ~~variable, depending on individual system design, with up to a 92 day interval.~~ The 92-day Frequency corresponds to the testing requirements for pumps as contained in the ASME Section XI code. ~~; however, the design of fuel transfer systems is such that pumps will operate automatically or must be started manually in order to maintain an adequate volume of fuel oil in the day [and engine mounted] tanks during or following DG testing. In such a case a 31 day Frequency is appropriate. Since proper operation of fuel transfer systems is an inherent part of DG OPERABILITY, the Frequency of this SR should be modified to reflect individual designs.]~~

SR 3.8.1.7

See SR 3.8.1.2.

SR 3.8.1.8 B

Transfer of each 6.9 kV shutdown board ~~[4.16 kV ESF bus]~~ power supply from the normal offsite circuit to the alternate offsite circuit demonstrates the OPERABILITY of the alternate circuit distribution network to power the shutdown loads. The ~~[~~18-month~~]~~ Frequency of the Surveillance is based on

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BASES (continued)

engineering judgment taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel-cycle lengths. Operating experience has shown that these components usually pass the SR when performed on the [18-month] Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is ~~has been~~ modified by two Notes ~~Note 1, which states that the SR must not be performed in MODE 1 or 2.~~ The reason for Note 1 ~~this~~ is that during operation with the reactor critical, performance of ~~this SR~~ Surveillance could ~~potentially~~ cause perturbations to the electrical distribution systems that ~~could result in a~~ challenge ~~to~~ continued steady-state operation and, as a result, ~~to~~ unit safety systems.

1 Since the DGs are shared between the two units, the requirement of Note 1 cannot be accomplished completely without taking both units out of operation. Therefore, DG testing will be performed when the unit associated with the DG is not in MODE 1 or 2, and the appropriate ACTIONS will be entered for the other operating unit.

Note 2 is ~~has been~~ included in this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.9

Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the DG load response characteristics and capability to reject the largest single load without exceeding predetermined voltage and frequency and while maintaining a specified margin to the overspeed trip. ~~[For this unit, the single load for each DG and its horsepower rating is as follows:]~~ As required by IEEE-308, (Ref. 12) the load rejection test is acceptable if the increase in diesel speed does not exceed 75% of the difference between synchronous speed and the overspeed trip setpoint, or 15% above synchronous speed, whichever is lower.

15 , while maintaining the remaining loads (e.g., the normal 480 VAC shutdown board loads),

1 The largest single load for each DG is the essential raw cooling water pump at 800 HP.

The time, voltage, and frequency tolerances specified in this SR are derived from Regulatory Guide 1.9 (Ref. 2) recommendations for response during load sequence intervals. The 3 seconds specified is equal to 60% of a typical 5-second load sequence interval associated with sequencing of the largest load. The voltage and frequency specified are consistent with the design range of the equipment

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BASES (continued)

powered by the DG. SR 3.8.1.9a corresponds to the maximum frequency excursion, while SR 3.8.1.9b and SR 3.8.1.9c are steady-state voltage and frequency values ~~that~~ the system must recover ~~to~~ following load rejection. The [18-month] Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. ~~X~~ 6).

In order to ~~assure~~ ensure that the DG is tested under load conditions that are as close to design basis conditions as possible, testing shall be performed using a power factor ≤ [0.9]. This power factor ~~range~~ is chosen to be representative of the actual design basis inductive loading that the DG would experience. (A)

This SR ~~has been~~ is modified by two Notes ~~Note 1, which states that the SR must not be performed in MODE 1 or 2~~. Note 1 The reason for ~~this~~ is that during operation with the reactor critical, performance of this ~~SR~~ Surveillance could ~~potentially~~ cause perturbations to the electrical distribution systems that ~~could result in a~~ challenge ~~to~~ continued steady-state operation and, as a result, ~~to~~ unit safety systems.

Since the DGs are shared between the two units, the requirement of Note 1 [illegible]not be accomplished [illegible]pletely without taking both units out of operation. Therefore, DG testing will be performed when the unit associated with the DG is not in MODE 1 or 2, and appropriate ACTIONS will be entered for the other operating unit.

Note 2 ~~has been added~~ is included to this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.10

This Surveillance demonstrates the DG capability to reject a full load without overspeed tripping or exceeding the predetermined voltage limits. The DG full-load rejection may occur because of a system fault or inadvertent breaker tripping. This Surveillance ~~verifies~~ ensures proper engine-generator load response under the simulated test conditions. This test ~~will~~ simulates the loss of the total connected load that the DG ~~will~~ experiences following a full-load rejection and ~~verify~~ verifies that the DG ~~will~~ not trip upon loss of the load. These acceptance criteria provide ~~for~~ DG damage protection. While the DG is not expected to experience this transient during an event and continue to be available, this response ~~will assure~~ ensures that the DG is not degraded for future application, including reconnection to the bus if the trip initiator can be corrected or isolated.

In order to ~~assure~~ ensure that the DG is tested under load conditions that are as close to design basis conditions as

(continued)

BASES (continued)

Since the DGs are shared between the two units, the requirement of Note 1 cannot be accomplished completely without taking both units out of operation. Therefore, DG testing will be performed when the unit associated with the DG is not in MODE 1 or 2, and the appropriate ACTIONS will be entered for the operating unit.

possible, testing shall be performed using a power factor (B) ≤ 0.9~~8~~. This power factor ~~range~~ is chosen to be representative of the actual design basis inductive loading that the DG would experience.

The 18-month Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. ~~X~~ 6) and is intended to be consistent with expected fuel-cycle lengths.

This SR ~~has been~~ is modified by two Notes. ~~Note 1, which states that the SR must not be performed in MODE 1 or 2.~~ Note 1 The reason for ~~this~~ is that during operation with the reactor critical, performance of this ~~SR~~ Surveillance could ~~potentially~~ cause perturbations to the electrical distribution systems that ~~could result in a~~ challenge ~~to~~ continued steady-state operation and, as a result, unit safety systems. Note 2 ~~has been~~ is added to this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.11

As required by Regulatory Guide 1.108 (Ref. ~~X~~ 6), paragraph 2.a.(1), this Surveillance demonstrates the as-designed operation of the standby power sources during loss of the offsite source. This test verifies all actions encountered from the loss of offsite power, including shedding of the non-essential loads and energization of the emergency buses and respective loads from the DG. It further demonstrates the capability of the DG to automatically achieve the required voltage and frequency within the specified time.

The DG auto~~matic~~ start time of 10 seconds is derived from requirements of the accident analysis to respond to a design basis large-break LOCA. The frequency should be restored to within 2% of nominal following a load sequence step. The Surveillance should be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability has been achieved.

The requirement to verify the connection and power supply of permanent and auto-connected loads is intended to satisfactorily show the relationship of these loads to the DG loading logic. In certain circumstances, many of these loads can not actually be connected or loaded without undue

(continued)

BASES (continued)

Emergency Core Cooling System

loads, testing, that adequately shows the capability of the DG system to perform these functions,

hardship or potential for undesired operation. ~~for~~ to instance, ECCS injection valves are are desired ~~to~~ not ~~be~~ stroked open, or high pressure injection systems are not capable of being operated at full flow, or RHR systems performing a decay heat removal function not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of ~~the~~ connection and loading of ~~these functions~~ is ~~acceptable~~. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The DG engines for WBN have an oil circulation and soakback system that operates continuously to preclude the need for a prelube and warmup when a DG is started from standby.

The Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.108 (Ref. ~~X~~ 6), paragraph 2.a.(1), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel-cycle lengths.

This ~~SR~~ is ~~has been~~ modified by two Notes. ~~Note 1, stating that all DG starts may be preceded by an engine prelube period to minimize wear and tear on the DGs during testing.~~ For the purpose of this testing, ~~the DGs shall be started from~~ standby conditions, that is, ~~with the engine coolant and oil being continuously circulated and temperature maintained consistent with~~ manufacturer ~~recommendations~~.

6

for Note 1

Since the DGs are shared between the two units, the requirement of Note 1 cannot be accomplished completely without taking both units out of operation. therefore, DG testing will be performed when the unit associated with the DG is not in MODE 1,2,3, or 4 and the appropriate ACTIONS will be entered for the other operating unit.

~~This SR has been modified by Note 2, which states that the SR must not be performed in MODE 1, 2, 3, or 4.~~ Surveillance The reason ~~for this~~ is that performing the ~~SR~~ would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

Note ~~3. has been~~ 2 is added to this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.12

This Surveillance demonstrates that the DG automatically starts and achieves the required voltage and frequency within the specified time (~~[~~10~~]~~ seconds) from the design basis actuation signal (LOCA signal) and operates for ≥ 5 minutes. The 5-minute period provides sufficient time to demonstrate stability. SR 3.8.1.12d and SR 3.8.1.12e ensure that permanently connected loads and emergency loads are energized from the offsite electrical power system on an ESF signal without loss of offsite power.

(continued)

BASES (continued)

loads, testing, that adequately shows the capability of the DG system to perform those functions,

to

are

The requirement to verify the connection of permanent and auto-connected loads is intended to satisfactorily show the relationship of these loads to the DG loading logic. In certain circumstances, many of these loads can not actually be connected or loaded without undue hardship or potential for undesired operation. For instance, ECCS injection valves are desired ~~to~~ not be stroked open, or high pressure injection systems not capable of being operated at full flow, or RHR systems performing a decay heat removal function not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of ~~the~~ connection and loading of ~~these functions~~ is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

6

The DG engines for WBN have an oil circulation and soak back system that operates continuously to preclude the need for a prelube and warmup when a DG is started from standby.

This SR ~~has been~~ is modified by two Notes. ~~Note 1, which states that all DG starts may be preceded by an engine prelube period to minimize wear and tear on the DGs during testing.~~ For the purpose of this testing, the DGs shall be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations.

1

Since the DGs are shared between the two units, the requirement of Note 1 cannot be accomplished completely without taking both units out of operation. Therefore, DG testing will be performed when the unit associated with the DG is not in MODE 1 or 2 and the appropriate ACTIONS will be entered for the other operating unit.

~~This SR has been modified by Note 2, which states that the SR must not be performed in MODE 1 or 2.~~ The reason for Note 1 ~~this~~ is that during operation with the reactor critical, performance of this Surveillance ~~SR~~ could ~~potentially~~ cause perturbations to the electrical distribution systems that ~~could result in a~~ challenge ~~to~~ continued steady-state operation and, as a result, ~~to~~ unit safety systems.

Note ~~3 has been~~ 2 is added to this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

The Frequency of 18 months takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with the expected fuel-cycle lengths. Operating experience has shown that these components usually pass the SR when performed ~~on the~~ at 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

15 Verification of the DG trip/trip bypass functions should be done by testing of the controls circuits only; running of the DG during adverse conditions (engine overspeed, generator differential current ~~high temperature~~) is not prudent or necessary.

BASES (continued)

SR 3.8.1.13

This Surveillance demonstrates that DG non-critical protective functions (e.g., high jacket water temperature) are bypassed on a loss-of-voltage signal concurrent with an ESF actuation test signal and critical protective functions (engine overspeed and generator differential current) ~~and low lubricating oil pressure)~~ trip the DG to avert substantial damage to the DG unit. The non-critical trips are bypassed during DBAs and provide an alarm on an abnormal engine condition. This provides the operator with sufficient time to react appropriately. The DG availability to mitigate the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the DG.

alarm

1 Since the DGs are shared between the two units, the requirement of Note 1 cannot be accomplished completely without taking both units out of operation. therefore, DG testing will be performed when the unit associated with the DG is not in MODE 1, 2, 3, or 4 and the appropriate ACTIONS will be entered for the other operating unit.

The 18-month Frequency is based on engineering judgment, taking into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel-cycle lengths. Operating experience has shown that these components usually pass the SR when performed ~~on~~ at the [18-month] Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The SR ~~has been~~ is modified by two Notes. ~~Note 1, which states that the SR must not be performed in MODE 1, 2, 3, or 4.~~ Note 1 The reason for ~~this~~ is that performing the ~~SR~~ surveillance would remove a required DG ~~from service.~~

Note 2 ~~has been~~ is added to this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.14

Regulatory Guide 1.108 (Ref. ~~X~~ 6), paragraph 2.a.(3), requires demonstration once per 18 months that the DGs can start and run continuously at full-load capability for an interval of not less than 24 hours, ≥ 2 hours of which is at a load equivalent to 110% of the continuous duty rating and the remainder of the time at a load equivalent to the continuous duty rating of the DG.

~~The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent~~

(continued)

The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

BASES (continued)

~~teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.~~ The DG starts for this Surveillance can be performed either from standby or hot conditions. The provisions for prelub~~ricating~~ (e) and warmup, discussed in SR 3.8.1.2, and for gradual loading, discussed in SR 3.8.1.3, are applicable to this SR.

In order to assure that the DG is tested under load conditions that are as close to design conditions as possible, testing shall be performed using a power factor of ≤ ~~[0.9]~~ 0.8 (A). This power factor ~~range~~ is chosen to be representative of the actual design basis inductive loading that the DG would experience.

1

Since the DGs are shared between the two units, the requirements of Note 2 cannot be [illegible]mplished completely [illegible]hout taking both units out of operation. Therefore, DG testing will be performed when the unit associated with the DG is not in MODE 1 or 2 and the appropriate ACTIONS will be entered for the other operating unit.

The 18-month Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 7), paragraph 2.a.(3), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel-cycle lengths.

This SR is ~~Surveillance has been~~ modified three Notes. ~~by Note 1, which~~ Note 1 states that momentary transients due to changing bus loads do not invalidate this test. Similarly, momentary ~~PF~~ power factor transients above the ~~PF~~ power factor limit will not invalidate the test.

~~This SR has been modified by Note 2, which states that the SR must not be performed in MODE 1 or 2.~~ The reason for Note 2 ~~this~~ is that during operation with the reactor critical, performance of this ~~SR~~ Surveillance could ~~potentially~~ cause perturbations to the electrical distribution systems that ~~could result in a~~ challenge ~~to~~ continued steady-state operation and, as a result, ~~to~~ unit safety systems.

Note 3 ~~has been added to~~ is included in this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.15

This Surveillance demonstrates that the diesel engine can restart from a hot condition, such as subsequent to shutdown from normal Surveillances, and achieve the required voltage and frequency within 10 seconds. The 10-second time is derived from the requirements of the accident analysis to respond to a design basis large-break LOCA. ~~The requirement that the diesel have operated for at least 2 hours at~~

(continued)

Δ The requirement that the diesel have operated for at least 2 hours at full-load conditions prior to performance of this Surveillance is based on manufacturers recommendations for achieving hot conditions.

Δ BASES (continued)

The load band is provided to avoid routine overloading of the DG. Routine overloads may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

~~full-load conditions prior to performance of this Surveillance is based on manufacturer's recommendations for achieving hot conditions.~~

~~The Surveillance demonstrates the DG capability to respond to an accident signal while hot, such as subsequent to shutdown from normal Surveillances.~~ The 18-month Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. ~~7~~ 6), paragraph 2.a.(5).

This SR ~~has been~~ is modified by ~~Note 1~~ a Note. the reason for the Note is to ensure that the test is performed with the diesel sufficiently hot. Momentary transients due to changing bus loads do not invalidate this test.

The DG engines for WBN have an oil circulation and soak back system that operates continuously to preclude the need for a prelube and warmup when a DG is started from standby.

~~This SR has also been modified by Note 2, which states that all DG starts may be preceded by an engine prelube period to minimize wear and tear on the diesel during testing.~~

6

SR 3.~~8~~8.1.16

As required by Regulatory Guide 1.108 (Ref. ~~7~~ 6), paragraph 2.a.(6), this Surveillance ~~assures~~ ensures that the manual synchronization and automatic load transfer from the DG to the offsite source can be made and that the DG can be returned to ready-to-load status when offsite power is restored. It also ensures that the auto-start logic is reset to allow the DG to reload if a subsequent loss of offsite power occurs. The DG is considered to be in ready-to-load status when the DG is at rated speed and voltage, the output breaker is open and can receive and auto-close signal on bus undervoltage, and the load sequence timers are reset.

The Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.108 (Ref. ~~7~~ 6), paragraph 2.1.(6), and takes into consideration unit conditions required to perform the Surveillance.

This SR ~~has been~~ is modified by ~~Note 1, which states that the SR must not performed in MODE 1, 2, 3, OR 4.~~ two Notes. Note 1 The reason for ~~this~~ is that performing the ~~SR~~ Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

1

Since the DGs are shared between the two units, the requirement of Note 1 cannot be accomplished completely without taking both units out of operation. Therefore, DG testing will be performed when the unit associated with the DG is not in MODE 1, 2, 3, or 4 and the appropriate ACTIONS will be entered for the other operating unit.

(continued)

BASES (continued)

add to the last paragraph on the previous page.

Note 2 ~~has been~~ is added to this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

A

SR 3.8.1.17

Demonstration of the test mode override ensures that the DG availability under accident conditions will not be compromised as the result of testing and the DG will automatically reset to ready-to-load operation if a LOCA actuation signal is received during operation in the test mode. Ready-to-load operation is defined as the DG running at rated speed and voltage with the DG output breaker open. These provisions for automatic switchover are required by IEEE-308 (Ref. 9), paragraph 6.2.6(2).

The requirement to automatically energize the emergency loads with offsite power is essentially identical to that of SR 3.8.1.12. The intent in the requirement associated with SR 3.8.1.18.b is to show that the emergency loading was not affected by the DG operation in test mode. In lieu of actual demonstration of the connection and loading of these loads, testing which adequately shows the capability of the emergency loads to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The 18-month Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 7), paragraph 2.a.(8), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel-cycle lengths.

This SR has been modified by Note 1, which states that the SR must not be performed in MODE 1, 2, 3, or 4. The reason for this is that performing the SR would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

Note 2 has been added to this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

(continued)

BASES (continued)

SR 3.8.1.~~18~~ 17

As required by Regulatory Guide 1.108 (Ref. ~~8~~ 6), paragraph 2.a.(2), each DG is required to demonstrate proper operation for the DBA loading sequence to ensure that voltage and frequency are maintained within the required limits. Under accident conditions, prior to connecting the DGs ~~diesel generators~~ to their respective 6.9KV Shutdown board ~~appropriate bus,~~ all loads are shed except load center feeders and those motor control centers that power Class 1E loads (referred to as "permanently connected" loads). Upon reaching A minimum of 95% ~~90%~~ of rated voltage and frequency, the DGs are then connected to their respective 6.9KV Shutdown board ~~bus.~~ Loads are then sequentially connected to the ~~bus~~ by the ~~[~~automatic load sequencer~~]~~. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading of the DGs due to high motor-starting currents. The ~~[10%]~~ load-sequence time interval specified in FSAR Table 8.3-3 ~~tolerance~~ ensures that sufficient time exists for the DG to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding ESF equipment time delays are not violated. Reference ~~8~~ 3 provides a summary of the automatic loading of ~~ESF buses~~ 6.9 KV Shutdown boards.

The Frequency of ~~[~~18 months~~]~~ is consistent with the recommendations of Regulatory Guide 1.108 (Ref. ~~8~~ 6), paragraph 2.a.(2), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel-cycle lengths.

This SR ~~has been~~ is modified by two Notes ~~Note 1, which states that the SR must not be performed in MODE 1, 2, 3, or 4~~. The reason for Note 1 ~~this~~ is that performing the Surveillance ~~SR~~ would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Since the DGs are shared between the two units, the requirement of Note 1 cannot be accomplished completely without taking both units out of operation. Therefore, DG testing will be performed when the unit associated with the DG is not in MODE 1, 2, 3 or 4 and the appropriate ACTIONS will be entered for the other operating unit.

Note 2 ~~has been~~ is added to this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.~~19~~ 18

In the event of DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.

(continued)

BASES (continued)

6

The DG engines for WBN have an oil circulation and soakback system that operates continuously to preclude the need for a prelube and warmup when a DG is started from standby.

in the Bases for

This Surveillance demonstrates the DG operation, as discussed ~~under~~ in the Bases for SR 3.8.1.11 ~~above,~~ during a loss of offsite power actuation test signal in conjunction with an ESF actuation signal. In lieu of actual demonstration of the connection and loading of ~~these~~ loads, testing ~~which~~ that adequately shows the capability of the DG ~~and sequencer(s)~~ system to perform these functions, is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of [18 months] takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with an expected fuel-cycle length of [18 months].

Since the DGs are shared between the two units, the requirement of Note 1 cannot be accomplished completely without taking both units out of operation. Therefore, DG testing will be performed when the unit associated with the DG is not in MODE 1, 2, 3, or 4 and the appropriate ACTIONS will be entered for the other operating unit.

This SR ~~has been~~ is modified by two Notes. ~~Note 1, which states that all DG starts may be preceded by an engine prelube period to minimize wear and tear on the DGs during testing.~~ For the purpose of this testing, the DGs ~~shall~~ must be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations for DGs.

~~This SR has been modified by Note 2, which states that the SR must not be performed in MODE 1, 2, 3, or 4.~~ Note 1 The reason for ~~this~~ is that the performance of the ~~SR during these modes~~ Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

Note ~~3 has been~~ 2 is added to this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

1

SR 3.8.1.~~20~~ 19

This Surveillance demonstrates that the DG starting independence has not been compromised. Also, this Surveillance demonstrates that each engine can achieve proper speed within the specified time when the DGs are started simultaneously.

The 10-year Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. ~~X~~ 6), paragraph 2.b. ~~and Regulatory Guide 1.137 (Ref. X), paragraph C.2.f.~~ 7

A

(continued)

6 BASES (continued)

The DG engines for WBN have an oil circulation and soakback system that operates continuously to preclude the need for a prelube and warmup when a DG is started from standby.

~~This SR has been modified by a Note that all DG starts may be preceded by an engine prelube period to minimize wear on the DG during testing.~~ For the purpose of this testing, the DGs must ~~shall~~ be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations.

8 Insert additional SR's found on the following pages

~~DG~~ Diesel Generator Test Schedule

The DG test schedule (Table 3.8.1-1) implements the recommendations of Revision 3 to Regulatory Guide 1.9 (Ref. 2). The purpose of this test schedule is to provide ~~sufficiently~~ timely ~~that is~~ test data to establish a confidence level associated with the goal to maintain DG reliability above ~~0.95~~ 16 0.975 per demand. test

According to ~~Per~~ Regulatory Guide 1.9, Revision 3, each DG unit should be tested at least once every 31 days. Whenever a DG has experienced 4 ~~four~~ or more valid failures in the last 25 valid tests, the maximum time between tests is reduced to 7 days. Four failures in 25 valid tests is a failure rate of 0.16, or the threshold of acceptable DG performance, and hence may be an early indication of the degradation of DG reliability. When considered in the light of a long history of tests, however, 4 ~~four~~ failures in the last 25 valid tests may only be a statistically probable distribution of random events. Increasing the test Frequency will allow for a more timely accumulation of additional test data upon which to base judgment of the reliability of the DG. The increased test Frequency must be maintained until seven consecutive, failure-free tests have been performed.

The frequency for accelerated testing is 7 days, but not less than 24 hours. Therefore, the interval between tests should not be less than 24 hours, and no~~t~~ more than 7 days. A successful test at an interval of less than 24 hours should be considered an invalid test and not count towards the seven consecutive failure free starts. A test interval in excess of 7 days constitutes a failure to meet SRs. ~~Surveillance Requirements.~~

(continued)

INSERT

SR 3.8.1.20

This SR demonstrates that the DG 125V DC distribution panel and associated charger are functioning properly, with all required circuit breakers closed and buses energized from normal power. The 7 day Frequency takes into account the redundant DG capability and other indications available in the control room that will alert the operator to system malfunctions.

SR 3.8.1.21

Verifying battery terminal voltage while on float charge for the DG batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery (or battery cell) and maintain the battery (or a battery cell) in a fully charged state. The voltage requirements are based on the critical cell voltage as specified by the manufacturer. The 7 day Frequency is consistent with the manufacturers recommendations and IEEE-450 (Ref. 13).

SR 3.8.1.22

Visual inspections to detect corrosion of the battery cells and connections, or measurement of the resistance of each inter-cell, inter-rack, inter-tier, and terminal connections, provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance. The 92-day Frequency is considered acceptable based on operating experience related to detecting corrosion trends.

SR 3.8.1.23

Visual inspection of the battery cells, cell plates, and battery racks provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance. This SR is consistent with IEEE-450 (Ref.13), which recommends detailed visual inspection of cell conditions and rack integrity on a yearly basis.

SR 3.8.1.24 and SR 3.8.1.25

Visual inspection and resistance measurements of inter-cell, inter-rack, inter-tier, and terminal connections provides an indication of physical damage or abnormal deterioration that could indicate degraded battery condition. The anti-corrosion material is used to help ensure good electrical connections and to reduce terminal deterioration. The visual inspection for corrosion is not intended to require removal of and inspection under each terminal connection. For the purposes of trending, inter-cell and inter-tier connections are measured from battery post to battery post and inter-rack and terminal connections are measured from the terminal lug to battery post. The Surveillance Frequency of 12 months is consistent with IEEE-450 (Ref. 13), which recommends cell-to-cell and terminal connection resistance measurement on a yearly basis.

SR 3.8.1.26

A battery-service test is a special test of battery capability, "as found," to satisfy the design requirements (battery duty cycle) of the DG battery subsystem. The discharge rate and test length corresponds to the design duty cycle requirements as specified in Reference 14. The Surveillance Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 15) and Regulatory Guide 1.129 (Ref. 16), which state that the battery-service test should be performed during refueling operations or at some other outage, with intervals between tests not to exceed 18 months.

This SR is modified by three Notes. The reason for Note 1 is to allow the once-per-60-months performance of SR 3.8.1.27 in lieu of SR 3.8.1.26. This substitution is acceptable because SR 3.8.1.27 represents a more severe test of battery capacity than SR 3.8.1.26. The reason for Note 2 is that performing the Surveillance would remove a required DG battery subsystem from service. Note 3 is added to this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.27

A battery-performance test is a test of constant current capacity of a battery after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 13) and IEEE-485 (Ref. 17). These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

The Surveillance Frequency for this test is 60 months, or every 12 months if the battery shows degradation or has reached 85% of its expected life. However, if the battery has reached 85% of its expected life, shows no degradation, and retains 100% capacity, a Surveillance Frequency of 24 months is permitted. Degradation is indicated, according to IEEE-450 (Ref. 13), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is below the manufacturer's rating. All these Frequencies are consistent with the recommendations in IEEE-450 (Ref. 13).

This SR is modified by two Notes. The reason for Note 1 is that performing the Surveillance would remove a required DG battery subsystem from service. Note 2 is added to this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

BASES (continued)

REFERENCES

SEE NEXT PAGE FOR 3.

1. 10 CFR 50, Appendix A, GDC 17, ~~Title 10, Code of Federal Regulations, Part 50, General Design Criterion 17,~~ "Electric Power Systems."

~~2. FSAR, Chapter 8, "Electrical".~~

4 ~~3~~. Regulatory Guide 1.93, Rev. 0, "Availability of Electric Power Sources," December 1974.

2 ~~4~~. Regulatory Guide 1.9, Rev. 3, "Selection, Design, and Qualification of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Units," ~~[date]~~. January 1992. (Draft)

~~5. 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."~~

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

5 ~~7~~. 10 CFR 50, Appendix A, GDC 18, ~~Title 10, Code of Federal Regulations, Part 50, General Design Criterion 18,~~ "Inspection and Testing of Electric Power Systems."

6 ~~8~~. Regulatory Guide 1.108, Rev. 1, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," August 1977.

7 ~~9~~. Regulatory Guide 1.137, Rev. 1 "Fuel Oil Systems for Standby Diesel Generators," ~~date.~~ October 1979

8 ~~10~~. Watts Bar FSAR, Section 15.4, "Condition IV - Limiting Faults." ~~15, Accident Analysis.~~

9 ~~11~~. Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," July 2, 1984.

10. Watts Bar FSAR, Section 6, "Engineered Safety Features."

(continued)

BASES (continued)

12. IEEE Standard 308-1978, "IEEE Standard Criteria for Class IE Power Systems for Nuclear Power Generating Stations."

3. Watts Bar FSAR, Section 8.2, "Offsite (preferred) Power System," and Tables 8.3-1 to 8.3-3, "Safety Related Standby (Onsite) Power Sources and Distribution Boards," "Shutdown Board Loads Automatically Stripped Following a Loss of Nuclear Unit and Preferred (offsite) Power," and "Diesel Generator Load Sequentially Applied Following a Loss of Nuclear Unit and Preferred (offsite) Power."

11. ANSI C84.1-1982

13. IEEE-450-1980, "IEEE Recommended Practice for Maintenance Testing and Replacement of Large Lead Storage Batteries for Generating Stations and Subsystems."

14. Watts Bar FSAR, Section 8.3.1.1, ~~AC Power~~ "Standby AC Power System."

15. Regulatory Guide 1.32, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants," February 1977, U.S. Nuclear Regulatory Commission.

16. Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Generating Stations and Subsystems," U.S. Nuclear Regulatory Commission, December 1974.

17. IEEE-485-1978, "Recommended Practices for sizing Large Lead Storage Batteries for Generating Stations and Subsystems."

(continued)

JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.8.1

Generic Justifications

A. Modified to reflect WBN plant specific design.

B. Modified to reflect WBN plant specific terminology.

Specific Justifications

1. Modified to reflect WBN plant specific design of onsite power system. Watts Bar nominally has 4 OPERABLE DGs supplying two units. A fifth is maintained as a spare. All four OPERABLE DGs are required for each unit. Unit 1 requires both DG 1A-A and DG 2A-A to supply power to its train A emergency safeguards loads, and it requires DG 1B-B and 2B-B to supply train B safeguards loads.

2. WBN system design does not require the sequencer to schedule loading of the shutdown boards when offsite power is available. The sequences at WBN are only required to support operability of the DGs. Therefore, their explicit inclusion in the LCO is not required.

3. A Note has been added to reflect the fact that WBN design provides this CS-DG which may be substituted for any required DG.

4. The use of the qualifier "required" in reference to offsite circuits has been deleted. WBN design only includes two "qualified" circuits required by the LCO. Since both are required it is not necessary to uniquely identify them as such with the adjective "required".

5. Completion Time for LCO 3.8.1 Required Actions B.3.1 and B.3.2 were agreed to be 24 hours in final disposition meetings between the Industry and the NRC.

6. WBN DGs have been modified with a soakback system that eliminates the need for a prelube and warmup (see TVA January 12, 1984 letter to NRC, and SSER 3 section 9.5.7).

7. Items c,d,and e are only operative during test mode so there are no bypass arrangements required for normal operation mode (see WBN FSAR section 8.3.1.1 page 8.3-12).

8. Modification has been made to reflect the fact that DG batteries are separate from the batteries included in DC sources subsystem (i.e. LCO 3.8.4).

9. This information is given later in this section of the BASES. It is not necessary to repeat it here.

JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.8.1 (cont.)

10. FSAR table 8.3-3 provides a range rather than a single number for the sequencer delay associated with a given safeguards load.

11. Reference to a "specific list of" features has been deleted since there is no requirement, in accordance with the SFDP, to maintain such a list.

12. Modified to reflect WBN specific design. Specifically, the Condition of three or more AC sources inoperable in the standard had to be modified to reflect the fact that two DGs on the same train do not count separately in determining the number of required sources inoperable before immediate entry into LCO 3.0.3 is required.

13. The wording provided in the standard is specific to a certain plant design and not consistent with WBN design. It has been replaced by wording that is more appropriate for the WBN design.

14. The statement that <u>all</u> redundancy in electrical power supplies may not have been lost in this condition is not strictly correct for the WBN design. Conditions G and H can be entered with <u>some</u> redundancy intact. However, all redundancy in AC electrical supplies cannot be <u>guaranteed</u>.

15. Additional clarification has been provided to aid interpretation of requirements.

16. Target reliability value has been changed from .95 in the standard to .975 to "comport with the reliability levels assumed in a licensees coping analysis for Station Blackout" per SECY-92-025, resolution of Generic Issue B-56 "Diesel Generator Reliability".

17. The BASES wording discussing the minimum voltage at which DGs are connected to their respective shutdown boards has been raised from 90% rated voltage to a minimum of 95% rated voltage to more closely comply with R.G. 1.9 (i.e. Requires that DGs be connected to their electrical bus when they have reached rated voltage and frequency).

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

19. The offsite power sources are not classified as 1E.

20. The steady state voltage values for D/G operation have not been established but are expected to be much tighter than the $\pm$ 10 tolerance normally specified.

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources – Shutdown

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

a. One qualified circuit between the offsite transmission network and the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems - Shutdown;" and

b. ~~One~~ Two diesel generator (DG)s capable of supplying ~~the~~ one train of the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10. either Train A or Train B ④

APPLICABILITY: ~~In~~ MODES 5 and 6.
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required qualified offsite circuit inoperable.	------------NOTE------------ ~~With one required qualified train deenergized as a result of Condition A,~~ enter applicable Conditions and Required Actions ~~for~~ AC ~~D~~istribution subsystems. ~~LCO 3.8.10.~~ --------------------------	④ ~~deenergized required~~ of LCO 3.8.10, "Distribution Systems- Shutdown"
	A.1 Declare affected required feature(s) with no offsite power available inoperable. ~~and enter applicable Conditions and Required Actions of the LCOs for the inoperable required feature(s).~~	Immediately ④
	~~AND~~	
		(continued)

---- NOTE ----
The C-S DG may be substituted for any of the required DGs.
② _ _ _ _ _ _ _

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued) 4	~~A.1.2 Initiate action to restore required offsite power circuit to OPERABLE status.~~	~~Immediately~~
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
	AND	
4	A.2.4 Initiate action to ~~S~~suspend operations involving positive reactivity additions which would exceed limits specified in LCO 3.1.2, "Shutdown Margin," or LCO 3.9.1, "Boron Concentration."	Immediately
	AND	
	A.2.5 Initiate action to restore required offsite power circuit to OPERABLE status.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One required DG inoperable.	B.1	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	B.2	Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>		
	B.3	Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
	<u>AND</u>		
4	B.4	Initiate action to Suspend operations involving positive reactivity additions, which would exceed limits specified in LCO 3.1.2, "Shutdown Margin" or LCO 3.9.1 "Boron Concentration".	Immediately
	<u>AND</u>		
	B.5	Initiate action to restore required DG to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.2.1	For all AC sources ~~equipment~~ required to be OPERABLE the following SRs are required to be met: SR 3.8.1.1, SR 3.8.1.2, SR 3.8.1.4, SR 3.8.1.5, SR 3.8.1.6, SR 3.8.1.7, SR 3.8.1.11, ~~SR 3.8.3.1~~, ~~SR 3.8.3.4~~. 3 SR 3.8.1.17, SR 3.8.1.20 SR 3.8.1.21	In accordance with applicable SRs

SR 3.8.1.22 SR 3.8.1.23 SR 3.8.1.24
SR 3.8.1.25

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources – Shutdown

BASES

BACKGROUND

A description of the AC sources is provided in the Bases for ~~Specification~~ LCO 3.8.1, "AC Sources – Operating."

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC sources ~~and~~ during ~~Modes~~ MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that ~~(Ref. 1)~~:

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel-handling accident.

In general, when the unit is shut down the ~~TS~~ Technical Specification requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many ~~design basis accidents~~ (DBAs), which are analyzed in MODES 1, 2, 3 and 4 have no specific analyses in MODES 5 and 6. Worst-case bounding events are deemed not-credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and ~~in~~ minimal in consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

During MODES 1, 2, 3 and 4 various ~~deviations~~ allowance from the analysis assumptions and design requirements are allowed within the REQUIRED ACTIONS. This is in recognition that certain testing and maintenance activities must be conducted

(continued)

BASES (continued)

provided an acceptable level of risk is not exceeded. During MODES ~~4~~ 5 and ~~5~~ 6, performance of a significant number of required testing and maintenance activities is also required. In MODES 5 and 6 ~~these shutdown plant conditions the~~ activities are generally planned and administratively controlled. Relaxations from MODE 1, 2, 3 and 4 LCO requirements are acceptable during shutdown MODES based on:

a. ~~1)~~ The fact that time in an outage is limited. This is a risk-prudent goal as well as a utility economic consideration.

b. ~~2)~~ Requiring appropriate compensatory measures for certain conditions. These may include administrative controls and/or reliance on systems which do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses.

c. ~~3)~~ Prudent utility consideration of the risk associated with multiple activities which could affect multiple systems.

d. ~~4)~~ Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3 and 4 OPERABILITY requirements) with systems assumed to function during an event.

In the event of ~~an~~ this accident during shutdown, ~~the AC Sources Shutdown~~ LCO ~~will assure~~ ensures the capability to support systems necessary to avoid immediate difficulty, ~~of the plant~~, assuming a loss of either all offsite power or a loss of all onsite diesel generator (DG) power. ~~However,~~

~~a. Redundant and independent systems are required to be OPERABLE; or~~

~~The~~ AC sources satisfy Criterion 3 of the NRC ~~Interim~~ Policy Statement. ~~(Ref. 3 and 4)~~.

LCO

One offsite circuit capable of supplying the onsite Class 1E power distribution subsystem(s) of LCO 3.8.10 "Distribution Systems – Shutdown," ensures that all required loads are powered from offsite power. ~~An OPERABLE DG~~ Two OPERABLE DGs, associated with a ~~D~~istribution ~~S~~ystem train required to be OPERABLE by LCO 3.8.10, ensures a diverse power

1

(continued)

BASES (continued)

source is available to provide electrical power support ~~in the event of a single failure consisting of~~ assuming a loss of the offsite circuit. Together, OPERABILITY of the required offsite circuit and ~~DG~~ the two DGs ensure the availability of sufficient AC sources to operate the ~~plant~~ unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel-handling accidents~~,~~ and reactor vessel draindown).

A

Offsite power from the Watts Bar Hydro 161-kV switchyard to the onsite Class 1E distribution system is from two independent immediate access circuits. Each of the two circuits are routed from the switchyard through a 161-kV transmission line and 161- to 6.9-kV transformer (common station service transformers) to the onsite Class 1E distribution system. The low and medium voltage power system starts at the high-side of the common station service transformers.

~~Each~~ The qualified offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the ~~ESF buses~~. Qualified offsite circuits are those ~~which~~ that are described in the FSAR, have been reviewed and accepted by the staff as meeting the requirements of GDC 17, and are part of the licensing basis for the unit. ~~[Offsite circuit #1 consists of Safeguards Transformer 'B', which is supplied from Switchyard Bus 'B', and is fed through breaker 52-3 powering the ESF transformer XNB01, which, in turn, powers the #1 ESF bus through its normal feeder breaker. The second offsite circuit consists of the Startup Transformer, which is normally fed from the Switchyard Bus 'A', and is fed through breaker PA0201 powering the ESF transformer, which, in turn, powers in #2 ESF bus through its normal feed Breaker.]~~

1 B

~~Each~~ The DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ~~ESF bus~~ 6.9 kV shutdown board on detection of bus ~~undervoltage~~ loss-of-voltage. This ~~will~~ sequence must be accomplished within [~~12~~ 10] seconds. Each DG bus must be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the ~~ESF busses~~ 6.9 kV shutdown boards. These capabilities are required to be met from a variety of initial conditions such as: DG in standby ~~hot~~ with the engine hot, DG in standby ~~cold~~ at ambient conditions, and DG operating in parallel test mode.

Proper sequencing of loads, including tripping of non-essential loads, is a required function for DG OPERABILITY. ~~[In addition proper sequencer operation is an integral part of offsite circuit OPERABILITY if its INOPERABILITY in any way impacts on the ability to start and maintain energized any loads required OPERABLE by LCO 3.8.10.]~~

(continued)

(2) A Note has been added to indicate that the C-S DG may be substituted for any of the required DGs. However, the CS DG cannot be declared OPERABLE until it is connected electrically in place of another DG, and it has satisfied applicable Surveillance Requirements.

BASES (continued)

It is acceptable for trains to be cross tied during shutdown conditions allowing a single offsite power circuit to supply all required trains.

APPLICABILITY

△ The AC sources required to be OPERABLE in MODES 5 and 6 and ~~also any time~~ during movement of irradiated fuel assemblies provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core in case of an inadvertent draindown of the reactor vessel;

b. Systems needed to mitigate a fuel-handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

△ AC power requirements for MODES 1, 2, 3 and 4 are covered in ~~Specification~~ LCO 3.8.1. ~~"AC Sources – Operating."~~

ACTIONS

△ A.1 ~~2~~

(5) An offsite circuit would be considered inoperable if it were not available to one required ESF train. ~~Although, two~~ In cases where two trains are required per LCO 3.8.10, ~~Distribution System - Shutdown,~~ the one train with offsite power available may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, and ~~/or~~ operations with a potential for draining the reactor vessel. ~~By allowing~~ the allowance of the option to declare required features inoperable with no offsite power available, appropriate restrictions will be implemented in accordance with the affected required features LCOs' ACTIONS.

(continued)

BASES (continued) △ A.2.5

option may △

A.2.1, A.2.2, A.2.3, A.2.4, B.1, B.2, ~~B.3~~ ~~and~~ B.4 and B.5

With the offsite circuit not available to all required trains, the option would still exist to declare all required features inoperable. Since this ~~would likely~~ involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With ~~the~~ either required DG inoperable, the minimum required diversity of AC power sources is not. available It is, therefore, required to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, any activities that could potentially result in inadvertent draining of the reactor vessel, and operations involving positive reactivity additions. The Required Action to suspend positive reactivity additions does not preclude actions to maintain ~~or~~ increase reactor vessel inventory provided the required or shutdown margin is maintained. (1) △ (4) INSERT

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions ~~will~~ minimize the probability or the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC sources and to continue this action until restoration is accomplished in order to provide the necessary AC power to the unit~~'s~~ safety systems.

△ A.1.2, A.2.5 and B.5 △

Notwithstanding ~~these~~ performance of the above conservative Required Actions, the unit is still without sufficient AC power sources to operate in a safe manner. Therefore, action must be initiated to restore the minimum required AC power sources and continue until the LCO requirements are restored.

The Completion Time of "immediately" is consistent with the required times for actions requiring prompt attention. The restoration of the required AC electrical power sources should be completed as quickly as possible in order to minimize the time during which the unit~~'s~~ safety systems may be without sufficient power. △

Pursuant to LCO 3.0.6, the distribution system ACTIONS ~~would~~ are not ~~be~~ are entered even if all AC sources to it ~~were~~ are inoperable resulting in deenergization. Therefore, the Required Actions of Condition A ~~have been~~ are modified by a Note to △

(continued)

INSERT

A.2.4 and B.4 I-1

Required Actions ⅄ precludeX operations involving systems that contain large volumes of water (i.e. CVCS, SIS, and RWST) at reduced boron concentrations with respect to the RCS that could dilute the boron concentration of the RCS to less than that required to maintain the SDM requirements of LCO 3.1.2 in MODE 5 or less than the boron concentration specified in the COLR in MODE 6. This Required Action does not preclude positive reactivity additions that cannot reduce the SDM or boron concentration to less than the limits specified in LCO 3.1.2 or the COLR. The addition of water with a boron concentration greater than that required to maintain the reactor shutdown within the requirements of LCO 3.1.2 or the COLR, but less than the RCS, is permitted. Positive reactivity additions such as small volume chemical additions and normal plant cooldowns are also permitted as long as the SDM limits are met.

BASES (continued)

indicate that when Condition A is entered with no AC power to one ~~ESF bus~~ 6.9KV Shutdown Board, ACTIONS for LCO 3.8.10 must be immediately entered. Note This allows Condition A to provide requirements for the loss of the offsite circuit ~~without regard to~~ whether ~~a division~~ or not a train is deenergized. LCO 3.8.10 ~~would~~ provides the appropriate restrictions for the situation involving a deenergized ~~division~~ train.

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 lists SRs from LCO 3.8.1 that are required to be performed. Refer to the corresponding Bases for ~~Specification~~ LCO 3.8.1 ~~and 3.8.3~~ for a discussion of each SR.

REFERENCES

1. Watts Bar FSAR, Section ~~15~~ 8.0, ~~"Accident Analysis."~~ "Electric Power."

2. ~~Title 10, Code of Federal Regulations, Part 50, Appendix A,~~ 10 CFR 50, Appendix A, GDC 17 "General Design Criteria for Nuclear Power Units."

3. ~~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."~~

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

(continued)

JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.8.2

Generic Justifications

A. Modified to reflect WBN plant specific design

B. Modified to reflect WBN plant specific terminology

Specific Justifications

1. Modified to reflect WBN plant specific design of onsite power system. Watts Bar nominally has 4 OPERABLE DGs supplying two units. A fifth is maintained as a spare. All four OPERABLE DGs are required for each unit. Unit 1 requires both DG 1A-A and DG 2A-A to supply power to its train A emergency safeguards loads, and it requires DG 1B-B and 2B-B to supply train B safeguards loads.

2. A Note has been added to reflect the fact that WBN design provides this CS-DG which may be substituted for any required DG.

3. Modification has been made to reflect the fact that DG batteries are separate from the batteries included in DC sources subsystem (i.e. LCO 3.8.4).

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

5. Editorial change for clarity. Two trains are not specifically required by LCO 3.8.10, the only requirement in LCO 3.8.10 is for sufficient distribution equipment to support required operable components. These may be instances where two trains are required and that is the purpose of Required Action A.1.

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel ~~,~~ Oil, Lub~~ricating~~e Oil, and Starting Air Δ

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

APPLICABILITY: When associated DG is required to be OPERABLE.

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
(A) A. One or more DGs with fuel level < 60,000 ~~[33,000]~~ gal~~lons~~ and ≥ 52,488 ~~[28,285]~~ gal~~lons~~ in storage tank. 7day	A.1	Restore fuel oil ~~level in DG storage tank~~. Δ to within limits	48 hours
B. One or more DGs with Δ lub~~ricating~~e oil inventory < 300 ~~[500]~~ gal~~lons~~ and ≥ 280 ~~[425]~~ gal~~lons~~. (A)	B.1	Restore lube oil inventory.	48 hours
C. One or more DGs with stored fuel oil total particulates ≥ 10 mg/l.	C.1	Restore fuel oil total particulates < 10 mg/l.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more DGs with new fuel oil properties not within limits of the Diesel Fuel Oil Testing Program.	D.1 Restore stored fuel oil properties to within limits.	30 days
E. One or more DGs with starting air receiver pressure < [~~225~~ 200] ~~PSI~~ psig but ≥ [~~125~~] ~~PSIG~~ psig (A)	E.1 Restore starting air receiver pressure to ≥ [~~225~~ 200] ~~PSI~~ psig Δ (A)	48 hours
F. Required Action and associated Completion Time not met. OR Δ One or more DGs with Diesel fuel oil, lube oil, or starting air subsystem ~~inoperable~~ not within limits for reasons other than Condition A, B, C ~~or~~ D, or E.	F.1 Declare associated ~~diesel generator~~ DG inoperable. ~~and enter the Applicable Conditions and Required Actions for inoperable DG, AC Sources LCO 3.8.1, "AC Sources - Operating," or LCO 3.8.2, "AC sources - Shutdown."~~ Δ (4)	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	(A) (B) Verify each 7 day fuel oil Δ storage tank contains ≥ [~~33,000~~ 60,000] gal~~lons~~ of fuel.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.3.2	Verify lub~~ricating~~e oil inventory is ≥ ~~[500]~~ 300 gal~~lons~~ per engine. (A) (1)	31 days
SR 3.8.3.3	(6) Verify fuel oil properties of ~~new and stored fuel oil~~ fuel oil stored in and prior to transfer to the 7-day storage tanks are tested in accordance with, and maintained within the limits of the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program
SR 3.8.3.4	Verify ~~required~~ each DG air start receiver pressure ≥ ~~[225]~~ 200 ~~PSI~~ psig. (A)	31 days
SR 3.8.3.5	(A) Check for and remove accumulated water from each 7 day fuel oil storage tank~~, and day tank, [and engine mounted tank]~~.	[31] days
SR 3.8.3.~~6~~7	(B) For ~~the~~ each 7 day fuel oil ~~system~~ storage tank: (4) a. Drain ~~each~~ the fuel oil~~, storage tank~~. b. Remove the sediment~~, from the storage tank~~ and c. Clean the ~~storage~~ tank.	10 years
SR 3.8.3.6	(2) Perform a visual inspection for leaks in the exposed fuel oil system piping while the DG is running.	18 months

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Diesel Fuel oil, Lube~~ricating~~ Oil, and Starting Air

four interconnected storage tanks embedded in the building foundation

BASES

(A)

BACKGROUND

Each diesel generator (DG) is provided with ~~a storage tank~~ four interconnected storage tanks embedded in the building foundation having a fuel oil capacity sufficient to operate that diesel for a period of 7 ~~seven~~ days while the DG ~~diesel generator~~ is supplying maximum post loss-of-coolant accident (LOCA) load demand discussed in the FSAR Section 8.0 (Ref. 1). The maximum load demand is calculated using the assumption that a minimum of any two DGs ~~diesel generators~~ are available. This onsite fuel oil capacity is sufficient to operate the DGs ~~diesel generators~~ for longer than the time ~~it would take~~ to replenish the onsite supply from outside sources. (B)

(A) An approximately 550 gal skid-mounted day tank is provided for each diesel engine. Each DG incorporates two diesel engines operating in tandem and directly coupled to the generator. Each skid-mounted day tank has fuel capacity for approximately 2 hours of full-load operation (Ref. 1).

Fuel oil is transferred from the 7-day storage tanks to the skid-mounted day tanks by each ~~either of two transfer~~ pumps located on that skid-mounted day ~~associated with each storage tank~~. Redundancy of pumps and piping precludes the failure of one pump, or the rupture of any pipe, valve, or tank to result in the loss of more than one ~~DG. All outside tanks, pumps and piping are located underground.~~ (1) diesel engine.

In the event that the piping between the last isolation valve and the skid-mounted day tank breaks, the use of one DG can be lost. This occurs only after the two hour supply of fuel in the skid-mounted day tank has been used.

For proper operation of the standby DGs ~~diesel generators~~, it is necessary to ensure the proper quality of the fuel oil. Regulatory Guide 1.137 (Ref. 2) addresses the recommended fuel oil practices as supplemented by ANSI N195-1976 (Ref. 3). The fuel oil properties governed by these SRs ~~Surveillance Requirements~~ are the appearance ~~water and sediment content~~, the kinematic viscosity, specific gravity (or API gravity), ~~and~~ impurity level, and flash point. (B)

~~The diesel generator lubrication system is designed to provide sufficient lubrication to permit proper operation of its associated diesel generator under all loading conditions. The system is required to circulate the lube oil to the diesel engine working surfaces and to remove excess heat generated by friction during operation. Each engine oil sump contains an inventory capable of supporting a minimum of [7] days operation. [The onsite storage in addition to the engine oil sump is sufficient to ensure seven days continuous operation.] This provides sufficient supply to allow the operator to replenish lube oil from outside sources.~~

During operation of the DGs, fuel oil pumps driven by the diesel engines transfers fuel from the skid-mounted day tanks to the diesel engine fuel manifolds. Level controls mounted on the skid-mounted day tanks automatically start and stop the storage tank transfer pumps. In addition, alarms both locally and in the control room annunciate low level and high level in any skid-mounted day tank. In the unlikely event of a failure in one of the supply trains, the associated skid-mounted day tank low-level alarm annunciates when the fuel oil remaining in the tank provides approximately 1 hour of full-load operation, thus allowing the operator to take corrective action to prevent the loss of the diesel.

(continued)

BASES (continued)

See next page for Insert (A)

Each ~~diesel generator~~ DG has an air start system with adequate capacity for ~~5~~ five successive start attempts on the ~~diesel generator~~ DG without recharging the air start receiver(s).

APPLICABLE SAFETY ANALYSES

The initial conditions of ~~design basis~~ Design Basis Accident (DBA) and transient ~~and accident~~ analyses in FSAR Chapter 6, ~~Engineered Safety Features,~~ and Chapter 15, ~~Accident Analyses,~~ (Ref. 5) assume Engineered Safety Features (ESF) systems are OPERABLE. The ~~diesel generators~~ DGs are designed to provide sufficient capacity, capability, redundancy and reliability to ensure the availability of necessary power to ESF systems so that fuel, Reactor Coolant System (RCS) and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for ~~LCO Sections~~ Specifications 3.2 "Power Distribution Limits," 3.4 "Reactor Coolant System (RCS)," and 3.6 "Containment Systems."

Since Diesel Fuel, Lube ~~ricating~~ Oil, and Air Start Subsystems support the operation of the standby AC power sources, they satisfy Criterion 3 of the NRC ~~Interim~~ Policy Statement ~~as it was applied to AC Sources (Ref. 4 and 5).~~

LCO

Stored diesel fuel oil is required to have sufficient supply for 7 days of full load operation. It is also required to meet specific standards for quality. Additionally, sufficient lube ~~ricating~~ oil supply must be available to ensure the capability to operate at full load for 7 days. This requirement, in conjunction with an ability to obtain replacement supplies within this 7 days, supports the availability of DGs required to shutdown the reactor and maintain it in a safe condition for an anticipated operational occurrence (AOO) or a postulated ~~design basis accident~~ DBA with loss of offsite power. DG skid-mounted day tank fuel oil requirements, as well as transfer capability from the 7-day storage tank to the skid-mounted day tank, are address in LCO 3.8.1, "AC Sources-Operating," and ~~3.8.2.~~ LCO 3.8.2, "AC Sources-Shutdown."

The starting air system is required to have a minimum capacity for ~~5~~ five successive ~~diesel generator~~ DG start attempts without recharging the air start receivers.

(continued)

Each of the engin~~es~~e in the ~~tandem/generator~~ DG sets is provided with its own lu~~bricating~~be oil ~~system, which~~ is an integral part of each of the five ~~diesel/generator~~ units. The piping and components for the skid-mounted lubrication system are vendor supplied, safety-related, ANSI B31.1, Seismic Category I. The diesel engine lubrication system for each diesel engine is a combination of four subsystems (Ref. 4): the main lubricating subsystem, the piston cooling subsystem, and the scavenging oil subsystem and the motor-driven circulating pump, and soak back pump system. The main lubricating subsystem supplies oil under pressure to the various moving parts of the diesel engine. The piston cooling subsystem supplies oil for piston cooling and lubrication of the piston pin bearing surfaces. The scavenging oil subsystem supplies the other systems with cooled and filtered oil. Oil is drawn from the engine sump by the scavenging pump through a strainer in the strainer housing located on the front side of the engine. From the strainer the oil is pumped through oil filters and a cooler. The filters are located on the accessory racks of the engines. The oil is cooled in the lub~~ricating~~e oil cooler by the closed circuit cooling water system in order to maintain proper oil temperature during engine operation.

Each engine crankcase sump contains 330 gal~~lons~~ of lub~~ricating~~e oil, ample for at least ~~seven~~ 7 days of ~~diesel generator~~ DG unit full load operation without requiring replenishment. The established oil consumption rate is 0.83 gal~~lons~~ per hour. An additional standby oil reserve of approximately 935 gal~~lons~~ is stored onsite to replenish the engines for longer periods of operation and after their periodic test operations.

BASES (continued)

APPLICABILITY △ AC Sources (LCO 3.8.1 and 3.8.2) are required to ensure the availability of the required power to shutdown the reactor and maintain it in a safe shutdown condition after an AOO ~~anticipated operational occurrence~~ or a postulated ~~Design Basis Accident~~ ~~(~~DBA~~)~~. Since stored diesel fuel oil, lub~~ricating~~e oil, and starting air subsystem~~s~~ support ~~AC Sources—Operating~~ ~~(~~LCO 3.8.1~~)~~ and ~~AC Sources—Shutdown~~ ~~(~~LCO 3.8.2~~)~~, stored diesel fuel oil, lub~~ricating~~e oil and starting air Subsystem are required to be within limits when the associated ~~diesel generator~~ DG is required to be OPERABLE.

ACTIONS

A.1

△ In this Condition, the 7-day fuel oil supply for a DG is not available. However, the Condition is restricted to fuel oil level reductions ~~which still~~ that maintain at least a 6-day supply. These circumstances may be caused by events such as; ~~1)~~ a. full load operation required after an inadvertent b. start while at minimum required level; or ~~2)~~ feed-and-bleed operations which may be necessitated by increasing particulate levels or any number of other oil quality degradations. This restriction allows ~~will allow~~ sufficient time for obtaining ~~to obtain~~ the requisite replacement volume and performing the analyses required prior to addition of the fuel oil to the tank. A period of 48 hours is considered sufficient to complete restoration of the required level prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (>6 days), the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

B.1

A

△ With lub~~ricating~~e oil inventory ~~less than 500~~ < 300 gal~~lons~~ per diesel engine, sufficient lub~~ricating~~e oil to support seven days of continuous ~~diesel generator~~ DG operation at full load conditions may not be available. However, the Condition is restricted to lub~~ricating~~e oil volume reductions ~~which still~~ that maintain at least a 6-day supply. This ~~will~~ restriction allows sufficient time to obtain the requisite replacement volumes. A period of 48 hours is considered sufficient to complete restoration of

(continued)

BASES (continued)

the required volume prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (>6 days), the low rate of usage, the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

C.1

This Condition is entered as a result of a failure to meet the acceptance criterion of SR 3.8.3.5. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and ~~for~~ errors in laboratory analysis can produce failures ~~which~~ do not follow a trend. that Since the presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, and particulate concentration is unlikely to change significantly between surveillance frequency intervals, and proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG inoperable. The 7-day Completion Time ~~will~~ allows for further evaluation, resampling and reanalysis of the DG fuel oil.

D.1

for ing

With the new fuel oil properties defined in the Bases for SR 3.8.3.4 not within the required limits, a period of 30 days is allowed ~~to~~ restore the stored fuel oil properties. This provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable and ~~to~~ restore the stored fuel oil properties. This restoration may involve feed-and-bleed procedures, filtering, or combinations of these ~~other~~ procedures. Even DG if a ~~diesel generator~~ start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the ~~diesel generator~~ would still be capable of performing its intended function.
DG

(continued)

BASES (continued)

E.1

five psig, 200 A psig

△ With staring air receiver pressure less than [~~225~~] ~~PSI,~~ sufficient capacity for ~~5~~ successive ~~diesel generator~~ start attempts does not exist. However, as long as the receiver pressure is greater than [~~200~~] ~~PSI,~~ there is adequate capacity for at least one start attempt, and the ~~diesel generator~~ can be considered OPERABLE while the air receiver pressure is restored to the required limit. A period of 48 hours is considered sufficient to complete restoration to the required pressure prior to declaring the ~~diesel generator~~ inoperable. This period is acceptable based on the remaining air start capacity, the fact that most ~~diesel generator~~ starts are accomplished on the first attempt, and the low probability of an event during this brief period.

F.1 one or more DGs 4

E, or starting air subsystem

△ With a Required Action and associated Completion Time not met, or ~~the stored diesel~~ fuel oil, ~~or~~ lub~~ricating~~ oil not within limits for reasons other than addressed by Conditions A through ~~D~~ E, the associated ~~diesel generator~~ may be incapable of performing its intended function and must be immediately declared inoperable. This declaration also requires entry into Applicable Conditions and Required Actions for LCO 3.8.1 or LCO 3.8.2.

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

unit

△ This SR provides verification that there is an adequate inventory of fuel oil in the storage tanks to support each DG~~'s~~ operation for 7 days at full load. The 7-day period is sufficient time to place the unit ~~facility~~ in a safe shutdown condition and to bring in replenishment fuel from an offsite location.

The 31-day Frequency is adequate to ensure that a sufficient supply of fuel oil is available, since low-level alarms are provided and facility operators would be aware of any large uses of fuel oil during this period.

(continued)

BASES (continued)

The DG lube oil sump is designed to hold adequate inventory for 7 days of full-load operation without the level reaching the manufacturers recommended minimum level.

3

plus the manufacturer's recommended minimum oil level.

SR 3.8.3.2

A 300

This Surveillance ensures that sufficient lub~~ricating~~e oil inventory is available to support at least 7 days of full-load operation for each DG. The ~~[500]~~ gal requirement is based on the DG manufacturer's consumption values for the run time of the diesel. ~~Implicit in this SR is the requirement to verify the capability to transfer the lubricating oil from its storage location to the DG, when the DG lubricating oil sump does not hold adequate inventory for 7 days of full-load operation without the level reaching the manufacturers recommended minimum level.~~

A 31-day Frequency is adequate to ensure that a sufficient lub~~ricating oil~~e supply is onsite, since DG starts and run time are closely monitored by the unit staff.

SR 3.8.3.3

6 that is to be transferred to the 7 day storage tank

The tests listed below are a means of determining whether new fuel oil is of the appropriate grade and has not been contaminated with substances that would have an immediate, detrimental impact on diesel engine combustion~~/operation~~. If results from these tests are within acceptable limits, the fuel oil may be added to the storage tanks without concern for contaminating the entire volume of fuel oil in the storage tanks. The tests, limits, and applicable American Society for Testing Materials (ASTM) standards are as follows:

Insert moved from next page

a. Sample the new fuel oil in accordance with ASTM D405~~X~~M-[1990]; (Ref. 6)

b. Verify in accordance with the tests specified in ASTM D975-[1990] that the sample has an absolute specific gravity at 60/60°F of ≥ 0.83 ~~but~~ and ≤ 0.89 or an API gravity at 60°F of ≥ 27° ~~but~~ ≤ 39°, a kinematic viscosity at 40°C of ≥ 1.9 centistokes ~~but~~ ≤ 4.1 centistokes, and a flash point ≥ 125°F; and

5

c. Verify that the new fuel oil has a clear and bright appearance with proper color when tested in accordance with ASTM D4176 [1986] Method A or B (Ref. 6).

(continued)

BASES (continued)

6 transfer of new fuel to the 7-day storage tanks

Move to pg B 3.8-6

△ These tests are to be conducted prior to adding the new fuel to the storage tank(s), but in no case shall the time between ~~receipt of new fuel~~ and conducting the tests exceed 31 days.

B Failure to meet any of the above limits is cause for rejecting the new fuel oil, but does not represent a failure to meet the LCO concern since the fuel oil is not added to the storage tanks.

△ Within 31 days following the addition of ~~initial~~ new fuel-oil sample [1987] (Ref. 6), ~~the fuel oil~~ the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D975 [1990] (Ref. 6) are met for new fuel oil when tested in accordance with ASTM D975, except that the analysis for sulfur may be performed in accordance with ~~ASTM D1522 or~~ ASTM D2622 1552 [1990]. The 31-day period is acceptable because the fuel-oil properties of interest, even if they were not within stated limits, would not have an immediate effect on DG operation. This Surveillance ensures the availability of high-quality fuel oil for the DGs.

Fuel-oil degradation during long-term storage shows up as an increase in particulate, due mostly to oxidation. The presence of particulate does not mean the fuel oil will not burn properly in a diesel engine. The particulate can cause fouling of filters and fuel-oil injection equipment, however, which can cause engine failure.

A Particulate concentrations should be determined in accordance with ASTM D2276 [1989] (Ref. 6), Method A. This method involves a gravimetric determination of total particulate concentration in the fuel oil and has a limit of 10 mg/l. It is acceptable to obtain a field sample for subsequent lab testing in lieu of field testing. ~~[In this case(s) where the total stored fuel oil volume is contained in two or more interconnected tanks,~~ Of the four interconnected tanks which comprise a 7day each tank must be considered and tested separately. The Frequency of this test takes into consideration fuel-oil degradation trends that indicate that particulate concentration is unlikely to change significantly between Frequency intervals.

SR 3.8.3.4 △

This Surveillance ~~verifies~~ ensures that, without the aid of the refill compressor, sufficient air-start capacity for each DG

(continued)

BASES (continued)

A start cycle is defined by the DG vendor, but usually is measured in terms of time (seconds of cranking) or engine cranking speed.

is available. The system design requirements provide for a minimum of five engine start cycles without recharging. The pressure specified in this SR is intended to reflect the lowest value at which the five starts can be accomplished. ~~If the pressure is less than the value specified in SR Condition E is entered~~.

The 31-day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal air-start pressure.

SR 3.8.3.5

Microbiological fouling bacterial is a major cause of fuel-oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the ~~fuel-oil day [and engine-mounted] tanks and from~~ storage tanks once every 31 days ~~will~~ eliminates the necessary environment for survival. This is the most effective means of controlling microbiological fouling. In addition, it ~~will~~ eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water ~~will~~ minimizes fouling ~~as well as~~ and provides data regarding the watertight integrity of the fuel-oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 7). This SR is for preventative maintenance. The presence of water does not necessarily represent failure of this SR provided the accumulated water is removed during the performance of the surveillance.

SR 3.8.3.6

This SR verifies by visual inspection, that the exposed fuel oil system piping is free of leaks. The test is performed while the DG is running to provide adequate assurance of piping leaktightness and weld integrity. The 18 month Frequency is based on engineering judgement and is consistent with the refueling cycle testing performed on the DGs.

(2) SR 3.8.3.~~6~~7

Draining of the fuel oil stored in the supply tanks, removal of accumulated sediment, and tank cleaning are required at 10-year intervals by Regulatory Guide 1.137 (Ref. 9), paragraph 2.f. This SR also requires the performance of the ASME Section XI examinations of the tanks. To preclude the introduction of (Ref. 7) surfactants in the fuel system, the cleaning

(7)

(continued)

BASES (continued)

should be accomplished using sodium hypochlorite solutions, or their equivalent, rather than soap or detergents. This SR is ~~a~~ for preventative maintenance ~~SR~~. The presence of sediment does not necessarily represent a failure of this SR provided that accumulated sediment is removed during performance of ~~this SR~~ the Surveillance.

REFERENCES

1. ~~[ ]~~ Watts Bar FSAR, Section ~~[9.5.4.2]~~ 8.3, "Onsite (Standby) Power System."
2. Regulatory Guide 1.137, "Fuel Oil Systems for Standby Diesel Generators."
3. ANSI N195-1976, "Fuel Oil Systems for Standby Diesel Generators," Appendix B.
4. ~~NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment", November 1987.~~
5. ~~Thomas E. Murley (NRC) letter to Robert F. Janecek (BWROG) 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 Standard Technical Specifications."~~

(A) 4. Watts Bar FSAR, Section 9.5.7, "Diesel Engine Lubrication System."

(A) 5. Watts Bar FSAR, Section 15, "Accident Analyses," and Section 6, "Engineered Safety Features."

6. ASTM Standards:
D4057 - 1990
D975 - 1990
D4176 - 1986
D1552 - 1990
D2622 - 1987
D2276 - 1989

7. 7. ASME, Boiler and Pressure Vessel Code, Section XI

(continued)

JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.8.3

Generic Justifications

A. Modified to reflect WBN plant specific design.

B. Modified to reflect WBN plant specific terminology.

Specific Justifications

1. Modified to reflect the fact WBN design set driving a common generator (see FSAR page 8.3-16 and Section 9.5.7).

2. Modified to include a Surveillance satisfying TVA commitment to resolve SER 9.5.4.1 concerns related to 10 year hydro test. This was accepted by SSER 5, 9.5.4.1.

3. Since WBN lube oil sump is designed to hold adequate inventory for 7 days operation, there is no implicit requirement to verify the capability to transfer lube oil from its storage location to the DG.

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

5. Change to correct error in the STS. API gravity is given in degrees, but specific gravity is not.

6. From a Tech Spec standpoint, only fuel that is transferred to the 7-day storage tank for each DG is considered "new fuel" consistent with the proposed fuel oil program, Specification 5.7.2.16, fuel received onsite and stored for other purposes should not be required to meet Tech Spec requirements.

7. The Watts Bar 7-day storage tanks were purchased as ASME, Section VIII as noted in FSAR 9.5.4.1 Section XI is not applicable to these tanks.

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources — Operating

LCO 3.8.4 The Train A and Train B DC ~~electrical~~ power subsystems shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required DC electrical power subsystem inoperable.	A.1	Restore DC electrical power subsystem to OPERABLE status.	2 hours
B. Required Action and Associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is ≥ ~~[129]~~ V on float charge. A ≥128 V (132 for Vital Battery V)	7 days

(continued)

1 NOTE
Vital Battery V may be substituted for any of the required vital batteries.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.~~2~~ 3	Verify no visible corrosion at terminals and connectors. OR 80E-6 A 100E-6 80E-6 Verify connection resistance ~~[of these items~~ is ≤ [~~10 x 10^-6~~ ohms] for inter-cell connections, ≤ [~~10 x 10^-6~~ ohms] for inter-rack connections, ≤ [~~10 x 10^-6~~ ohms] for inter-tier connections, and ≤ [~~10 x 10^-6~~ ohms] for terminal connections~~]~~.	92 days
SR 3.8.4.~~3~~ 4	Verify cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration.	12 [~~12~~] months
SR 3.8.4.~~4~~ 5	Cell-to-cell and terminal Remove visible terminal corrosion, verify ~~that~~ connections are clean ~~and tight~~, and coat~~, terminals~~ with anti-corrosion material~~.~~ ed 12	12 [~~12~~] months
SR 3.8.4.~~5~~ 6	80 E-6 Verify connection resistance ~~[of these items~~ is ≤ [~~10 x 10^-6~~ ohms] for inter-cell connections, ≤ [~~10 x 10^-6~~ ohms] for inter-rack connections, ≤ [~~10 x 10^-6~~ ohms] for inter-tier connections, and ≤ [~~10 x 10^-6~~ ohms] for terminal connections~~]~~. 80 E-6 100 E-6 A	12 [~~12~~] months

2 SR 3.8.4.2 Verify the alternate feeder breakers to each required battery charger are open. 7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.4.~~6~~ 7	------------------NOTE~~S~~------------------ 1. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. 2. Credit may be taken for unplanned events that satisfy this SR. --- Verify each battery charger ~~will supply~~ supplies ≥ ~~[400]~~ 150 amps at ≥ ~~[~~125~~/250]~~ V for ≥ ~~[8]~~ 4 hours. 4 10 4	~~[~~18 months~~]~~
SR 3.8.4.~~7~~ 8	------------------NOTES------------------ 1. SR 3.8.4.~~8~~ 9 may be performed in lieu of SR 3.8.4.~~7~~ 8 once per 60 months. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. 3. Credit may be taken for unplanned events that satisfy this SR. --- Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery-service test.	~~[~~18 months~~]~~

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.~~8~~ 9	------------------NOTES------------------ 1. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. 2. Credit may be taken for unplanned events that satisfy this SR. -- Verify battery capacity is ≥ [80%] of the manufacturer's rating when subjected to a performance discharge test.	60 months AND -----NOTE------ Only applicable when battery shows degradation or has reached [85%] of the expected life --------------- 12 months 3 OR ----NOTE---- Only applicable when battery does not show degradation and has reached 85% of the expected life and is ≥ 100% capacity -------------- 24 months

△

A

system whose safety function is to provide control power for engineered safety features equipment, emergency lighting, vital inverters, and other safety-related DC powered equipment for the entire unit. The system capacity is sufficient to supply these loads during normal operation and to permit safe shutdown and isolation of the reactor for the "loss of all AC power" condition. The system is designed to perform its safety function subject to a single failure.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 DC Sources — Operating

BASES

A 5

BACKGROUND

The 125 VDC vital power system is composed of the four redundant channels (channels I and III compose Train A and channels II and IV compose Train B) and consists of four lead-acid-calcium batteries, six battery chargers (including two spare chargers), four distribution boards, battery racks, and the required cabling, instrumentation and protective features. Each channel is electrically and physically independent from the equipment of all other channels so that a single failure in one channel will not cause a failure in another channel. Each channel consists of a battery charger which supplies normal DC power, a battery for emergency DC power, and a battery board which facilitates load grouping and provides circuit protection. These four channels are used to provide emergency power to the 120 V AC vital power system which furnishes control power to the reactor protection system. No automatic connections are used between the four redundant channels.

The station DC electrical power system provides the AC emergency power system with control power. It also provides both motive and control power to selected safety-related equipment and ~~preferred~~ AC Vital Bus power (via inverters). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. The DC electrical power system also conforms to the recommendations of Regulatory Guide 1.6 (Ref. 2) and IEEE-308 (Ref. 3).

B

The Vital ~~[125/250]~~ # Vdc electrical power system is a ~~consists of two independent and redundant safety-related~~ Class 1E ~~DC electrical power subsystems ([Train A and B])~~. ~~Each subsystem consists of [two] 125 Vdc batteries [(each battery [50%] capacity)], associated battery charger(s) for each battery, and all the associated control equipment and interconnecting cabling. [The 250 Vdc source is obtained by use of the two 125 Vdc batteries connected in series.] [Additionally there is [one] spare battery charger per subsystem, which provides backup service in the event that the preferred battery charger is out of service. If the spare battery charger is substituted for one of the preferred battery chargers, then the requirements of independence and redundancy between subsystems are maintained.]~~

~~During normal operation, the [125/250] Vdc load is powered from the battery chargers with the batteries floating on the system. In case of loss of normal power to the battery charger, the DC load is automatically powered from the station batteries.~~

Insert on next page 1 A

Each of the Train A and B ~~The [Train A and B]~~ DC Electrical Power Subsystems provides the control power for its associated Class 1E AC-power-load group, 6.9 ~~[4.16]~~ kV switchgear, and [480] V load centers. The DC ~~electrical power~~ subsystems ~~also~~ provide DC electrical power to the inverters, which in turn power the AC vital buses. Additionally, they power the emergency DC lighting system.

A

(continued)

Battery boards I, II, III, and IV have a charger normally connected to them and also have manual access to a spare (backup) charger for use upon loss of the normal charger. Additionally, battery boards I, II, III, and IV have manual access to the fifth vital battery system. The fifth 125V DC Vital Battery System is intended to serve as a temporary replacement for any one of the four 125V DC vital batteries during their testing, maintenance, and outages with no loss of system reliability under any mode of operation.

BASES (continued)

The DC-power distribution system is described in more detail in Bases for ~~Specifications~~ LCO 3.8.9, "Distribution System—Operating," and LCO 3.8.10, "Distribution System-Shutdown."

Each battery has adequate storage capacity to carry the required load continuously for at least 2 hours. and to perform three complete cycles ~~of intermittent loads~~ (Ref. 4). (13)

Each 125 Vdc battery is separately housed in a ventilated room apart from its charger and distribution centers. Each subsystem is located in an area separated physically and electrically from the other subsystem to ensure that a single failure in one subsystem does not cause a failure in a redundant subsystem. There is no sharing between redundant Class 1E subsystems such as batteries, battery chargers, or distribution panels.

The batteries for Train A and B DC Electrical Power Subsystems are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end-of-life cycles (A) derating for minimum ambient temperature and the 100% design demand. Battery size is based on ~~125%~~ of required capacity and, after selection of an available commercial battery, results in a battery capacity in excess of 150% of required capacity. (13) The voltage limit is 2.13 V per cell, which corresponds to a total minimum voltage output of ~~128 V~~ 129 V per battery (133.6 V for vital Battery V) (Ref. ~~4~~ 8). (A) The battery design complies with ~~The~~ the criteria for sizing large lead storage batteries ~~are~~ as defined in IEEE-485 (Ref. 5). (A)

Each Train A and B DC electrical power subsystem has ample power-output capacity for the steady-state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger ~~also~~ has sufficient capacity to restore the battery bank from the design minimum charge to its fully charged state ~~within 24 hours~~ within 12 hours (with accident loads being supplied) following a 30 minute AC power outage and in approximately 34 hours. (A) (6) (while supplying normal steady-state loads following a 2 hour AC power outage) (Ref. ~~4~~ 8).

APPLICABLE SAFETY ANALYSES

The initial conditions of ~~design basis~~ Δ (Ref. 10) transient ~~and accident~~ Δ Accident (DBA) and analyses in the FSAR, Chapter 6, ~~"Engineered Safety Features"~~, and Chapter 15, ~~"Accident Analyses"~~, assume that Engineered Safety Features Δ (ESF) systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for ~~the~~

(continued)

The LCO is modified by a Note which indicates that Vital Battery V may be substituted for any of the required vital batteries. However, the fifth battery cannot be declared OPERABLE until it is connected electrically in place of another battery and it has satisfied applicable Surveillance Requirements

1

7 ~~DGs~~, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining the DC sources OPERABLE during accident conditions in the event of:

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

b. A worst-case single failure.

△ DC ~~S~~ources ~~— Operating~~ satisf~~ies~~ Criterion 3 of the NRC ~~Interim~~ Policy Statement~~. as described in References 10 and 11.~~

LCO

A the ~~The~~ DC electrical power subsystem~~s~~ associated channels, (each ~~subsystem~~ channel consisting of ~~[two] batteries,~~ a battery bank battery charger ~~[for each battery]~~ and the corresponding control equipment and interconnecting cabling within the ~~train~~ channel) are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated ~~Design Basis Accident~~ ~~(DBA)~~. △ Loss of any ~~train~~ DC electrical ~~power subsystem~~ channel does not prevent the minimum safety function from being performed (Ref. 4).

An OPERABLE DC electrical power subsystem requires ~~all required~~ 125 VDC Vital batteries~~,~~ and respective chargers to be operating and connected to their associated DC bus~~(es)~~ I-IV. A

APPLICABILITY

The DC electrical power sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure safe unit operation and to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

(continued)

BASES (continued)

b. Adequate core cooling is provided, and containment integrity and other vital functions are maintained in the event of a postulated DBA.

DC electrical power requirements for MODES 5 and 6 are addressed in the Bases for ~~Specification~~ LCO 3.8.5, "DC Sources Shutdown." Δ

ACTIONS

A.1

Δ

Condition A represents one train with a loss of ability to completely respond to an event, and a potential loss of ability to remain energized during normal operation. It is therefore, imperative that the operator's attention focus on stabilizing the ~~plant~~ unit, minimizing the potential for complete loss of DC power to ~~be~~ the affected train. The 2 hour limit is consistent with the allowed time for an inoperable DC Distribution System train.

If one of the required DC electrical power subsystems is inoperable (e.g., inoperable battery, inoperable battery charger(s), or inoperable battery charger and associated inoperable battery), the remaining DC electrical power subsystem has the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst-case single failure~~,~~ of the OPERABLE subsystem would, however, result in ~~the complete loss of the remaining 125 Vdc electrical power subsystems with attendant loss of ESF functions,~~ a situation where the ability of the 125V DC electrical power subsystem to support its required ESF function is not assured, continued power operation should not exceed 2 hours. The 2-hour Completion Time is based on Regulatory Guide 1.93 (Ref. 6) and reflects a reasonable time to assess unit status as a function of the inoperable DC electrical power subsystem and, if the DC electrical power subsystem is not restored to OPERABLE status, prepare to effect an orderly and safe unit shutdown.

8

B.1 and B.2

Δ

If the inoperable DC electrical power subsystem cannot be restored to OPERABLE status ~~in 6 hours,~~ within the required Completion Time, the unit must be ~~placed in~~ brought to a MODE in which the LCO does not apply. ~~This is done by placing the unit in~~ To achieve this status the unit must be brought to at least MODE 3 within 6 hours and ~~in~~ to MODE 5 within 36 hours. The allowed Completion Times are

(continued)

BASES (continued)

unit conditions

reasonable, based on operating experience ~~related~~ to ~~the amount of time required to~~ reach the required ~~MODES~~ from full power in an orderly manner and without challenging unit systems. The Completion Time to bring the unit to MODE 5 is consistent with the time required in Regulatory Guide 1.93 (Ref. 6).

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery (or battery cell) and maintain the battery (or a battery cell) in a fully charged state. The voltage requirements are

(A) critical

based on the ~~nominal~~ design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations. The 7-day Frequency is consistent with manufacturer's recommendations and IEEE-450 (Ref. 7).

SR 3.8.4.~~2~~3

Visual inspection to detect corrosion of the battery cells and connections, or measurement of the resistance of each inter-cell, inter-rack, inter-tier, and terminal connection, provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance.

(11)

The limits established for this SR shall be no more than 20% above the resistance as measured during installation or not above the ceiling value established by the manufacturer.

The Surveillance Frequency for these inspections, which can detect conditions that can cause power losses due to resistance heating, is 92 days. This Frequency is considered acceptable based on operating experience related to detecting corrosion trends.

(2)

SR 3.8.4.2

Verifying that the alternate feeder breakers to each required battery charger is open ensures that independence between the power trains is maintained. The 7day Frequency is based on engineering judgement, is consistent with procedural controls governing breaker operation, and ensures correct breaker position.

(continued)

BASES (continued)

SR 3.8.4.~~3~~ 4

Visual inspection of the battery cells, cell plates, and battery racks provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance.

STET

~~This SR is consistent with IEEE-450 (Ref. 7), which recommends detailed visual inspection of cell condition and rack integrity on a yearly basis.~~

SR 3.8.4.~~4~~ 5 and SR 3.8.4.~~5~~ 6

Visual inspection and resistance measurements of inter-cell, inter-rack, inter-tier, and terminal connections provide an indication of physical damage or abnormal deterioration that could indicate degraded battery condition. The anti-corrosion material is used to help ensure good electrical connections and to reduce terminal deterioration. The visual inspection for corrosion is not intended to require removal of and inspection under each terminal connection. The removal of visible corrosion is a preventive maintenance SR. The presence of visible corrosion does not necessarily represent a failure of this SR provided visible corrosion is removed during performance of SR 3.8.4.~~4~~ 5

A

For the purposes of trending, inter-cell and inter-tier connections are measured from battery post to battery post and inter-rack and terminal connections are measured from the terminal lug to battery post.

The connection resistance limits for SR 3.8.4.~~5~~ 6 shall be no more than 20% above the resistance as measured during installation, or not above the ceiling value established by the manufacturer.

11

STET

~~The Surveillance Frequency of 12 months is consistent with IEEE-450 (Ref. 7), which recommends cell-to-cell and terminal connection resistance measurement on a yearly basis.~~

4

expected worst case loading of 132 amps plus a margin of approximately 15%

SR 3.8.4.~~6~~ 7

150 4 4 4

This SR requires that each battery charger be capable of supplying ~~[400]~~ amps and [125] V for ≥ ~~[8]~~ hours. These requirements are based on the ~~design capacity of the chargers~~ (Ref. 4). According to Regulatory Guide 1.32 (Ref. 8), the battery charger supply is required to be based

(continued)

BASES (continued)

The SR is modified by two Notes. The reason for Note 1 is that performing the Surveillance would remove a required DC electrical power subsystem from service, perturb the electrical distribution system, and challenge safety systems. Note 2 is added to this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

on the largest combined demands of the various steady-state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensures that these requirements can be satisfied. ~~This Surveillance is required to be performed during MODES 5 and 6 since it would require the DC electrical power subsystem to be inoperable during performance of the test.~~

The Surveillance Frequency is acceptable, given the unit conditions required to perform the test and the other administrative controls existing to ensure adequate charger performance during these 18-month intervals. In addition, this Frequency is intended to be consistent with expected fuel-cycle lengths.

SR 3.8.4.~~7~~8

A battery-service test is a special test of ~~the~~ battery~~'s~~ capability, "as found," to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length ~~should~~ correspond, to the design duty cycle requirements as specified in ~~Reference~~ ~~4~~ 11 5.

A

The Surveillance Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 8) and Regulatory Guide 1.129 (Ref. 9), which state that the battery-service test should be performed during refueling operations or at some other outage, with intervals between tests not to exceed 18 months.

This SR is modified by three Notes. The reason for Note 1 is to ~~A Note to SR 3.8.4.7~~ allow~~s~~ the once-per-60-months performance of SR 3.8.4.~~8~~9 in lieu of SR 3.8.4.~~7~~8. This substitution is acceptable because SR 3.8.4.~~8~~9 represents a more severe test of battery capacity than SR 3.8.4.~~7~~8.

~~This Surveillance is required to be performed during MODES 5 and 6 since it would require a DC electrical power subsystem to be inoperable during performance of the test.~~

The reason for Note 2 is that performing the Surveillance would remove a required DC electrical power subsystem from service, perturb the electrical distribution system, and challenge safety systems. Note 3 is added to this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

(continued)

BASES (continued)

This SR is modified by two Notes. The reason for Note 1 is that performing the Surveillance would remove a required DC electrical power subsystem from service, perturb the electrical distribution system, and challenge safety systems. Note 2 is added to this SR to acknowledge that credit may be taken for unplanned events that satisfy this SR.

SR 3.8.4.~~8~~9

A battery-performance test is a test of constant current capacity of a battery~~, normally done in the "as found" condition,~~ after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

The acceptance criteria for this Surveillance ~~is~~ are consistent with IEEE-450 (Ref. 7) and IEEE-485 (Ref. 5). These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

The Surveillance Frequency for this test is 60 months, or every 12 months if the battery shows degradation or has reached 85% of its expected life. However, if the battery has reached 85% of its expected life, shows no ~~no~~ degradation, and retains 100% capacity, a Surveillance Frequency of 24 months is permitted. Degradation is indicated, according to IEEE-450 (Ref. 7), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is below the manufacturer's rating. All these Frequencies are consistent with the recommendations in IEEE-450 (Ref. 7).

~~This Surveillance is required to be performed during MODES 5 and 6, since it would require the DC electrical power subsystem to be inoperable during performance of the test.~~

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17, ~~Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 17~~, "Electric Power System."

2. Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems," U.S. Nuclear Regulatory Commission, March 10, 1971.

3. IEEE-308 [~~1978~~ 1971], "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations," ~~Institute of Electrical and Electronic Engineers~~.

(continued)

9 4. Watts Bar FSAR, Section 8.3.2, "DC Power System."

BASES (continued)

9 10. ~~4~~ Watts Bar FSAR, Section 15, "Accident Analysis" and Section 6, "Engineered Safety Features."

5. IEEE-485 [1983], "Recommended Practices for Sizing Large Lead Storage Batteries for Generating Stations and Substations," Institute of Electrical and Electronic Engineers, June 1983.

6. Regulatory Guide 1.93, "Availability of Electric Power Sources," U.S. Nuclear Regulatory Commission, December 1974.

7. IEEE-450 [1987], "IEEE Recommended Practice for Maintenance Testing and Replacement of Large Lead Storage Batteries for Generating Stations and Subsystems," Institute of Electrical and Electronic Engineers.

8. Regulatory Guide 1.32, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Units," February 1977, U.S. Nuclear Regulatory Commission.

9. Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Generating Stations and Subsystems," U.S. Nuclear Regulatory Commission, December 1974.

~~10. Thomas E. Murley (NRC) letter to W.S. Wilgus dated May 9, 1989, 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."~~

~~11. WCAP - 11618, "MERITS Program - Phase II, Task 5, Criteria Application", dated November 1987, including Addendum 1, dated April 1989.~~

A 11. TVA Calculation EEB-MS-TI11-0003, "125 VDC Vital Battery and Charger Capacity Evaluation."

(continued)

JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.8.4

Generic Justifications

A. Modified to reflect WBN plant specific design.

B. Modified to reflect WBN plant specific terminology.

Specific Justifications

1. LCO modified to reflect WBN specific design of DC sources. A fifth 125v DC Vital battery system is intended to serve as a temporary replacement for any one of the four normal 125 VDC vital batteries (see FSAR 8.3.2.1.1).

2. The WBN DC sources specifications include an additional Surveillance (i.e. additional relative to the RSTS standard) to verify that the alternate feeder breakers to each required battery charger are open. This proposed change was intended to address SER 8.3.1.7 regarding alarm indications for alternate feeders (see TVA letters to NRC dated 6/21/85 & 5/3/85).

3. The latest draft of IEEE Std 450 which has been approved by the IEEE Battery Working Group and is up for ballot to the IEEE Power Generation Committee contains the following change to Section 5.2 of IEEE Std 450:

   5.2 PERFORMANCE

   > Annual performance test of battery capacity should be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. However, if the battery delivers a capacity of 100% or greater of the manufacturer's rated capacity, performance testing at two year intervals is acceptable until the battery shows signs of degradation. Degradation is indicated when the battery capacity drops more than 10% from its capacity on the previous performance test, or is below 90% of the manufacturer's rating.

   Based on this proposed change to IEEE Std. 450, the existing Surveillance Frequency for SR 3.8.4.9 was modified to reflect this change.

4. TVA's reasons for proposing these test parameters are outlined in a March 24, 1985 letter from TVA (D.E. McCloud) to the NRC (E. Adensam). TVA's position is that Tech. Spec. limits should be based on actual safety requirements rather than on design capability of the chargers.

5. Information has been added to clarify and explain the specific design features of the DC power system that provide for redundancy and separation. This explanation is consistent with the FSAR and with general understanding of plant personnel.

JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.8.4 (cont.)

6. More accurate information explicitly describing the capacity of the Battery Charger has been added. This information is found in FSAR section 8.3.2.1.1 Page 8.3-58.

7. Modification has been made to reflect the fact that DG batteries are separate from the batteries included in DC sources subsystem (i.e. LCO 3.8.4).

8. These modifications have been made to clarify:

   1) Since there are 2 channels per train, the worst case single failure would have to occur in one of the channels of the opposite train.

   2) Even with a worst case failure, a complete loss of the remaining 125 VDC electric power subsystem would not necessarily result. A more accurate description is that the ability to support ESF functions is not assured

9. Reference added for plant specific design information.

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

11. In a normal inter-cell connection of this type, the typical inter-cell connection resistance values are equal to approximately 30 x 10^{-6} ohms. A value of 80 x 10^{-6} ohms has been selected as an operability point for the batteries. Based on this value, the maximum heat dissipation that the connection would see would be 47 watts (at 885 amperes) for the first minute of the duty cycle and approximately 15 watts (at 500 amperes) for the remainder of the duty cycle. This heat is spread over four 1 inch square lead battery posts which will easily dissipate this minor amount of thermal energy. For inter-tier connections, a value of 80 x 10^{-6} ohms has been selected and for inter-rack and terminal connections, a value of 100 x 10^{-6} ohms. The logic used to justify the inter-cell connections is the same for the inter-tier, inter-rack and terminal connections.

    For the purposes of trending, inter-cell and inter-tier connections are measured from battery post to battery post and inter-rack and terminal connections are measured from the terminal lug to the battery post.

JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.8.4 (cont.)

12. The phrase "and tight," has been removed from the Surveillance Requirement. IEEE Std. 450-1987 removed the recommendation for annual torquing from the maintenance section of the standard. The reason for the removal of this recommendation was that the industry had found that detailed resistance measurements were a better indication of the batteries ability to perform its intended function. The manufacturers of Class 1 E batteries have accepted 450's recommendation and removed this recommendation from their battery manuals. Also, the removal of the annual torquing recommendation has removed a possible failure mode from the batteries, shorting out a cell or group of cells if the torque wrench is dropped. Dropping the annual torquing requirement will also reduce cell post deformation caused by improper torquing of the electrical connections.

13. Change to delete information not applicable to WBN design.

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources – Shutdown

LCO 3.8.5 DC electrical power subsystem shall be OPERABLE to support the DC electrical power distribution subsystem(s) required by LCO 3.8.10. "Distribution Systems – Shutdown."

APPLICABILITY: ~~In~~ MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required DC electrical power subsystems inoperable.	A.1 Declare affected required feature(s) inoperable ~~and enter applicable Conditions and Required Actions of the LCOs for the inoperable required feature(s)~~.	Immediately
	AND	
	A.1.2 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
		(continued)

6

---- NOTE ----
Vital Battery V may be Substituted for any of the required vital batteries.
1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
	AND	
	A.2.4 Initiate action to ~~S~~suspend operations involving positive reactivity additions which would exceed limits specified in LCO 3.1.2, "Shutdown Margin," and LCO 3.9.1, "Boron Concentration." 6	Immediately
	AND	
	A.2.5 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	For DC sources ~~all equipment~~ required to be OPERABLE the following SRs are ~~required to be met:~~ applicable: 6 SR 3.8.4.1, SR 3.8.4.2, SR 3.8.4.3, SR 3.8.4.4, SR 3.8.4.5, SR 3.8.4.6, SR 3.8.4.7. SR 3.8.4.8, SR 3.8.4.9 2	In accordance with applicable SRs

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources — Shutdown

BASES

BACKGROUND

A description of the DC sources is provided in the Bases for ~~Specification~~ 3.8.4, "DC Sources — Operating."
△ LCO

APPLICABLE SAFETY ANALYSES

(Ref. 2)
Accident (DBA) and

The initial conditions ~~of design basis~~ transient ~~and accident~~ analyses in the FSAR, ~~Chapter 6~~ ~~"Engineered Safety Features"~~ and Chapter 15 ~~"Accident Analyses")~~, assume that (ESF) systems are OPERABLE. The DC ~~e~~lectrical ~~p~~ower ~~s~~ystem provides normal and emergency DC electrical power ~~for the DGs~~, emergency auxiliaries, and control and switching during all MODES of operation.

△ engineered safety features
③

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum DC electrical power sources during MODES 5 and 6, and during movement of irradiated fuel assemblies, ensures that ~~(Ref. 1)~~:

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel-handling accident.

The DC ~~S~~ources satisfy Criterion 3 of the NRC ~~Interim~~ Policy Statement. ~~(Ref. 2 and 3)~~.

LCO

Ⓑ channels
channel

DC electrical power subsystem~~s~~, each ~~subsystem~~ consisting of ~~two battery, one battery charger per battery~~ and ~~the~~ corresponding control equipment and interconnecting cabling within the ~~train~~, are required OPERABLE to support required

Ⓑ a battery bank, associated battery charger,
channel Ⓑ
to be
△

(continued)

As a minimum, one train (i.e., channels I and III, or II and IV) shall be OPERABLE.

4

BASES (continued)

"Distribution System – Shutdown."

trains of Distribution System ~~divisions~~ required OPERABLE by LCO 3.8.10. This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel-handling accidents, and inadvertent reactor vessel draindown).

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies, provide assurance that:

The LCO is modified by a Note which indicates that Vital Battery V may be substituted for any of the required Vital Batteries. However, the fifth battery cannot be declared OPERABLE until it is connected electrically in place of another battery and it has satisfied applicable Surveillance Requirements.

1

a. Required ~~F~~features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core in case of an inadvertent draindown of the reactor vessel;

b. Required ~~F~~features needed to mitigate a fuel-handling accident are available;

c. Required ~~F~~features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

DC electrical power requirements for MODES 1, 2, 3 and 4 are covered in ~~Specification~~ LCO 3.8.4. ~~"DC Sources – Operating."~~

ACTIONS

A.1, A.2.1, A.2.2, A.2.3 ~~and~~ A.2.4 and A.2.5

If two trains are required per LCO 3.8.10, ~~Distribution System – Shutdown,~~ the remaining train with DC power available may be capable of supporting sufficient systems to allow continuation of CORE ALTERATIONS, fuel movement, ~~and~~/or operations with a potential for draining the reactor vessel. By allowing the option to declare required features inoperable with associated DC power source(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCO ACTIONS. In many instances this option may ~~would likely~~ involve undesired administrative

(continued)

BASES (continued)

Notwithstanding performance of the above conservative Required Actions, the unit is still without sufficient DC power sources to operate in a safe manner. Therefore, action must be initiated to restore the minimum required DC power sources and continue until the LCO requirements are restored.

efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and any activities that could ~~potentially~~ result in inadvertent draining of the reactor vessel, and operations involving stop ~~positive reactivity additions~~). The required action to suspend positive reactivity additions does not preclude actions to maintain ~~or~~ increase reactor vessel inventory provided the required shutdown margin is maintained. INSERT I-2 6

Suspension of these activities shall not preclude completion of actions to establish ~~a~~ the safe conservative condition. These actions ~~will~~ minimize probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required DC electrical power subsystems and to continue this action until restoration is accomplished in order to provide the necessary DC electrical power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required DC electrical power subsystems should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

SR 3.8.4.1 requires performance of all Surveillances required by SR 3.8.4.1 through SR 3.8.4.~~8~~ 9. Therefore, see the corresponding Bases for ~~Specification~~ LCO 3.8.4 for a discussion of each SR. 2

REFERENCES 5

2 ~~1.~~ Watts Bar FSAR, Section 15, "Accident Analysis" and Section 6 "Engineered Safety Features."

~~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."~~

1. Watts Bar FSAR, Section 8.0, "Electric Power."

(continued)

INSERT

A.2.4 I-2

Required Action~~s~~ A precludes operations involving systems that contain large volumes of water (i.e. CVCS, SIS, and RWST) at reduced boron concentrations with respect to the RCS that could dilute the boron concentration of the RCS to less than that required to maintain the SDM requirements of LCO 3.1.2 in MODE 5 or less than the boron concentration specified in the COLR in MODE 6. This Required Action does not preclude positive reactivity additions that cannot reduce the SDM or boron concentration to less than the limits specified in LCO 3.1.2 or the COLR. The addition of water with a boron concentration greater than that required to maintain the reactor shutdown within the requirements of LCO 3.1.2 or the COLR, but less than the RCS, is permitted. Positive reactivity additions such as small volume chemical additions and normal plant cooldowns are also permitted as long as the SDM limits are met.

BASES (continued)

~~3. WCAP-11618, "MERITS Program - Phase II, Task 5, Criteria Application", dated November 1987, including Addendum 1, dated April 1989.~~

(continued)

JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.8.5

Generic Justifications

A. Modified to reflect WBN plant specific design.

B. Modified to reflect WBN plant specific terminology.

Specific Justifications

1. LCO modified to reflect WBN specific design of DC sources. A fifth 125v DC Vital battery system is intended to serve as a temporary replacement for any one of the four normal 125 VDC vital batteries (see FSAR 8.3.2.1.1).

2. The WBN DC sources specifications include an additional Surveillance (i.e. additional relative to the RSTS standard) to verify that the alternate feeder breakers to each required battery charger are open. This proposed change was intended to address SER 8.3.1.7 regarding alarm indications for alternate feeders (see TVA letters to NRC dated 6/21/85 & 5/3/85).

3. Modification has been made to reflect the fact that DG batteries are separate from the batteries included in DC sources subsystem (i.e. LCO 3.8.4).

4. Wording added to clarify the fact that an entire train of DC subsystem channels is required to be OPERABLE.

5. Reference added for plant specific design information.

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

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Cell Parameters

1 125 V Vital Batteries and 125 V Diesel Generator (DG) Batteries

LCO 3.8.6 Battery cell parameters for ~~[Train A] and [Train B] batteries~~ shall be within the Category A and Category B limits of Table 3.8.6-1. △

APPLICABILITY: When associated DC electrical power subsystems and DGs 1 are required to be OPERABLE~~. per LCO 3.8.4 and LCO 3.8.5.~~

ACTIONS

---NOTE---
Separate Condition entry is △ allowed for each ~~battery.~~ Bank B

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

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR Δ One or more batteries with average electrolyte temperature of the ~~pilot~~ representative cells ~~is not within limits.~~ OR One or more batteries with battery cell parameters not within Category C values.	B.1 Declare associated battery inoperable. ~~and enter applicable Conditions and Required Actions for DC electrical power subsystems,~~ ~~LCO 3.8.4, "DC Sources-Operating," and LCO 3.8.5, "DC Sources-Shutdown," or for inoperable DGs, LCO 3.8.1, "AC Sources-Operating," and LCO 3.8.2, "AC Sources-Shutdown."~~ 2 X	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	Verify battery cell parameters meet Table 3.8.6-1 Category A limits.	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.6.2	Verify battery cell parameters meet Table 3.8.6-1 Category B limits. (~~[illegible]~~ 113.5 V for Vital Battery V) ① ④ an inadvertent (143V for DG Batteries)	92 days **AND** Once within 24 hours after a battery discharge ≤ [110] V ~~volts~~ **AND** Once within 24 hours after ~~a~~ an inadvertent battery overcharge ≥ [150] V ~~volts~~
SR 3.8.6.3	Verify average electrolyte temperature of representative cells is ≥ [60]°F. 9 9 for the vital batteries and ≥50°F for the DG batteries. Ⓐ	92 days

Table 3.8.6-1 (page 1 of 1)
Battery Cell Parameters Requirements △

PARAMETER		CATEGORY A: LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: LIMITS FOR EACH CONNECTED CELL	CATEGORY C: ALLOWABLE VALUE FOR EACH CONNECTED CELL
Electrolyte Level		> Minimum level indication mark, and ≤ 1/4 inch above maximum level indication mark[a]	> Minimum level indication mark, and ≤ 1/4 inch above maximum level indication mark[a]	Above top of plates, and not overflowing
Float Voltage		≥ 2.13 V	≥ 2.13 V	> 2.07 V
Specific Gravity[b][c]	CELL	≥ [1.200]	≥ [1.195] AND	Not more than 0.020 below average of all connected cells AND
	BATTERY		Average of all connected cells > [1.205]	Average of all connected cells ≥ [1.195]

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum level during equalizing charges provided it is not overflowing.

(b) Corrected for electrolyte temperature and level. Level correction is not required, however, when battery charging is < [2] 9.3 amps when on float charge for vital batteries and < 0.5 amps for DG batteries

(c) Or battery charging current is < N/A [2] amps when on float charge. This is acceptable only during a maximum of ~~[7] days~~ 31 days following a battery recharge.

A

3

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.6 Battery Cell Parameters

1 A discussion of the DG batteries is provided in the Bases for LCO 3.8.1, "AC Sources - Operating."

BASES

BACKGROUND

This LCO ~~3.8.6, Battery Cell Parameters~~ delineates the limits on electrolyte temperature, level, float voltage and specific gravity for the DC ~~Power Source~~ subsystem and diesel generator (DG) batteries. A discussion of 1 the 125V vital ~~these~~ batteries and their OPERABILITY requirements is provided in the Bases for LCO 3.8.4, "DC Sources - Operating," and LCO 3.8.5, "DC Sources - Shutdown."

APPLICABLE SAFETY ANALYSES

The initial conditions of ~~design basis~~ Accident (DBA) and transient ~~and accident~~ analyses in FSAR (Ref. 2) Chapter 6, ~~Engineering Safety Features~~, and Chapter 15, ~~Accident Analyses~~, assume ESF systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for 1 ~~the DGs~~, emergency auxiliaries and control and switching during all MODES of operation.

△ engineered safety features

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining at least one train of DC sources OPERABLE during accident conditions, in the event of:

△ a. ~~an~~ assumed loss of all offsite AC power or all onsite AC power; and

△ b. ~~a~~ worst-case single failure.

△ Battery cell parameters satisfy ~~the requirements of Selection~~ Criterion 3 of the NRC ~~Interim~~ Policy Statement ~~as applied to DC Sources~~.

LCO

△ Battery cell parameters must remain within acceptable limits to ensure availability of the required DC power to shutdown the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated ~~design basis accident~~ DBA. Electrolyte limits are conservatively established, allowing continued DC electrical system function even with Category A and Category B limits not met.

(continued)

BASES (continued)

APPLICABILITY

1 subsystems and DGs are are

The battery cell parameters ~~is~~ required ~~solely~~ for the support of the associated DC electrical power subsystems. and DGs Therefore, battery electrolyte is only required when the DC Power ~~Source is~~ required to be OPERABLE. Refer to the Applicability discussion in Bases for LCO 3.8.4~~, DC Sources — Operating~~ and LCO 3.8.5. ~~DC Sources — Shutdown.~~

1 LCO 3.8.1, LCO 3.8.2,

ACTIONS

A.1, A.2 and A.3

With one or more cells in one or more batteries not within limits (i.e., Category A limits not met, or Category B limits not met, or Category A and B limits not met) but within the allowable value (Category C limits are met) specified in Table 3.8.6-1, in the accompanying LCO, △ operation is permitted for a limited period since sufficient capacity exists to perform the intended function.

The ~~pilot~~ cell electrolyte level and float voltage are 1 required to be verified to meet the Category C allowable values within ~~one~~ hour (Required Action A.1). This check will provide a quick indication of the status of the remainder of the battery cells. One hour provides time to inspect the electrolyte level and to confirm the float voltage of the ~~pilot~~ cells. One hour is considered a reasonable amount of time to perform the required verification.

Verification that the Category C allowable values are met (Required Action A.2) provides assurance that during the time needed to restore the parameters to the Category A and Category B limits, the battery will still be capable of performing its intended function. A period of 24 hours is allowed to complete the required verification because specific gravity measurements must be obtained for each connected cell. Taking into consideration both the time required to perform the required verification and the assurance that the battery cell parameters are not severely degraded, this time is considered reasonable.

INSERT 2 3

Continued operation is only permitted for 31 days before battery cell parameters must be restored to within Category A and Category B limits. ~~Taking into~~ With the consideration that

(continued)

INSERT

In the event the battery cell parameter is out of limits and a battery recharge is desired as corrective action, footnote "c" of Table 3.8.6-1 may be utilized in lieu of specific gravity measurements. In this event, once Required Action A.2 has been satisfied using float current in lieu of specific gravity, the footnote does not require additional specific gravity testing.

BASES (continued)

to Δ

while battery capacity is degraded, sufficient capacity exists to perform the intended function and allow time to fully restore the battery cell parameters to normal limits, this time is acceptable prior to declaring the Battery inoperable.

B.1

With one or more batteries with one or more battery cell parameters outside the Category C allowable value for any connected cell, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding DC electrical power subsystem must be declared inoperable. Additionally, other potentially extreme conditions, such as not completing the Required Actions of Condition A within the required Completion Time or average electrolyte temperature of representative cells falling below 60°F, will also be cause for immediately declaring the associated DC electrical power subsystem inoperable.

A

for the vital batteries or 50°F for DG batteries

or DG 1

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

that

Δ This SR verifies Category A battery cell parameters are consistent with IEEE-450 (Ref. 1), which recommends regular battery inspections (at least one per month) including voltage, specific gravity, and electrolyte temperature of ~~pilot~~ cells.

~~representative~~ pilot 6

4 an inadvertent

SR 3.8.6.2

The quarterly inspection of specific gravity and voltage is consistent with IEEE-450 (Ref. 1). In addition, within 24 hours of a battery discharge < 110 V or ~~a~~ battery overcharge > 153 V, the battery must be demonstrated to meet Category B limits. This inspection is also consistent with IEEE-450 (Ref. 1), which recommends special inspections following a severe discharge or overcharge, to ensure that no significant degradation of the battery occurred as a consequence of such discharge or overcharge.

(143 V for DG batteries)

(continued)

A

BASES (continued)

for the vital batteries
and ≥ 50° F for the DG batteries

SR 3.8.6.3

≥ that

This Surveillance, verification that the average temperature of representative cells is ~~≥~~ 60°F, is consistent with a recommendation of IEEE-450 (Ref. 1), ~~which~~ states that the temperature of electrolytes in representative cells should be determined on a quarterly basis.

Lower-than-normal temperatures act to inhibit or reduce battery capacity. This SR ensures that the operating temperatures remain within an acceptable operating range. This limit is based on manufacturer's recommendations.

Table 3.8.6-1

This table delineates the limits on electrolyte level, float voltage, and specific gravity for three different categories. The meaning of each category is discussed below.

representative pilot 6

Category A defines the normal parameter limit for each designated ~~pilot~~ cell in each battery. The cells selected as ~~pilot~~ cells are those whose temperature, voltage and electrolyte specific gravity approximate the state of charge of the entire battery.

The Category A limits specified for electrolyte level are based on manufacturer's recommendations and are consistent with the guidance in IEEE-450 (Ref. 1), with the extra 1/4 inch allowance above the high-water-level indication for operating margin to account for temperatures and charge effects. In addition to this allowance, Footnote a. to Table 3.8.6-1 permits the electrolyte level to be above the specified maximum level during equalizing charge, provided it is not overflowing. These limits ensure that the plates suffer no physical damage, and that adequate electron transfer capability is maintained in the event of transient conditions. IEEE-450 (Ref. 1) recommends that electrolyte level readings should be made only after the battery has been at float charge for at least 72 hours.

5

that

The Category A limit specified for float voltage is ≥ 2.13 V per cell. This value is based on the recommendations of IEEE-450 (Ref. 1), ~~which~~ state that prolonged operation of cells ~~below~~ < 2.13 V can reduce the life expectancy of cells.

(continued)

BASES (continued)

~~representative~~ pilot (6)

The Category A limit specified for specific gravity for each ~~pilot~~ cell is $\geq$ 1.200 (0.015 below the manufacturer's fully charged nominal specific gravity or a battery charging current that had stabilized at a low value). This value is characteristic of a charged cell with adequate capacity. According to IEEE-450 (Ref. 1), the specific gravity readings are based on a temperature of 77°F (25°C).

The specific gravity readings are corrected for actual electrolyte temperature and level. For each 3°F (1.67°C) above 77°F (25°C), 1 point (0.001) is added to the reading; 1 point is subtracted for each 3°F below 77°F. The specific gravity of the electrolyte in a cell will increase with a loss of water due to electrolysis or evaporation. Footnote b. to Table 3.8.6-1 requires the above-mentioned correction for electrolyte level and temperature, with the exception that level correction is not required when battery charging current is < 2 amps on float charge. This current provides, in general, an indication of overall battery condition.

(6) ~~representative~~ pilot

Because of specific-gravity gradients that are produced during the recharging process, delays of several days ~~3 to 7~~ may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific-gravity measurement for determining the state of charge of the designated ~~pilot~~ cell. This phenomenon is discussed in IEEE-450 (Ref. 1). Footnote c. to Table 3.8.6-1 allows the float charge current to be used as an alternate to specific gravity for up to ~~7 days~~ 31 days following a battery equalizing recharge. (3)

INSERT (2)(3)

Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out.

The Category B limits specified for electrolyte level and float voltage are the same as those specified for Category A and have been discussed above. The Category B limit specified for specific gravity for each connected cell is $\geq$ 1.195 (0.020 below the manufacturer's fully charged, nominal specific gravity) with the average of all connected cells ~~$\geq$~~ 1.205 (0.010 below the manufacturer's fully charged, nominal specific gravity). These values are based on manufacturer~~'s~~ recommendations. The minimum specific-gravity value required for each cell ensures that

(continued)

INSERT

Recharge as used in this footnote is synonymous with a battery equalizing charge. The allowance to utilize charging current is limited to 31 days ~~after the completion of the equalizing~~ after the completion of the equalizing charge and return to float charge. The 31 days allows time for the specific gravity to stabilize and is consistent with Required Action A.3.

BASES (continued)

the effects of a highly charged or newly installed cell will not mask overall degradation of the battery. Footnote b. to Table 3.8.6-1 requires correction of specific gravity for electrolyte temperature and level. This level correction is not required when battery charging current is < 2 amps on float charge.

Δ Category C defines the allowable values for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category C allowable value, the assurance of sufficient capacity described above no longer exists and the battery must be declared inoperable.

Δ The Category C allowable values specified for electrolyte level (above the top of the plates and not overflowing) ensure that the plates suffer no physical damage and maintain adequate electron transfer capability. The Category C allowable value for float voltage is based on IEEE-450 (Ref. 1), which states that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement.

Δ The Category C allowable value of average specific gravity ≥ 1.195 is based on manufacturer's recommendations ~~(≥ 1.195~~, (0.020 below the manufacturer's recommended fully charged, nominal specific gravity). In addition to that limit, it is required that the specific gravity for each connected cell must be no less than 0.020 below the average of all connected cells. This limit ensures that the effect of a highly charged or new cell will not mask overall degradation of the battery. The footnotes to Table 3.8.6-1 that apply to Category A specific gravity are also applicable to Category C specific gravity.

REFERENCES

Δ 1. IEEE-450 1980 ~~1980~~, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

(A) 2. Watts Bar FSAR, Section 15, "Accident Analysis," and Section 6, "Engineered Safety Features."

(continued)

BASES (continued)

△ ~~2. IEEE-308-1978, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations."~~

(continued)

JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.8.6

Generic Justifications

A. Modified to reflect WBN plant specific design.

B. Modified to reflect WBN plant specific terminology.

Specific Justifications

1. Modification has been made to reflect the fact that DG batteries are separate from the batteries included in DC sources subsystem (i.e. LCO 3.8.4).

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

3. Watts Bar is proposing a 31 day allowance for footnote "C" to Table 3.8.6-1 which is consistent with Required Action A.3.

4. The words "an inadvertent" have been added to the overcharge conditions under the FREQUENCY column. By strict Literal Interpretation of these STS's, you would have to perform a complete quarterly inspection, excluding resistance measurements, to declare the battery operable after a lead-calcium battery is "High Level Equalized." The use of this type of equalization is becoming more and more common on lead-calcium stationary batteries to correct electrolyte problems. There are basically three types of equalization that are applied to lead-calcium stationary batteries. The table below shows these three types and their voltage ranges.

Type of Battery Charge	Voltage Range (volts per cell) on Equalization	Voltage Range for a 60 cell Battery (vdc)
Normal Float	2.20 to 2.25	132.0 to 135.0
Low Level Equalization	2.33 to 2.38	139.8 to 142.8
High Level Equalization	2.50 to 2.60	150.0 to 160.0

When lead-calcium stationary batteries are normally equalized, it is performed on line and at the "low Level." However, due to the high internal impedance of the lead-calcium type of battery, it can take a number of weeks of this type of equalization to provide any useful results. Recently, manufacturer's have given a number of utilities permission to perform "High Level Equalization" for limited time durations on "their" lead-calcium stationary batteries to quickly correct electrolyte problems with the batteries. This "High Level Equalization" can not be considered a "severe overcharge" because it is 1) recommended by the manufacturer, 2) is only used for limited time durations and 3) is a controlled equalization of the battery. Therefore, by adding the words, "an inadvertent", this will eliminate this type of problem and only require a full set of readings when there has been a problem with the systems.

5. IEEE 450 does not require a 72 hour delay per Section 4.3.

6. "Representative Cells" as used in IEEE-450 refers to the quarterly temperature surveillance for every 6th cell (suggested) for the battery. Category A limits in IEEE-450 are for a pilot cell, i.e., one single cell of a battery as specified by Section 4.3.1(8).

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters – Operating

LCO 3.8.7 The required Train A and Train B inverters shall be OPERABLE.

1

~~-----------------------------NOTE-----------------------------~~
~~[One/Two] inverters may be disconnected from [its/their] associated DC bus(es) for ≤ 24 hours to perform an equalizing charge on [its/their] associated [common] battery providing:~~

~~a. The associated AC vital bus(es) [is/are] energized from [its/their] [Class 1E] constant voltage source transformer[s]; and~~

~~b. All other AC vital buses for both trains are energized from their associated OPERABLE inverters.~~
~~--~~

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

ACTIONS

B associated 120V AC instrumentation distribution Panel

CONDITION	REQUIRED ACTION		COMPLETION TIME
2 A. One ~~[required]~~ inverter inoperable.	6 ~~A.1~~	~~Power AC vital bus from its [Class 1E] constant voltage source transformer.~~	~~2 hours~~
	~~AND~~		
	A.~~2~~ 1	Restore ~~[required]~~ inverter to OPERABLE status.	24 hours

(continued)

-------- NOTE --------
Enter the applicable Conditions and Required Actions of LCO 3.8.9, "Distribution System - Operating" for ~~[illegible]~~ AC vital Bus.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct inverter voltage, ~~[~~frequency~~]~~, and alignment to required AC vital bus~~es~~	7 days

3

and from associated vital battery board and 480 V shutdown board.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 Inverters – Operating

BASES

there is one inverter per AC vital bus making a total of four inverters per unit.

BACKGROUND

The inverters are the preferred source of power for the AC vital buses because of the stability and reliability they achieve in being powered from the 120 V DC battery source. (A) The function of the inverter is to convert DC electrical power to AC electrical power, thus providing an uninterruptible power source for the instrumentation and controls for the Reactor Protection System (RPS) and the Engineered Safety Feature Actuation System (ESFAS). Specific details on inverters and their operating characteristics are found in ~~[FSAR Chapter]~~. (4) the Watts Bar FSAR Chapter 8, (Ref. 1).

APPLICABLE SAFETY ANALYSES △

The initial conditions of ~~design basis~~ Accident (DBA) and (Ref. 2) transient ~~and accident~~ analyses in ~~[~~the FSAR, Chapter 6, ~~"Engineered Safety Features,"~~ and Chapter 15, ~~"Accident Analyses"]~~, assume ESF systems are OPERABLE. The DC to AC inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and ESFAS instrumentation and controls 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~~."~~

engineered safety features

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and is based on meeting the design basis of the unit. This includes maintaining required AC Vital buses OPERABLE during accident conditions in the event of:

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

b. A worst-case single failure.

Inverters are a part of the Distribution SystemS ~~– Operating~~ and as such satisfy Criterion 3 of the NRC I~~nterim~~ Policy Statement. ~~as described in References 1 and 2.~~ △

(continued)

BASES (continued)

LCO

The inverters ensure the availability of AC electrical power for the systems instrumentation required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA.

Maintaining the required inverters OPERABLE ensures that the redundancy incorporated into the design of the RPS and ESFAS instrumentation and controls is maintained. The ~~four~~ eight battery-powered inverters ~~[(two per train)]~~ ensure an uninterruptible supply of AC electrical power to the AC vital buses even if the ~~4.16 kV safety buses~~ 6.9 kV shutdown boards are de-energized. (A) (B)

(3) Via its associated 480 V shutdown board and vital battery board

OPERABLE inverters require the associated AC vital bus to be powered by the inverter which has the correct DC voltage (~~[120]~~ 125 V) applied from a battery to the inverter input, and inverter output AC voltage and frequency within tolerances.

(1)
~~This LCO is modified by a Note which allows [one/two] inverters to be disconnected from a [common] battery for ≤ 24 hours, if the vital bus(es) is/are powered from a Class 1E constant voltage transformer during the period and all other inverters are operable. This allows an equalizing charge to be placed on one battery. If the inverters were not disconnected, the resulting voltage condition might damage the inverter(s). These provisions minimize the loss of equipment that would occur in the event of a loss of offsite power. The 24-hour time period for the allowance minimizes the time during which a loss of offsite power could result in the loss of equipment energized from the affected AC vital bus while taking into consideration the time required to perform an equalizing charge on the battery bank. When utilizing the allowance, if one or more of the provisions is not met (e.g., 24-hour time period exceeded, etc.), LCO 3.0.3 must be entered immediately.~~

(continued)

BASES (continued)

APPLICABILITY

The inverters are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

Inverter requirements for MODES 5 and 6 are covered in the Bases for ~~Specification~~ LCO 3.8.8, "Inverters - Shutdown."

ACTIONS

B

A.1 ~~and A.2~~

associated 120V AC instrument power distribution panel.

With a required inverter inoperable, its associated AC vital bus becomes inoperable until it is ~~manually~~ re-energized from its ~~Class 1E constant voltage source transformer~~. Required Action A.1 allows ~~up~~ to 2 hours ~~to perform this task~~.

6

For this reason, a Note has been ~~added~~ included in Condition A requiring the entry into Conditions and Required Actions of LCO 3.8.9. This ensures that the vital bus is returned OPERABLE status within 2 hours.

The 2-hour Completion Time is consistent with the 2-hour Completion Time for an inoperable DC bus, and an inoperable AC vital bus (see ~~Specification~~ LCO 3.8.~~X~~ 9, "Distribution Systems – Operating"). Required Action A.~~2~~ allows 24 hours to fix the inoperable inverter and return it to service. The 24-hour limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the AC vital bus is powered from its ~~constant voltage source~~, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible, battery-backed, inverter source to the AC vital buses is the preferred source for powering instrumentation trip setpoint devices.

(continued)

BASES (continued)

△ B.1 and B.2

If the inoperable devices or components cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought

~~The unit must be placed in a~~ MODE in which the LCO does not apply. ~~If the inoperable devices or components cannot be restored to OPERABLE status within the associated Completion Time.~~ ~~This is done by placing the unit in~~ at least MODE 3 within 6 hours and ~~in~~ to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, ~~related to the amount of time required~~ to reach the required ~~MODES~~ unit conditions from full power conditions in an orderly manner and without challenging ~~unit~~ systems.

To achieve this status, the unit must be brought to

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital buses (including those from the associated vital battery boards and 480V shutdown boards,) energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation of the RPS and ESFAS connected to the AC vital buses. The 7-day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that will alert the operator to inverter malfunctions. 3

REFERENCES

4 1. Watts Bar FSAR, Section 8.3.1, "AC Power System." ~~Thomas E. Murley (NRC) letter to W. S. Wilgus dated May 9, 1989, forwarding the NRC Staff review of the "Nuclear Steam Supply System Vender Owners Groups' Application of the Commission's Interim Policy Statement Criteria to the Standard Technical~~ Specifications.

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

5 Watts Bar FSAR, Section 15, "Accident Analyses," and Section 6, "Engineered Safety Features."

(continued)

JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.8.7

Generic Justifications

A. Modified to reflect WBN plant specific design.

B. Modified to reflect WBN plant specific terminology.

Specific Justifications

1. Note indicating that inverters may be disconnected from associated DC busses to perform equalizing charging is not necessary since the battery boards associated with each charger are capable of maintaining 140 volts during equalizing charge periods. All loads can tolerate the 140 volt equalizing voltage. (See FSAR section 8.3.2.1.1).

2. The word "required" has been deleted from Condition A since all inverters are required (i.e. there are no spare inverters).

3. This modification was made by TVA to address NRC concerns about interconnection of redundant power divisions (TVA 6/21/85 letter to NRC).

4. Modified to provide a reference for WBN specific design info.

5. Modified to provide a reference for WBN specific Safety Analysis info.

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

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Inverters – Shutdown

LCO 3.8.8 Inverters shall be OPERABLE to support the onsite Class 1E AC vital bus electrical power distribution subsystem(s) required by LCO 3.8.10, Δ "Distribution Systems – Shutdown."

APPLICABILITY: Δ ~~3A~~ MODES 5 and 6.
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required inverters inoperable.	A.1~~.1~~ Declare affected required feature(s) inoperable. ~~and enter applicable Conditions and Required Actions of the LCOs for the inoperable required feature(s).~~ (4)	Immediately
	~~AND~~	
	Δ ~~A.1.2 Initiate action to restore required inverters to OPERABLE status.~~	~~Immediately~~
	Δ OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
	<u>AND</u>	
	A.2.4 Initiate action to ~~S~~suspend operations involving positive reactivity additions which would exceed limits specified in LCO 3.1.2, "Shutdown Margin," or LCO 3.9.1, "Boron Concentration."	Immediately
	<u>AND</u>	
	A.2.5 Initiate action to restore required inverters to OPERABLE status.	Immediately

4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.8.1	Verify correct inverter voltage, frequency, and alignments to required AC vital buses and from associated vital battery board and 480V shutdown board.	7 days

1 and from associated vital battery board and 480V shutdown board.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Inverters – Shutdown

BASES

engineered safety features

BACKGROUND

△ A description of the inverters is provided in the Bases for ~~Specifie tion~~ LCO 3.8.7, "Inverters – Operating."

APPLICABLE SAFETY ANALYSES

△ The initial conditions of design ~~basis~~ Accident (DBA) and transient ~~and accident~~ analyses in the FSAR, Chapter 6 (Ref. 1), ~~"Engineered Safety Features,"~~ and Chapter 15, ~~"Accident Analyses,"~~ assume ESF systems are OPERABLE. The DC to AC inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and ESFAS instrumentation and controls so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded.

Reactor Protection System

Engineered Safety Features Actuation System

The ~~Operability~~ OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum inverters to each AC vital bus during MODES 5 and 6 ensures that ~~(Ref. 1)~~:

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is available to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident.

△ The inverters were previously ~~considered~~ identified as part of the Distribution system and as such, satisfy Criterion 3 of the NRC ~~Interim~~ Policy Statement. ~~(Ref. 2 and 3).~~

LCO

The inverters ensure the availability of electrical power for the instrumentation for systems required to shutdown the

(continued)

BASES (continued)

(2) As a minimum, either the channel I and III or II and IV inverters shall be OPERABLE to support the distribution systems required by LCO 3.8.10, "Distribution Systems – Shutdown."

reactor and maintain it in a safe condition after an anticipated operational occurrence or postulated DBA. The battery-powered inverters provide uninterruptible supply of AC electrical power to the AC vital buses even if the ~~4.16 kV safety buses~~ (B) 6.9 kV Shutdown Boards are deenergized. OPERABILITY of the inverters requires that the AC vital bus be powered by the inverter (1) via its associated 480V shutdown board and vital battery board. This ensures the availability of sufficient inverter power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel-handling accidents ~~,~~ and inadvertent reactor vessel draindown).

APPLICABILITY

The inverters required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

b. Systems needed to mitigate a fuel-handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

Δ Inverter requirements for MODES 1, 2, 3, and 4 are covered in ~~Specification~~ LCO 3.8.7. ~~"Inverters – Operating."~~

ACTIONS

Δ A.1~~.1~~, A.~~1.2~~, A.2.1, A.2.2, A.2.3, ~~A.2.4~~ and A.2.5

If two trains are required per LCO 3.8.10, ~~"Distribution System – Shutdown,"~~ the remaining OPERABLE ~~I~~inverters may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, operations with a potential for draining the reactor vessel ~~and~~/or

(continued)

BASES (continued)

Notwithstanding performance of the above conservative Required Actions, the unit is still without sufficient inverters to operate in a safe manner. therefore, action must be initiated to restore the minimum required inverters and continue until the LCO requirements are restored.

(4)

operations with a potential for positive reactivity additions. The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory provided the required shutdown margin is maintained. the allowance of By ~~allowing~~ the option to declare required features inoperable with associated Inverter(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCOs' Required Actions. In many instances this option may ~~would likely~~ involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement ~~handling~~ of irradiated fuel assemblies, activities that could potentially result in inadvertent draining of the reactor vessel, and operations involving positive reactivity additions).

INSERT I-5 (4)

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions ~~will~~ minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required inverters and to continue this action until restoration is accomplished in order to provide the necessary inverter power to the unit safety systems.

(B)

The Completion Time of Immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required inverters should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power or powered from ~~a constant voltage source transformer~~ its associated 120V AC instrument power distribution panel.

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed, including those from the associated vital battery boards and 480 V shutdown boards, (1) and AC vital buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation of the RPS ~~Reactor Protection System~~ and ESFAS ~~Engineered Safety Feature Actuation System~~ connected to the AC vital buses. The 7-day Frequency takes into account the redundant capability of the inverters and other indications

(continued)

INSERT

A.2.4 ~~and B.4~~ I-05

Required Actions A precludes operations involving systems that contain large volumes of water (i.e. CVCS, SIS, and RWST) at reduced boron concentrations with respect to the RCS that could dilute the boron concentration of the RCS to less than that required to maintain the SDM requirements of LCO 3.1.2 in MODE 5 or less than the boron concentration specified in the COLR in MODE 6. This Required Action does not preclude positive reactivity additions that cannot reduce the SDM or boron concentration to less than the limits specified in LCO 3.1.2 or the COLR. The addition of water with a boron concentration greater than that required to maintain the reactor shutdown within the requirements of LCO 3.1.2 or the COLR, but less than the RCS, is permitted. Positive reactivity additions such as small volume chemical additions and normal plant cooldowns are also permitted as long as the SDM limits are met.

BASES (continued)

available in the control room that will alert the operator to inverter malfunctions.

REFERENCES

③ 1. Watts Bar FSAR, Section 15, "Accident Analysis" and Section 6 "Engineered Safety Features".

△ ~~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."~~

△ ~~3. WCAP-11618, "MERITS Program - Phase II, Task 5, Criteria Application", dated November 1987, including Addendum 1, dated April 1989.~~

(continued)

JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.8.8

Generic Justifications

A. Modified to reflect WBN plant specific design.

B. Modified to reflect WBN plant specific terminology.

Specific Justifications

1. This modification was made by TVA to address NRC concerns about interconnection of redundant power divisions (TVA 6/21/85 letter to NRC).

2. Wording has been added to clarify the fact that channels I & III and II & IV are associated with power trains A and B respectively and that a minimum of one of these power trains is required to be OPERABLE to meet requirements during shutdown.

3. Modified to provide a reference for WBN specific Safety Analysis info.

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

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems – Operating

LCO 3.8.9 Train A and Train B AC, AC vital bus, and DC electrical power distribution subsystems shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ~~or more~~ ~~[required]~~ 2 ~~AC buses, load centers, motor control centers, or distribution panels, in one~~ AC power distribution subsystem inoperable. 5 1 electrical	A.1 Restore ~~required~~ AC electrical power distribution subsystem to OPERABLE status.	8 hours AND 16 hours from discovery of failure to meet ~~the~~ LCO
B. One AC vital bus inoperable.	B.1 Restore ~~[required]~~ AC vital bus subsystem to OPERABLE status.	2 hours AND 16 hours from discovery of failure to meet the LCO
C. One ~~[required]~~ 2 DC electrical power distribution subsystem inoperable.	C.1 Restore ~~[required]~~ DC electrical power distribution subsystem to OPERABLE status.	2 hours AND 16 hours from discovery of failure to meet ~~the~~ LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1	Verify correct breaker alignments and voltage to required AC, DC, and AC vital bus electrical power distribution subsystems.	7 days

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.9 Distribution Systems – Operating

BASES

BACKGROUND

~~{VS-BW,CE,W,BWR/4:}~~ The onsite Class 1E AC, DC, and AC Vital bus electrical power distribution ~~system~~s are divided by trains into ~~{two}~~ redundant and independent AC, DC, and AC Vital bus electrical power distribution subsystems.

~~the alternate~~

The AC electrical power subsystem 6.9 kV shutdown board includes 4 ~~for each train consists of a primary ESF 4.16 kV bus and secondary [480 and 120] V buses, distribution panels, motor control centers and load centers.~~ Each ~~[4.16 kV ESF bus]~~ 6.9 kV Shutdown board has access to the two ~~at least [one~~ separate and independent preferred offsite sources of power~~]~~ as well as a dedicated onsite diesel generator (DG) source. Each ~~[4.16 kV ESF bus]~~ 6.9 kV shutdown board is normally connected to a preferred offsite source. ~~After a~~ During loss of ~~the preferred offsite~~ this normal power source ~~to a 4.16 kV ESF bus, a transfer~~ 6.9 kV transfer scheme to the alternate ~~offsite~~ source is accomplished by utilizing a time-delayed bus undervoltage relay. If ~~all~~ offsite source~~s~~ ~~are~~ is unavailable, the onsite emergency ~~DG~~ power system will supply power to the ~~4.16 kV ESF bus~~ 6.9 kV shutdown board. Control power for the ~~4.16~~ 6.9 kV breakers is supplied from the Class 1E batteries. Additional description of this system may be found in the Bases for ~~Specification~~ LCO 3.8.1, "AC Sources – Operating," and the Bases for ~~Specification~~ LCO 3.8.4, "DC Sources – Operating."

The ~~secondary~~ AC ~~electrical power~~ distribution ~~system for each train includes the safety related~~ includes the 480V shutdown boards and associated supply transformers, load centers, ~~motor control centers and distribution panels~~ and protective devices shown in Table B 3.8.9-1 and DC boards I, II, III, and IV.

The 120 ~~Vac~~ vital buses are arranged in ~~two~~ four load groups ~~per train~~ and are normally powered from the inverters. The alternate power supply for the vital buses ~~are Class 1E constant voltage source transformers~~ is a 120V AC instrument power distribution panel powered from the same train as the associated inverter, and its use is governed by LCO 3.8.7, "Inverters – Operating." STET Each ~~constant voltage source~~ transformer is powered from ~~a Class 1E AC bus.~~ the 480 V Shutdown boards 1A1-A or 1B1-B and 1B2-B or 1A2-A.

There are ~~two~~ four independent 125~~/250~~ Vdc ~~electrical power distribution subsystems (one for each train).~~ trains (labelled I, II, III, and IV) associated electrical power distribution subsystems. Each train receives normal power from an independent 480V AC shutdown board via its associated battery charger. Upon loss of ~~480V AC~~ shutdown board power, the DC buses are energized by their connected battery banks.

The list of all required distribution buses is located in Table B 3.8.9-1.

(continued)

BASES (continued)

engineered safety features

APPLICABLE SAFETY ANALYSES △

The initial conditions of (Ref. 2) design basis ~~transient and accident~~ Accident (DBA) and analyses in FSAR Chapter 6, ~~"Engineering Safety Features,"~~ and Chapter 15, ~~"Accident Analyses,"~~ assume ESF systems are OPERABLE. The AC, DC, and AC Vital Bus 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."

The OPERABILITY of the AC, DC, and AC Vital Bus electrical power distribution system is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining power 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.

Distribution Systems ~~Operating~~ satisfy Criterion 3 of the NRC ~~Interim~~ Policy Statement ~~as described in References 2 and 3.~~.

LCO △

The required power distribution subsystems listed in Table B 3.8.~~X~~9-1 ensure the availability of AC, DC, and AC Vital Bus electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. The AC, DC, and AC Vital Bus electrical power distribution subsystems are required to be OPERABLE.

Maintaining the Train A and B AC, DC, and AC Vital Bus electrical power distribution subsystems OPERABLE ensures that the redundancy incorporated into the design of ESF is not defeated. Therefore, a single failure within any system

(continued)

BASES (continued)

or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor.

OPERABLE AC, DC, and AC ~~Vital Bus~~ electrical power distribution subsystems, require the associated buses, load centers, motor control centers, and distribution panels to be energized to their proper voltages.

In addition, tie breakers between redundant safety related AC, DC, and AC Vital Bus power distribution subsystems, if they exist, must be open. This will prevent any electrical malfunction in any power distribution subsystem from propagating to the redundant subsystem which could cause the failure of redundant subsystem and possibly cause a loss of essential safety function(s). If any tie breakers are closed, the affected redundant electrical power distribution subsystems are considered inoperable. This applies to the onsite, safety related redundant electrical power distribution subsystems. It does not, however, preclude redundant ~~Class 1E 4.16 kV buses~~ 6.9 kV shutdown boards from being powered from the same offsite circuit.

APPLICABILITY

The electrical power distribution subsystems are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

△ Electrical power distribution subsystem requirements for MODES 5 and 6 are covered in the Bases for ~~Specification~~ LCO 3.8.10, "Distribution Systems - Shutdown."

ACTIONS

A.1 5

With one ~~or more required~~ AC electrical power distribution subsystems ~~AC buses, load centers, motor control centers, or distribution panels, except AC vital buses, in one train~~ inoperable, the remaining AC electrical

(continued)

BASES (continued)

B

power distribution subsystem in the other train is capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the required AC buses, and Shutdown boards ~~load centers, motor control centers, and distribution panels~~ must be restored to OPERABLE status within 8 hours.

Condition A worst scenario is one train ~~division~~ without AC power (i.e., no offsite power to the ~~division~~ and the associated DG inoperable). In this condition, the unit is more vulnerable to a complete loss of AC power. It is, therefore, imperative that the unit operators' attention be focused on minimizing the potential for loss of power to the remaining train ~~division~~ by stabilizing the unit, and on restoring power to the affected train ~~division~~. The 8 hour time limit before requiring a unit shutdown in this condition is acceptable because:

1. There is a potential for decreased safety if the unit operators' attention is diverted from the evaluations/actions necessary to restore power to the affected train ~~division~~, to the actions associated with taking the unit to shutdown within this time limit.

2. The potential for an event in conjunction with a single failure of a redundant component in the train ~~division~~ with AC power (the redundant component is verified OPERABLE in accordance with Specification 5.8, Safety Function Determination Program).

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, a DC bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met for up to 2 hours. This could lead to a total of 10 hours, since initial failure of the LCO, to restore the AC distribution system. At this time a DC circuit could again become inoperable, and AC distribution restored OPERABLE. This could continue indefinitely.

(continued)

BASES (continued)

△ The Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock". This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition A was entered. The 16-hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

B.1

With one AC vital bus inoperable, the remaining OPERABLE AC vital buses are capable of supporting the minimum safety functions necessary to shut down the unit and maintain it in the safe shutdown condition. Overall reliability is reduced, however, since an additional single failure could result in the minimum required ESF functions not being supported. Therefore, the required AC vital bus must be restored to OPERABLE status within 2 hours.

Condition B represents one AC vital Bus without power; potentially both the DC source and the associated AC source non-functioning. In this situation the ~~plant~~ unit is significantly more vulnerable to a complete loss of all non-interruptible power. It is therefore, imperative that the operator's attention focus on stabilizing the ~~plant~~ unit, minimizing the potential for loss of power to the remaining Vital Buses and restoring power to the affected Vital Bus.

This 2-hour limit is the more conservative than Completion Times allowed for the vast majority of components which would be without adequate vital AC power. Taking exception to LCO 3.0.2 for components without adequate vital AC power, which would have Required Action Completion Times shorter than 2 hours if declared inoperable, is acceptable in consideration of competing concerns:

a. ~~the~~ potential for decreased safety by requiring a change in ~~plant~~ unit conditions (i.e., requiring a shutdown) and not allowing stable operations to continue;

b. ~~the~~ potential for decreased safety by requiring entry into numerous Applicable Conditions and Required Actions for components without adequate vital AC power and not providing sufficient time for the operators to

(continued)

BASES (continued)

△ perform the necessary evaluations and actions to restore power to the affected ~~division;~~ train

c. ~~the~~ The potential for an event in conjunction with a single failure of a redundant component. This redundant component is verified OPERABLE in accordance with Specification 5.8. ~~SFOP~~.

The 2-hour Completion Time takes into account the importance to safety of restoring the AC vital bus to OPERABLE status, the redundant capability afforded by the other OPERABLE vital buses, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action B.1 establishes a limit on the maximum allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an AC bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met for up to 8 hours. This could lead to a total of 10 hours, since initial failure of the LCO, to restore the Vital Bus distribution system. At this time an AC ~~division~~ train could again become inoperable, and Vital Bus distribution restored OPERABLE. This could continue indefinitely.

This Completion time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock". This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition B was entered. The 16-hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

C.1

With one DC bus~~(es)~~ ~~in one train~~ inoperable the remaining DC electrical power distribution subsystems is capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution subsystem could result in the minimum required ESF functions not being

(continued)

BASES (continued)

supported. Therefore, the required DC buses must be restored to OPERABLE status within 2 hours.

Condition C represents one ~~division~~ train without adequate DC power; potentially both the battery significantly degraded and the associated charger non-functioning. In this situation the ~~plant~~ is significantly more vulnerable to a complete loss of all DC power. It is, therefore, imperative that the operator's attention focus of stabilizing the ~~plant~~, minimizing the potential for loss of power to the remaining ~~divisions~~ trains and restoring power to the affected ~~division~~ train.

This 2-hour limit is more conservative than Completion Times allowed for the vast majority of components which would be without power. Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Required Action Completion Times shorter than 2 hours, is acceptable in consideration of competing concerns:

a. ~~the~~ potential for decreased safety by requiring a change in ~~plant~~ unit conditions (i.e., requiring a shutdown) and not allowing stable operations to continue;

b. ~~the~~ potential for decreased safety by requiring entry into numerous Applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations~~,~~ and actions to restore power to the affected ~~division~~ train;

c. ~~the~~ potential for an event in conjunction with a single failure of a redundant component. ~~(~~This redundant component is assured OPERABLE in accordance with Specification 5.8. ~~— SFDP).~~

The 2-hour Completion Time for DC buses is consistent with Regulatory Guide 1.93, (Ref. 1).

The second Completion Time for Required Action C.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition C is entered while, for instance, an AC bus is inoperable and subsequently

(continued)

BASES (continued)

returned OPERABLE, the LCO may already have been not met for up to 8 hours. This could lead to a total of 10 hours, since initial failure of the LCO, to restore the DC distribution system. At this time an AC ~~division~~ train could again become inoperable, and DC distribution restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock". This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition C was entered. The 16-hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

D.1 and D.2

If the inoperable Distribution System cannot be restored to OPERABLE status within the required Completion Time

To achieve this status, the unit must be brought to

The unit must be brought to ~~placed in~~ a MODE in which the LCO does not apply ~~if the inoperable distribution subsystem cannot be restored to OPERABLE status within the associated Completion Time. This is done by placing the unit in~~ to at least MODE 3 within 6 hours and ~~in~~ to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience related, ~~to the amount of time required~~ to reach the required ~~MODES~~ unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the AC, AC Vital bus, and DC electrical power distribution systems are functioning properly, with all the required circuit breakers closed and the buses energized from normal power. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. The 7-day Frequency takes into account the redundant capability of the AC, AC Vital bus, and DC electrical power distribution subsystems, and other indications available in the control room that will alert the operator to subsystem malfunctions.

(continued)

BASES (continued)

REFERENCES

1. Regulatory Guide 1.93, "Availability of Electric Power Sources," U.S. Nuclear Regulatory Commission, December 1974.

2. Thomas E. Murley (NRC) letter to W. S. Wilgus dated May 9, 1989, forwarding the NRC Staff review of the "Nuclear Steam Supply System Vender Owners Groups' Application of the Commission's Interim Policy Statement Criteria to the Standard Technical Specifications.

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

4 2. Watts Bar FSAR, Chapter 6 "Engineered Safety Features, Chapter 8, "Electric Power," and Chapter 15, "Accident Analysis"

BASES (continued)

Table B 3.8.9-1 (page 1 of 1)
AC and DC Electrical Power Distribution System

TYPE	VOLTAGE	Train A*	Train B*
AC safety buses	[4160 V]	[ESF Bus] [NB01]	[ESF Bus] [NB02]
	[480 V]	Load Centers [NG01, NG03]	Load Centers [NG02, NG04]
	[480 V]	Motor Control Centers [NG01A, NG01I, NG01B, NG03C, NG03I, NG03D]	Motor Control Centers [NG02A, NG02I, NG02B, NG04C, NG04I, NG04D]
	[120 V]	Distribution Panels [NP01, NP03]	Distribution Panels [NP02, NP04]
DC buses	[125 V]	Bus [NK01] from battery [NK11] and charger [NK21] Bus [NK03] from battery [NK13] and charger [NK23]	Bus [NK02] from battery [NK12] and charger [NK22] Bus [NK04] from battery [NK14] and charger [NK24]
		Distribution Panels [NK41, NK43, NK51]	Distribution Panels [NK42, NK44, NK52]
AC vital buses	[120 V]	Bus [NN01] from inverter [NN11] connected to bus [NK01] Bus [NN03] from inverter [NN13] connected to bus [NK03]	Bus [NN02] from inverter [NN12] connected to bus [NK02] Bus [NN04] from inverter [NN14] connected to bus [NK04]

A
see next sheet for WBN specific table.

* Each train of the AC and DC electrical power distribution system is a subsystem.

(continued)

Table B 3.8.9-1 (page 1 of 1)

AC and DC Electrical Power Distribution System

TYPE	VOLTAGE	Train A *	Train B *
AC safety buses	6900 V 480 V	Shdn Bd 1A-A 2A-A Shdn Bd 1A1-A, 1A2-A 2A1-A, 2A2-A	Shdn Bd 1B-B 2B-B Shdn Bd 1B1-B, 1B2-B 2B1-B, 2B2-B
DC buses	125 V	Board I from Vital Battery Bank I** Board III from Vital Battery Bank III**	Board II from Vital Battery Bank II** Board IV from Vital Battery Bank IV**
AC vital buses	120 V	Vital channel 1-I Vital channel 2-I from inverter and DC Board I Vital channel 1-III Vital channel 2-III from inverter and DC Board III	Vital channel 1-II Vital channel 2-II from inverter and DC Board II Vital channel 1-IV Vital channel 2-IV from inverter and DC Board IV

* Each train of the AC and DC electrical power distribution system is a subsystem.

** Vital Battery Bank V may be substituted for any of the required Vital Battery Banks.

JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.8.9

Generic Justifications

A. Modified to reflect WBN plant specific design.

B. Modified to reflect WBN plant specific terminology.

Specific Justifications

1. Condition A is modified to remove the partial (or example) list of AC Power distribution system elements provided in the RSTS standard. This modification makes Condition A consistent with the LCO and Required Action A.1. The BASES Table 3.8.9-1 provides a more appropriate list of elements to be included in an AC Electrical Power Distribution Subsystem.

2. The word "required" has been deleted from Condition A and Condition C since all AC and DC Electrical Power Distribution Subsystems are required to be OPERABLE.

3. Wording in the RSTS standard that attempts to list the elements of a train of an electric power subsystem has been removed from this paragraph (i.e. second paragraph). The paragraph that follows (i.e. third paragraph) and Table B.3.8.9-1 include information on what is included in an AC Electrical Power Distribution system.

4. Modified to provide a reference for WBN specific Safety Analysis info.

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

3.8 ELECTRICAL POWER SYSTEMS

3.8.10 Distribution Systems – Shutdown

LCO 3.8.10 The necessary portion of AC, DC, and AC vital bus electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

APPLICABILITY: ~~In~~ MODES 5 and 6.
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC, DC, or AC vital bus electrical power distribution subsystems inoperable.	A.1~~.1~~ Declare associated supported required feature(s) inoperable. ~~and enter applicable~~ Conditions and Required Actions of the LCOs for the inoperable required feature(s).	Immediately
	AND	
	A.1.2 Initiate actions to restore required AC, DC, and AC vital bus electrical power distribution subsystems to OPERABLE status.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2.2	Suspend movement of irradiated fuel assemblies.	Immediately
		AND	
	A.2.3	Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
		AND	
	A.2.4	△ Initiate action to ~~S~~suspend operations involving positive reactivity additions.	Immediately
		AND	
	A.2.5	Initiate actions to restore required AC, DC, and AC vital bus electrical power distribution subsystems to OPERABLE status.	Immediately
△ **AND**			
	A.2.6	Declare associated required features inoperable and enter applicable Conditions and Required Actions ~~of the LCOs for the inoperable required features.~~	Immediately
		3 RHR subsystem(s) inoperable and not in operation. △	

~~for LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled," LCO 3.9.5, "RHR and Coolant Circulation - High Water Level," and LCO 3.9.6, "RHR and Coolant Circulation - Low Water Level."~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.10.1	Verify correct breaker alignments and voltage to required AC, DC, and AC vital bus electrical power distribution subsystems.	7 days

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.10 Distribution System – Shutdown

BASES

Engineered safety features

BACKGROUND △

A description of the AC, DC and AC vital bus electrical power distribution systems is provided in the Bases for ~~Specification~~ LCO 3.8.~~10~~ 9, "Distribution System s – Operating."

APPLICABLE SAFETY ANALYSES △

(Ref. 1) Accident (DBA) and

The initial conditions of design basis transient ~~and accident~~ analyses in FSAR Chapter 6, ~~"Engineered Safety Features,"~~ and Chapter 15, ~~"Accident Analyses,"~~ assumes (ESF) systems are OPERABLE. The AC, DC and AC vital bus 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.

The OPERABILITY of the AC, DC, and AC vital bus electrical power distribution system is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum AC, DC and AC vital bus electrical power distribution subsystems during MODES 5 and 6, ensures that ~~(Ref. 1)~~:

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel-handling accident.

The AC and DC electrical power distribution system satisfies Criterion 3 of the NRC ~~Interim~~ Policy Statement. ~~(Ref. 2 and 3).~~

(continued)

BASES (continued)

LCO

△ Various combinations of subsystems, equipment ed, and components will be required OPERABLE by other Technical Specification (TS) LCOs, depending on the specific ~~plant~~ unit condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. The Distribution System-Shutdown LCO explicitly requires energization of the portions of the electrical distribution system necessary to support OPERABILITY of ~~Technical Specification~~ TS required systems, equipment and components - all ~~both~~ specifically addressed ~~by their own~~ in each LCO, and implicitly required via the definition of OPERABILITY.

△ Maintaining these portions of the distribution system energized will ensure the availability of sufficient power to operate the unit in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel-handling accidents ~~,~~ and inadvertent reactor vessel draindown).

APPLICABILITY

The AC and DC electrical power distribution subsystems required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

b. Systems needed to mitigate a fuel-handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

△ the AC, DC, ~~and~~ AC ~~vital bus~~ electrical power distribution subsystem requirements for MODES 1, 2, 3 and 4 are

(continued)

and to declare associated required features inoperable and enter applicable Conditions and Required Actions for LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled," LCO 3.9.5, "RHR and Coolant Circulation - High Water Level," and LCO 3.9.6, "RHR and Coolant Circulation - Low Water Level."

BASES (continued)

covered in LCO ~~Specification~~ 3.8.9, "Distribution System — Operating."

ACTIONS

A.1, ~~A.1.2~~, A.2.1, A.2.2, A.2.3, A.2.4, A.2.5 and A.2.6

Although redundant required features may require redundant trains of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem train may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, and~~/or~~ operations with a potential for draining the reactor vessel. By allowing the option to declare required features associated with an ~~INOPERABLE~~ distribution subsystem ~~INOPERABLE~~, appropriate restrictions will be implemented in accordance with the affected distribution subsystems LCO's Required Actions. In many instances this would likely involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made~~.~~ i.e., to suspend CORE ALTERATIONS, handling of irradiated fuel assemblies, any activities that could potentially result in inadvertent draining of the reactor vessel, ~~and~~ operations involving positive reactivity additions,

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions will minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC and DC electrical power distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the ~~unit's~~ safety systems.

1

Notwithstanding performance of the above conservative Required Actions, the unit is still without sufficient AC and DC electrical power distribution subsystems to operate in a safe manner. Therefore, action must be initiated to restore the minimum required AC and DC electrical power distribution subsystems and continue until the LCO requirements are restored.

The Completion Time of Immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required distribution subsystems should be completed as quickly as possible in order to minimize the time the ~~unit's~~ safety systems may be without power.

SURVEILLANCE REQUIREMENTS

SR 3.8.10.1

This Surveillance verifies that the AC, DC, and AC vital bus electrical power distribution subsystems are functioning

(continued)

BASES (continued)

properly, with the buses energized. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. The 7-day Frequency takes into account the capability of the electrical power distribution subsystems, as well as other indications available in the control room that will alert the operator to subsystem malfunctions.

REFERENCES

1. ~~[Unit name] FSAR, Section [ ], "[Title]."~~

△ ~~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."~~

△ ~~3. WCAP 11618, "MERITS Program - Phase II, Task 5, Criteria Application", dated November 1987, including Addendum 1, dated April 1989.~~

1 Watts Bar FSAR, Section 15, "Accident Analysis," and Section 6, "Engineered Safety Features."

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources-Shutdown

BASES

BACKGROUND

A description of the AC sources is provided in the Bases for LCO 3.8.1, "AC Sources—Operating."

APPLICABLE SAFETY ANALYSIS

The OPERABILITY of the minimum AC Sources during MODES 5 and 6, ensures that:
and during movement of irradiated fuel assemblies

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel-handling accident.

In general, when the unit is shut down, the Technical Specification requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or loss of all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs), which are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6. Worst-case bounding events are deemed not-credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and minimal in consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

During MODES 1, 2, 3, and 4 various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in recognition that certain testing and maintenance activities

(continued)

JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.8.10

Generic Justifications

A. Modified to reflect WBN plant specific design.

B. Modified to reflect WBN plant specific terminology.

Specific Justifications

1. This paragraph describing requirements to initiate action to restore the minimum required systems to OPERABLE status has been deleted because it contains essentially the same information as the paragraph being added as an outcome of the MEREX meeting.

2. Modified to provide a reference for WBN specific Safety Analysis info.

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

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Boron concentration not within limit.	A.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	A.2	Suspend positive reactivity additions.	Immediately
	AND		
	A.3	Initiate actions to restore boron concentration to within limits.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.1.1 Verify boron concentration is within limit A as specified in COLR.	72 hours

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS), the refueling canal, and the refueling cavity during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $K_{eff} \leq 0.95$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant procedures.

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity are then flooded with borated water from the refueling water storage tank through the open reactor vessel by gravity feeding or by the use of the Residual Heat Removal (RHR) System pumps.

1 During refueling, the water volumes in the RCS, the refueling canal, and the refueling cavity are contiguous.

(continued)

BASES

BACKGROUND (continued)

(1) ~~However, the soluble boron concentration is not necessarily the same in each volume. If additions of boron are required during refueling, the CVCS makes it available through the RCS.~~

The pumping action of the RHR System in the RCS, and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added concentrated boric acid with the water in the refueling canal. The RHR System is in operation during refueling (see LCO 3.9.5 and LCO 3.9.6) to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS, the refueling canal, and the refueling cavity above the COLR limit.

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis and is conservative for MODE 6. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the plant refueling procedures that verify the correct fuel-loading plan (including full-core mapping) ensure that the K_{eff} of the core will remain ≤ 0.95 during the refueling operation. Hence, at least a 5% Δk/k margin of safety is established during refueling.

During refueling, the water volume in the spent fuel pool, the transfer canal, the refueling canal, the refueling cavity and the reactor vessel form a single mass. As a result, the soluble boron concentration is relatively the same in each of these volumes ~~(Ref. 2)~~. (A)

The limiting boron dilution accident analyzed occurs in MODE 5 (Ref. ~~3~~ 2). A detailed discussion of this event is provided in Bases B 3.1.2, "SHUTDOWN MARGIN – T_{avg} ≤ 200°F."

The RCS boron concentration satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

The LCO requires that a minimum boron concentration be maintained in the RCS, the refueling canal, and the refueling cavity while in MODE 6. The boron concentration limit specified in the COLR ensures that a core K_{eff} of ≤ 0.95 is maintained during fuel-handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a $K_{eff} \leq 0.95$. Above MODE 6, LCO 3.1.1 and LCO 3.1.2, "SHUTDOWN MARGIN," ensure that an adequate amount of negative reactivity is available to shut down the reactor and maintain it subcritical.

ACTIONS

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the plant in compliance with the LCO. If the boron concentration of any coolant volume in the RCS, the refueling canal, or the refueling cavity is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position.

A.3

In addition to immediately suspending CORE ALTERATIONS or positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, no unique design basis event must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In

(continued)

BASES

ACTIONS

A.3 (continued)

order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for plant conditions.

3

actions have been they

Once ~~boration is~~ initiated, ~~it~~ must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures that the coolant boron concentration in the RCS, the refueling canal, and the refueling cavity is within the COLR limits. The boron concentration of the coolant in each volume is determined periodically by chemical analysis.

A minimum Frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.

REFERENCES 2

1. 10 CFR 50, Appendix A, GDC 26, "Reactivity Control System Redundancy and Capability."

A 2. NS-57.2, ~~ANSI/ANS-57.2-1983~~, Section 6.4.2.2.3, 1983.

2 3. Watts Bar FSAR, Section ~~[~~15~~]~~, "Accident Analysis."

JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.9.1

A. This change reflects comments made by the industry to the NRC at the RSTS P&R review meeting held in Irvine, CA July 13 - 20, 1992.

1. This paragraph deleted since it is an error in the standard and conflicts with the third paragraph in "Applicable Safety Analysis". Westinghouse has acknowledged the error and notified NRC.

2. Watts Bar prefers to use the expanded references format.

3. A number of actions are acceptable to restore boron concentration including either batch or continuous boration, depending on plant conditions. The bases as currently worded imply "continuous boration", however, the Required Action does not require this. It only requires that "Actions" be initiated and continued.

3.9 REFUELING OPERATIONS

3.9.2 Unborated Water Source Isolation Valves

LCO 3.9.2 Each valve used to isolate unborated water sources shall be secured in the closed position.

APPLICABILITY: MODE 6.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each unborated water source isolation valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Required Action A.3 must be completed whenever Condition A is entered. --------------------- One or more valves not secured in closed position.	A.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2 Initiate actions to secure valve in closed position.	Immediately
	<u>AND</u>	
	A.3 Perform SR 3.9.1.1, boron concentration verification.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.2.1	Verify each valve that isolates unborated water sources is secured in the closed position.	31 days

B 3.9 REFUELING OPERATIONS

B 3.9.2 Unborated Water Source Isolation Valves

BASES

BACKGROUND	During MODE 6 operations, all isolation valves for reactor makeup water sources containing unborated water that are connected to the Reactor Coolant System (RCS) must be closed to prevent unplanned boron dilution of the reactor coolant. The isolation valves must be secured in the closed position. The Chemical and Volume Control System is capable of supplying borated and unborated water to the RCS through various flow paths. Since a positive reactivity addition made by reducing the boron concentration is inappropriate during MODE 6, isolation of all unborated water sources prevents an unplanned boron dilution.
APPLICABLE SAFETY ANALYSES	The possibility of an inadvertent boron dilution event (Ref. 1) occurring during MODE 6 refueling operations is precluded by adherence to this LCO, which requires that potential dilution sources be isolated. Closing the required valves during refueling operations prevents the flow of unborated water to the filled portion of the RCS. The valves are used to isolate unborated water sources. These valves have the potential to indirectly allow dilution of the RCS boron concentration in MODE 6. By isolating unborated water sources, a safety analysis for an uncontrolled boron dilution accident in accordance with the Standard Review Plan (Ref. 2) is not required for MODE 6. The RCS boron concentration satisfies Criterion 2 of the NRC Policy Statement.
LCO	This LCO requires that flow paths to the RCS from unborated water sources be isolated to prevent unplanned boron dilution during MODE 6 and, thus, avoid a reduction in SHUTDOWN MARGIN.

(continued)

BASES (continued)

APPLICABILITY

In MODE 6, this LCO is applicable to prevent an inadvertent boron dilution event by ensuring isolation of all sources of unborated water to the RCS.

For all other MODES, the boron dilution accident was analyzed and was found to be capable of being mitigated.

ACTIONS

The ACTIONS Table has been modified by a Note which allows separate Condition entry for each unborated water source isolation valve.

A.1

Continuation of CORE ALTERATIONS is contingent upon maintaining the unit in compliance with this LCO. With any valve used to isolate unborated water sources not secured in the closed position, all operations involving CORE ALTERATIONS must be suspended immediately. The Completion Time of "immediately" for performance of Required Action A.1, shall not preclude moving a component to a safe position.

Condition A has been modified by a Note to require that Required Action A.3 must be completed whenever Condition A is entered.

A.2

Preventing inadvertent dilution of the reactor coolant boron concentration is dependent on maintaining the unborated water isolation valves secured closed. Securing the valves in the closed position ensures that the valves cannot be inadvertently opened. The Completion Time of "immediately" requires an operator to initiate actions to close an open valve and secure the isolation valve in the closed position immediately. Once actions are initiated, they must be continued until the valves are secured in the closed position.

(continued)

JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.9.2

1. Watts Bar prefers to use the expanded reference section format.

3.9 REFUELING OPERATIONS

3.9.3 Nuclear Instrumentation

LCO 3.9.3 Two source range neutron flux monitors shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One source range neutron flux monitor inoperable.	A.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u> Insert I-1 7	
	A.2 Suspend positive reactivity additions.	Immediately
	<u>AND</u>	A
8	A.3 Initiate actions to restore source range neutron flux monitor to OPERABLE status.	7 days
B. Two source range neutron flux monitors inoperable.	B.1 Initiate actions to restore one source range neutron flux monitor to OPERABLE status.	Immediately
	<u>AND</u>	
	B.2 Perform SR 3.9.1.1, boron concentration verification.	4 hours <u>AND</u> Once per 12 hours thereafter

INSERT I-1

Suspend all operations involving positive reactivity additions that would reduce the boron concentration to less than the limit specified in the COLR.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.3.1 Perform CHANNEL CHECK.	12 hours
SR 3.9.3.2 Perform ~~ANALOG~~ CHANNEL ~~OPERATIONAL TEST~~ CALIBRATION. 6	~~7 days~~ 18 months

A,9

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

Neutron detectors are excluded from CHANNEL CALIBRATION.

B 3.9 REFUELING OPERATIONS

B 3.9.3 Nuclear Instrumentation

(1) Neutron Monitoring System (NMS)

BASES

BACKGROUND

(2) Changes

The source range neutron flux monitors are used during refueling operations to monitor the core reactivity ~~condition~~. The installed source range neutron flux monitors are part of the ~~Nuclear Instrumentation System (NIS)~~. These detectors are located external to the reactor vessel and detect neutrons leaking from the core.

(3) Fission chambers

primary

The installed source range neutron flux monitors are ~~BF3 detectors operating in the proportional region of the gas-filled detector characteristic curve.~~ The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux (1E+6 cps). ~~with a [5]% instrument accuracy.~~ The detectors also provide continuous visual indication in the control room and an audible alarm to alert operators to a possible dilution accident. The ~~NIS~~ is designed in accordance with the criteria presented in Reference 1.

(4)

NMS (1)

APPLICABLE SAFETY ANALYSES

Two OPERABLE source range neutron flux monitors are required to provide a signal to alert the operator to unexpected changes in core reactivity such as with a boron dilution accident (Ref. 2) or an improperly loaded fuel assembly. The need for a safety analysis for an uncontrolled boron dilution accident is eliminated by isolating all unborated water sources as required by LCO 3.9.2.

The source range neutron flux monitors satisfy Criterion 3 of the NRC Policy Statement.

LCO

This LCO requires that two source range neutron flux monitors must be OPERABLE to ensure that redundant monitorjng capability is available to detect changes in core reactivity.

(continued)

BASES (continued)

APPLICABILITY

In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity levels. In MODES 2, 3, 4, and 5, these same installed source range detectors and circuitry are also required to be OPERABLE by LCO 3.3.1, "Reactor Trip System Instrumentation."

ACTIONS

A.1 and A.2

A,7

that would reduce the boron concentration to less than the limit specified in the COLR

With only one source range neutron flux monitor OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Action A.1 shall not preclude moving a component to a safe position.

A,7

A.3

A,8

With only one source range neutron flux monitor OPERABLE, action shall be initiated to restore the inoperable monitor to OPERABLE status within 7 days. Seven days is a reasonable time in which corrective actions must be initiated considering the 72-hour boron sampling Frequency of SR 3.9.1.1, the suspension of CORE ALTERATIONS per Required Action A.1, and positive reactivity changes per Required Action A.2 above. Corrective actions, once started, must be continued until the monitor is restored to OPERABLE status.

B.1

With no source range neutron flux monitor OPERABLE, actions to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, actions shall be continued until a source range neutron flux monitor is restored to OPERABLE status.

Required Action A.2 does not preclude positive reactivity additions that cannot reduce the boron concentration to less than the limit specified in the COLR. The addition of water with a boron concentration greater than that required to maintain the reactor shutdown within the requirements of the COLR, but less than the RCS, is permitted. Positive reactivity additions, such as small volume chemical additions and normal cooldowns are also permitted provided the boron concentration in the COLR is maintained.

(continued)

BASES

ACTIONS (continued)

B.2

With no source range neutron flux monitor OPERABLE, there is no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be made, the core reactivity condition is stabilized until the source range neutron flux monitors are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration. The Frequency of once per 12 hours ensures that unplanned changes in boron concentration would be identified. The 12-hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

SR 3.9.3.1 is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions.

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.1, "Reactor Trip System Instrumentation."

SR 3.9.3.2

INSERT

A, 9

6

The performance of ~~an~~ ~~ANALOG~~ CHANNEL OPERATIONAL TEST ensures that the analog process control equipment and trip setpoints are within limits. The 7-day Frequency has been shown through operating experience to be conservative, considering operating history data for the setpoint drift, and is further justified because any malfunctions would be

(continued)

INSERT

SR 3.9.3.2 is the performance of a CHANNEL CALIBRATION every 18 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range neutron flux monitor consists of verifying equipment outputs to known electrical inputs ~~obtaining the detector plateau curves or preamp discriminator curves, evaluating these curves and comparing the curves with manufacturers data~~. The 18-month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown that these components usually pass the Surveillance when performed on the 18-month Frequency.

BASES

SURVEILLANCE REQUIREMENTS

A, 9

SR 3.9.3.2 (continued)

detected during the CHANNEL CHECK, which is performed every 12 hours.

REFERENCES

5

1. ~~10 CFR 50, Appendix A, GDC 13, GDC 26, GDC 28, and GDC 29.~~

2. ~~FSAR, Section [15.2.4].~~

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants:"

   GDC 13, "Instrumentation and Control,"

   GDC 26, "Reactivity Control System Redundancy and Capability,"

   GDC 28, "Reactivity Limits," and

   GDC 29, "Protection Against Anticipated Operational Occurrences."

2. Watts Bar FSAR (~~Unit 1~~), Section 15.2.4, "Uncontrolled Boron Dilution."

JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.9.3

A. This change reflects comments made by the industry to the NRC at the RSTS P&R review meeting held in Irvine, CA July 13 - 20, 1992.

1. Change to reflect plant specific terminology.

2. Editorial change. "Changes" to core reactivity are monitored by the source range monitors. "Condition" implies a reactivity meter.

3. Watts Bar has replaced the primary installed BF_3 detectors with fission chambers.

4. The accuracy of the source range detector is not relevant to the function it provides in this LCO. The key parameter is the change in count rates which is monitored to determine changes in core reactivity.

5. Watts Bar prefers to use the expanded reference section format.

6. ACOT has been changed to COT to reflect the use of digital systems, such as Eagle-21. Westinghouse is making this change to the RSTS.

7. LCO 3.9.3 Required Action A.2 which specifies suspension of positive reactivity additions has been revised to only apply to positive reactivity additions which would reduce the boron concentration. This change will permit addition of water with a boron concentration lower than that currently in the RCS, but greater than the COLR boron concentration limit. This change will also permit positive reactivity additions such as small volume chemical additions and normal plant cool down. Positive reactivity additions due to removal of control elements continue to be suspended in accordance with Required Action A.1.

8. LCO 3.9.3, Required Action A.3 which specifies that action be initiated to restore the source range neutron monitor within 7 days has been deleted. Completion of Required Actions A.1 and A.2 places the plant in a condition which effectively eliminates the possibility for a criticality event for which the source range neutron flux monitors are assumed to function. It is not considered that the additional Required Action to restore the monitor is necessary under these conditions. Further, since normal plant operating practice is to restore compliance with the LCO, Required Action A.3 is not needed.

9. SR 3.9.3.2 has been revised to specify performance of a CHANNEL CALIBRATION for the source range neutron flux monitors. The CHANNEL OPERATIONAL TEST is not appropriate in MODE 6, since the only function of the monitors is to provide information to the operator. Performance of the COT confirms that the channel will initiate equipment actuation, which are not required in MODE 6. In conjunction with the specification of the CHANNEL CALIBRATION, a note to the SR is needed to exclude the neutron detectors from the CHANNEL CALIBRATION. This note is required since it is not possible to perform the CHANNEL CALIBRATION on the detector. This note is specified for the same SR in LCO 3.3.1.

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by [four] bolts;

b. One door in each airlock closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either: A

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE Containment ~~Purge and Exhaust~~ Vent Isolation System. 1

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS. AND A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	A Verify each required containment ~~building~~ penetration is in the required status.	7 days
SR 3.9.4.2	Vent isolation 1 Verify each required containment ~~purge and exhaust~~ valve actuates to the isolation position on an actual or simulated actuation signal.	[18] months

B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, a release of fission-product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to contain fission-product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment

(continued)

BASES

BACKGROUND (continued)

OPERABILITY is required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed.

The requirements for containment penetration closure ensure that a release of fission-product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission-product radioactivity release from containment due to a fuel-handling accident during refueling.

INSERT A

2

~~The Containment Purge and Exhaust System includes two subsystems. The normal subsystem includes a 42-inch purge penetration and a 42-inch exhaust penetration. The second subsystem, a mini-purge system, includes an 8-inch purge penetration and an 8-inch exhaust penetration. During MODES 1, 2, 3, and 4, the two valves in each of the normal purge and exhaust penetrations are secured in the closed position. The two valves in each of the two mini-purge penetrations can be opened intermittently, but are closed automatically by the Engineered Safety Features Actuation System (ESFAS). Neither of the subsystems is subject to a Specification in MODE 5.~~

~~In MODE 6, large air exchangers are necessary to conduct refueling operations. The normal 42-inch purge system is used for this purpose, and all four valves are closed by the ESFAS in accordance with LCO 3.3.2.~~

~~[The mini-purge system remains operational in MODE 6, and all four valves are also closed by the ESFAS.~~

~~or~~

~~The mini-purge system is not used in MODE 6. All four 8-inch valves are secured in the closed position.]~~

(continued)

INSERT A

The Reactor Building Purge Ventilation System operates to supply outside air into the containment for ventilation and cooling or heating, to equalize internal and external pressures, and to reduce the concentration of noble gases within containment prior to and during personnel access. The supply and exhaust lines each contain two isolation valves. Because of their large size, the 24-inch containment lower compartment purge valves are physically restricted to ≤ 50 degrees open. The Reactor Building Purge and Ventilation System valves can be opened in MODES 5 and 6, but are closed automatically by the Engineered Safety Features Actuation System (ESFAS). In MODE 6, large air exchanges are necessary to conduct refueling operations. The normal 24-inch purge system is used for this purpose. The ventilation system must be either isolated or capable of being automatically isolated upon detection of high radiation levels within containment.

BASES

BACKGROUND (continued)

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during fuel movements (Ref. 1). NRC 7

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel-handling accident. The fuel-handling accident is a postulated event that involves damage to irradiated fuel (Ref. 2). Fuel-handling accidents, analyzed in Reference 3, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.7, "Refueling Cavity Water Level," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure that the release of fission-product radioactivity, subsequent to a fuel-handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. Standard Review Plan Section 15.7.4, Rev. 1 (Ref. ~~2~~ 3), defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values. The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values or the NRC staff-approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits). 3

3 Containment penetrations satisfy Criterion 3 of the NRC Interim Policy Statement.

LCO

1 This LCO limits the consequences of a fuel-handling accident in containment by limiting the potential escape paths for fission-product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge and exhaust penetrations. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these

ventilation system reactor building

(continued)

BASES

LCO (continued)

① Ventilation

penetrations are isolable by the Containment ~~Purge and Exhaust~~ Isolation System. The OPERABILITY requirements for this LCO ensure that the automatic purge and exhaust valve closure times specified in the FSAR can be achieved and, therefore, meet the assumptions used in the safety analysis to ensure that releases through the valves are terminated, such that radiological doses are within the acceptance limit.

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel-handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1, "Containment." In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel-handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1 and A.2

①

Ventilation

If the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, including the Containment ~~Purge and Exhaust~~ Isolation System not capable of automatic actuation when the purge and exhaust valves are open, the plant must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude moving a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This surveillance demonstrates that each of the containment penetrations required to be in its closed position is in

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1 (continued)

that position. The surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. Also the surveillance will demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed by an OPERABLE automatic containment ~~purge and exhaust~~ ventilation isolation signal. (1)

The surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. The surveillance interval is selected to be commensurate with the normal duration of time to complete fuel-handling operations. (5) ~~A surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO.~~ As such, this surveillance ensures that a postulated fuel-handling accident that releases fission-product radioactivity within the containment will not result in a release of fission-product radioactivity to the environment.

SR 3.9.4.2

This surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 18-month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. (3) LCO 3.3.6, "Containment Vent Isolation Instrumentation," ~~In LCO 3.3.2, "Engineered Safety Features Actuation System Instrumentation," the Containment Purge Isolation Signal System~~ requires a CHANNEL CHECK every ~~7 days~~ 12 hours (3) and an ~~ANALOG~~ CHANNEL OPERATIONAL TEST every 31 days to ensure the channel OPERABILITY during refueling operations. (6) Every 18 months a CHANNEL CALIBRATION is performed. The system actuation response time is demonstrated every 18 months, during refueling, on a STAGGERED TEST BASIS. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel-handling accident

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.4.2 (continued)

to limit a release of fission-product radioactivity from the containment.

REFERENCES

4

~~1. GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.~~

~~2. FSAR, Section [15.4.5].~~

~~3. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.~~

1. "Use of Silicone Sealant to Maintain Containment Integrity - ITS," GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.

2. Watts Bar FSAR ~~(Unit 1)~~, Section 15.4.5, "Design Basis Fuel Handling Accidents."

3. NUREG-0800, Standard Review Plan, Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents," Rev. 1, July 1981.

JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.9.4

A. This change reflects comments made by the industry to the NRC at the RSTS P&R review meeting held in Irvine, CA July 13 - 20, 1992.

1. Change to reflect plant specific terminology.

2. Change to reflect plant specific system design/operation.

3. Changes to correct errors in the RSTS. Westinghouse has acknowledged the errors and has notified NRC

4. Watts Bar prefers to use the expanded reference section format.

5. The number of times the SR is performed is dependent on the length of the outage, however sentence deleted since it provides no useful information about why the surveillance is required.

6. ACOT has been changed to COT to reflect the use of digital systems such as Eagle-21. Westinghouse is making this change to the RSTS.

7. Editorial change for clarification.

3.9 REFUELING OPERATIONS

3.9.5 Residual Heat Removal (RHR) and Coolant Circulation—High Water Level

LCO 3.9.5 One RHR loop shall be OPERABLE and in operation.

---------------------------NOTE---------------------------
The required RHR loop may be removed from operation for ≤ 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs provided no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration.

A,4

APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. RHR loop requirements not met.	A.1	Suspend operations involving a reduction in reactor coolant boron concentration.	Immediately
	<u>AND</u>		
	A.2	Suspend loading irradiated fuel assemblies in the core.	Immediately
	<u>AND</u>		
	A.3	Initiate action to satisfy RHR loop requirements.	Immediately
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.4	Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ [~~2,800~~] gpm.	12 hours

~~2800~~
2500

1

B 3.9 REFUELING OPERATIONS

B 3.9.5 Residual Heat Removal (RHR) and Coolant Circulation—High Water Level

BASES

BACKGROUND

The purposes of the RHR System in MODE 6 are to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s), where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding which is a fission-product barrier. One train of the RHR System is required to be operational in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit de-energizing the RHR pump for short durations, under the condition that the boron concentration is not diluted. This conditional de-energizing of the RHR pump does not result in a challenge to the fission-product barrier.

Although the RHR System does not meet a specific criterion of the NRC Policy Statement, it was identified in the NRC Policy Statement as an important contributor to risk

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

reduction. Therefore, the RHR System is retained as a Specification.

LCO

Only one RHR loop is required for decay heat removal in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange. Only one RHR loop is required to be OPERABLE because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least one RHR loop must be OPERABLE and in operation to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE RHR loop includes an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the low-end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

The LCO is modified by a Note that allows the required operating RHR loop to be removed from service for up to 1 hour per 8-hour period. ~~during the performance of CORE ALTERATIONS in the vicinity of reactor vessel hot legs.~~ A,4 This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot-leg nozzles, and RCS-to-RHR isolation valve testing. During this 1-hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23-ft water level was selected because it corresponds to the 23-ft requirement established for fuel movement in LCO 3.9.7, "Refueling Cavity Water Level." Requirements for the RHR

(continued)

BASES

APPLICABILITY (continued)

System in other MODES are covered by LCOs in Section 3.4, "Reactor Coolant System," and in Section 3.5, "Emergency Core Cooling Systems." RHR loop requirements in MODE 6, with the water level < 23 ft, are located in LCO 3.9.6, "Residual Heat Removal and Coolant Circulation—Low Water Level."

ACTIONS

RHR loop requirements are met by having one RHR loop OPERABLE and in operation, except as permitted in the Note to the LCO.

A.1

If RHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations cannot occur by the addition of water with a lower boron concentration than that contained in the RCS because all of unborated water sources are isolated.

2

Reduced boron concentrations can occur by the addition of water with lower boron concentration than that contained in the RCS. Therefore, actions that result in an unplanned boron dilution shall be suspended immediately.

A.2

If RHR loop requirements are not met, actions shall be taken immediately to suspend loading irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading a fuel assembly, is a prudent action under this condition.

(continued)

BASES

ACTIONS (continued)

A.3

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 and the refueling water level ≥ 23 ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately.

A.4

If RHR loop requirements are not met, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures dose limits are not exceeded.

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This surveillance demonstrates that the RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the RHR System.

REFERENCES 3

1. Watts Bar FSAR, Section ~~[~~5.5.7~~]~~, "Residual Heat Removal System."

JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.9.5

A. This change reflects comments made by the industry to the NRC at the RSTS P&R review meeting held in Irvine, CA July 13 - 20, 1992.

1. Change to reflect plant specific parameter value.

2. Change to correct error in the RSTS. This paragraph is contradictory to the preceding sentence and is not consistent with Bases 3.9.6.

3. Watts Bar prefers to use the expanded reference section format.

4. The LCO 3.9.5 note has been revised to delete "during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs." The note allowance to remove the required RHR loop from operation is based on the RCS heat-up rate without the RHR loop in operation, which applies to all MODE 6 operations (except during boron dilution operations, which are excluded by the note). There is no technical basis for restricting this allowance to just during CORE ALTERATIONS in the vicinity of the hot legs. This restriction is <u>not</u> specified in W-STS, Revision 4a.

3.9 REFUELING OPERATIONS

3.9.6 Residual Heat Removal (RHR) and Coolant Circulation — Low Water Level

LCO 3.9.6 Two RHR loops shall be OPERABLE, and one RHR loop shall be in operation.

A, 1 INSERT I-2

APPLICABILITY: MODE 6 with the water level < 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Fewer than the required number of RHR loops OPERABLE.	A.1	Initiate action to restore required RHR loops to OPERABLE status.	Immediately
	OR		
	A.2	Initiate actions to establish ≥ 23 ft of water above the top of reactor vessel flange.	Immediately A
B. No RHR loop in operation.	B.1	Suspend operations involving a reduction in reactor coolant boron concentration.	Immediately
	AND		
	B.2	Initiate action to restore one RHR loop to operation.	Immediately
	AND		
			(continued)

INSERT I-2

Only one RHR loop needs to be OPERABLE and in operation and

---------------------NOTE---------------------
Prior to initial criticality, the required RHR loop may be removed from operation for ≤ 1 hour per 8 hour period provided no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.3	Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.6.1	Verify one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ [~~2,800~~] gpm. 2000	12 hours

2

B 3.9 REFUELING OPERATIONS

B 3.9.6 Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level

BASES

BACKGROUND

The purposes of the RHR System in MODE 6 are to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchangers where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass lines. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant will eventually challenge the integrity of the fuel cladding, which is a fission-product barrier. Two trains of the RHR System are required to be OPERABLE, and one train in operation, in order to prevent this challenge.

Although the RHR System does not meet a specific criterion of the NRC Policy Statement, it was identified in the NRC Policy Statement as an important contributor to risk reduction. Therefore, the RHR system is retained as a Specification.

(continued)

BASES (continued)

LCO 1

¶ The LCO is modified by a Note that allows only one RHR loop to be OPERABLE and in operation prior to the initial criticality of the unit. The Note also allows the loop to be removed from service for up to 1 hour per 8-hour period, ~~during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.~~ provided no operations are permitted that would cause a dilution of RCS boron concentration. This allowance is provided only for the initial criticality since there is no decay heat present.

In MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, both RHR loops must be OPERABLE. Additionally, one loop of RHR must be in operation in order to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE RHR loop consists of an RHR pump, a heat exchanger, valves, piping, instruments and controls to ensure an OPERABLE flow path and to determine the low-end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

APPLICABILITY

Two RHR loops are required to be OPERABLE, and one RHR loop must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal. Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, "Reactor Coolant System," and in Section 3.5, "Emergency Core Cooling Systems." RHR loop requirements in MODE 6, with the water level ≥ 23 ft, are located in LCO 3.9.5, "Residual Heat Removal and Coolant Circulation—High Water Level."

ACTIONS

A.1 and A.2

If fewer than the required number of RHR loops are OPERABLE, actions shall be immediately initiated and continued until the RHR loop is restored to OPERABLE status and to operation, or until ≥ 23 ft of water level is established above the reactor vessel flange. When the water level is ≥ 23 ft above the reactor vessel flange, the Applicability changes to that of LCO 3.9.5, "Residual Heat Removal and Coolant Circulation—High Water Level," and only one RHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

(continued)

BASES

ACTIONS (continued)

B.1

If no RHR loop is in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations cannot occur by the addition of water with a lower boron concentration than that contained in the RCS because all of the unborated water sources are isolated.

B.2

If no RHR loop is in operation, actions shall be initiated immediately and continued to restore one RHR loop to operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE RHR loops and one operating RHR loop should be accomplished expeditiously.

B.3

If no RHR loop is in operation, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures that dose limits are not exceeded.

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

This surveillance demonstrates that one RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability, and to prevent thermal and boron stratification in the core. In addition, during operation of the RHR loop, with the water level in the vicinity of the reactor vessel nozzles, the RHR pump suction requirements must be met. The Frequency of 12 hours is

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.9.6.1 (continued) sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator for monitoring the RHR System in the control room.
REFERENCES 3	1. Watts Bar FSAR, Section ~~[~~5.5.7~~]~~, "Residual Heat Removal System."

JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.9.6

A. This change reflects comments made by the industry to the NRC at the RSTS P&R review meeting held in Irvine, CA July 13 - 20, 1992.

1. A NOTE has been added to LCO 3.9.6 which only requires one OPERABLE RHR loop and which permits removal of the required RHR loop from operation prior to initial criticality except during boron dilution operations. Even with the low water level (< 23 ft), there is insufficient decay heat in a new core, prior to initial criticality, to cause excessive RCS heat-up with the RHR loop removed from service. This NOTE has been added consistent with the existing STS NUREG-0452 for NTOL plants. The Note is only applicable for prior to initial criticality which affects Watts Bar.

2. Change to reflect plant specific parameter value.

3. Watts Bar prefers to use the expanded reference section format.

3.9 REFUELING OPERATIONS

3.9.7 Refueling Cavity Water Level

LCO 3.9.7 Refueling cavity water level shall be maintained ≥ 23 ft above the top of reactor vessel flange.

4 ------------ NOTE ------------
This LCO is not required to be met for the initial core loading.

APPLICABILITY: During CORE ALTERATIONS
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately
	<u>AND</u>	
	A.3 Initiate actions to restore refueling cavity ~~canal~~ water level to within limits. A 1	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.7.1 Verify refueling cavity water level is ≥ 23 ft above the top of reactor vessel flange.	24 hours

B 3.9 REFUELING OPERATIONS

B 3.9.7 Refueling Cavity Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies or performance of CORE ALTERATIONS within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pool. Sufficient water is necessary to retain iodine fission-product activity in the water in the event of a fuel-handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to < 25% of 10 CFR 100 limits, as provided by the guidance of Reference 3.

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS and during movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel-handling accident in containment, as postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel-assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel-rod iodine inventory (Ref. 1).

The fuel-handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 100 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel-handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Ref. 4 and 5).

Refueling cavity water level satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel-handling accident inside containment are within acceptable limits, as provided by the guidance of Reference 3.

4 INSERT

APPLICABILITY

LCO 3.9.7 "Refueling Cavity Water Level," is applicable during CORE ALTERATIONS and when moving irradiated fuel assemblies within containment. The LCO minimizes the possibility of a fuel-handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are not present in containment, there can be no significant radioactivity release as a result of a postulated fuel-handling accident. Requirements for fuel-handling accidents in the spent-fuel pool are covered by LCO 3.7.~~11~~ 13, "Fuel Storage Pool Water Level."

2

ACTIONS

A.1 and A.2

With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving CORE ALTERATIONS or movement of irradiated fuel assemblies within the containment shall be suspended immediately to ensure that a fuel-handling accident cannot occur.

The suspension of CORE ALTERATIONS and fuel movement shall not preclude moving a component to a safe position.

A.3

In addition to immediately suspending CORE ALTERATIONS or movement of irradiated fuel, actions to restore refueling ~~canal~~ cavity water level must be initiated immediately.

1

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis

(continued)

INSERT

A Note has been added which states that the LCO is not applicable for the initial core loading. In this specific case, the fuel assemblies and core components are not irradiated and the postulated fuel handling accident is not credible. The water level in the initial fuel loading will be maintained just below the reactor vessel flange to allow the refueling cavity and spent fuel pool to remain dry.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1 (continued)

for the analysis of the postulated fuel-handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel-handling accident inside containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.

REFERENCES

3

1. ~~Regulatory Guide 1.25, March 23, 1972.~~

2. ~~FSAR, Section [15.4.5].~~

3. ~~NUREG-0800, Section 15.7.4.~~

4. ~~10 CFR 100.10.~~

5. Malinowski, D. D., Bell, M. J., Duhn, E., and Locante, J., WCAP-828, Radiological Consequences of a Fuel Handling Accident, December 1971.

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," U.S. Nuclear Regulatory Commission, March 23, 1972.

2. Watts Bar FSAR (Unit 1), Section 15.4.5 "Fuel Handling Accident."

3. NUREG-0800, "Standard Review Plan," Section 15.7.4, "Radiological Consequences of Fuel-Handling Accidents," U.S. Nuclear Regulatory Commission.

4. Title 10, Code of Federal Regulations, Part 20.101(a), "Radiation Dose Standards for Individuals in Restricted Areas."

JUSTIFICATIONS FOR DEVIATIONS FROM RSTS 3.9.7

A. This change reflects comments made by the industry to the NRC at the RSTS P&R review meeting held in Irvine, CA July 13 - 20, 1992.

1. Change to correct error in the standard.

2. Change to reflect Watts Bar plant specific reference.

3. Watts Bar prefers to use the expanded reference section format.

4. The current STS Revision 4a only requires this specification when the fuel assemblies or control rods have been irradiated. The new STS added "Core Alteration" to the applicability as a simplification to the previous wording. However, this change now makes the LCO applicable during movement of core components whether they are irradiated or not. This goes beyond the basis for the LCO which is to mitigate a FHA when moving <u>irradiated</u> fuel/components. Watts Bar needs this change to permit the initial core loading water level to remain below the reactor vessel flange. This will ensure the refueling cavity and spent fuel area remain dry to support the proposed installation of new racks during Cycle 1.

FINAL DRAFT

APR 1 4 1985

3/4.9.13 REACTOR BUILDING PURGE ~~VENTILATION SYSTEM~~ Air Cleanup Units

LIMITING CONDITION FOR OPERATION

LCO 3.9.8

~~3.9.13~~ The Reactor Building Purge ~~Ventilation Systems~~ Air Cleanup Units shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

1 a. With one Reactor Building Purge ~~Ventilation System~~ Air Cleanup Unit inoperable, CORE ALTERATIONS or movement of irradiated fuel within the containment may proceed provided the inoperable air cleanup unit is isolated and the OPERABLE Reactor Building Purge Ventilation System is ~~capable of being powered from an OPERABLE emergency power source and is~~ in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.

2 b. With no Reactor Building Purge ~~Ventilation System~~ air cleanup unit OPERABLE, suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel within the containment until at least one Reactor Building Purge Ventilation System is restored to OPERABLE status.

3 ~~c. The provisions of Specification 3.0.4 are not applicable.~~

SURVEILLANCE REQUIREMENTS

~~4.9.13 The above required Reactor Building Purge Ventilation Systems shall be demonstrated OPERABLE:~~

5 ~~a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes;~~

~~b. At least once per 18 months, or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system, by:~~

~~1) Verifying that the system satisfies the in-place penetration and bypass leakage acceptance criteria of less than 1% and uses the test procedure guidance of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is equal to 14,000 cfm ±10%;~~

SR 3.9.8.1 Perform required filter testing in accordance with the Ventilation Filter Testing Program.

FINAL DRAFT

APR 14 1985

2) Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 10%; and

3) Verifying a system flow rate of 14,000 cfm ±10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation, by verifying within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 10%;

d. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 14,000 cfm ±10%; and

2) Verifying that on a High Radiation test signal, the system isolates.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of 14,000 cfm ±10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 14,000 cfm ±10%.

FINAL DRAFT

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 AUXILIARY BUILDING GAS TREATMENT SYSTEM

The limitations on the Auxiliary Building Gas Treatment System ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. Operation of the system with the heaters operating to maintain low humidity (≤ 70%) for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses. ANSI N510-1975 will be used as a procedural guide for surveillance testing.

3/4.9.13 REACTOR BUILDING PURGE VENTILATION SYSTEM

The limitations on the Reactor Building Purge Ventilation System ensure that all radioactive material released from an irradiated fuel assembly inside containment will be filtered through the HEPA filters and charcoal adsorbers prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumption of the safety analyses. ANSI N510-1975 will be used as a procedural guide for surveillance testing.

5

JUSTIFICATIONS FOR DEVIATIONS

NOTE: THIS SPECIFICATION IS NOT INCLUDED IN THE NUREG-1431 STANDARD TECHNICAL SPECIFICATIONS FOR WESTINGHOUSE PLANTS. THIS SPECIFICATION HAS BEEN ADDED BASED ON THE RESULTS OF THE CRITERIA APPLICATION REPORT FOR WATTS BAR. THE JUSTIFICATIONS DENOTE CHANGES FROM THE 1985 DRAFT TS FOR WATTS BAR AND THE CONVERSION TO THE NEW RSTS FORMAT.

1. Action a. was modified into Conditions and Required Actions (A.1 and A.2) consistent with the standard RSTS format. The Reactor Building purge fans are not safety related and do not receive 1E power. Isolation of the inoperable purge air filter performs the required safety function. Therefore references to the "ventilation system" have been replaced by "air cleanup units."

   Immediately was added as a completion time consistent with standard RSTS format and is consistent with the required actions for the Fuel Building Air cleanup system during movement of irradiated fuel (STS LCO 3.7.13).

2. Action b. was modified into Conditions and Required Actions (B.1 and B.2) consistent with the standard RSTS format.

3. Action c. has been deleted. Since the actions allow for unlimited operation, an exception to the new TR 3.0.4 is no longer necessary.

4. The requirements of SR 4.9.13.a and b. have been combined into a single surveillance requirement 3.9.5.1 which requires testing in accordance with the Ventilation Filter Testing Program, Section 5.7.2. Only the air cleanup filters perform a safety function and, therefore, verification of fan operation is not required.

5. The old format Bases have been expanded consistent with the Bases sections of the RSTS.

4.0 DESIGN FEATURES

4.1 Site

4.1.1 Site and Exclusion Boundaries

9 The site and exclusion boundaries [] shall be as ~~described or~~ ~~as~~ shown in Figure 4.1-1[]. 3

4.1.2 Low Population Zone (LPZ)

9 The LPZ [] shall be as ~~described or as~~ shown in Figure 4.1-2[] 3
5 (within the 3-mile circle).

4.2 Reactor Core

4.2.1 Fuel Assemblies 1 zirconium alloy 6

9 The reactor shall contain ~~[157]~~ 193 fuel assemblies. Each assembly shall consist of a matrix of ~~Zircaloy clad~~ fuel rods with an initial composition of natural ~~or~~ and/or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

7

4.2.2 [Control Rod] Assemblies

9 The reactor core shall contain ~~[157]~~ [53] 1 [control rod] assemblies. The control material shall be ~~[silver indium cadmium,~~ boron carbide ~~or hafnium metal]~~ ~~as approved by the NRC.~~ ~~and~~ with silver indium cadmium tips. 6

(continued)

(continued)

4.3 Fuel Storage

4.3.1 Criticality

(9) 4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of ~~[4.5]~~ 3.15 (8) weight percent;

b. K_{eff} no greater than 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties as described in [Section 9.1 of the FSAR];

c. A nominal ~~[9.15]~~ [10.7] (2) -inch center-to-center distance between fuel assemblies placed in [the high density fuel storage racks];]

[d. A nominal [10.95]-inch center-to-center distance between fuel assemblies placed in [low density fuel storage racks];]

[e. New or partially spent fuel assemblies with a discharge burnup in the "acceptable range" may be allowed unrestricted storage in [either] fuel storage racks(s); and]

[f. New or partially spent assemblies with a discharge burnup in the "unacceptable range" will be stored in compliance with the [FSAR, approved procedures, Licensee Controlled Specification, or etc.].] (3)

4.3.1.2 → The new fuel storage racks are designed and shall be maintained with: (9)

a. Fuel assemblies having a maximum U-235 enrichment of ~~[4.5]~~ 4.3 (1) weight percent;

b. K_{eff} no greater than 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties as described in [Section 9.1 of the FSAR];

c. K_{eff} no greater than 0.98 if moderated by aqueous foam, which includes an allowance for uncertainties as described in [Section 9.1 of the FSAR]; and

(continued)

4.3.~~1~~.2 (continued) 1

9

d. A nominal 21 ~~[10.95]~~-inch center-to-center distance between fuel assemblies placed in the storage racks.

4.3.~~2~~3 Drainage

9

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation ~~[23 ft]~~ 745'-1½".

4.3.~~3~~4 Capacity 1

9

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than ~~[1,737]~~ [1312] fuel assemblies.

1

(continued)

4

~~This figure shall consist of [a map of] the site area and provide, at a minimum, the information described in Section [2.1.2] of the FSAR relating to [the map].~~

Input Insert From
Page 4.0-4A Here

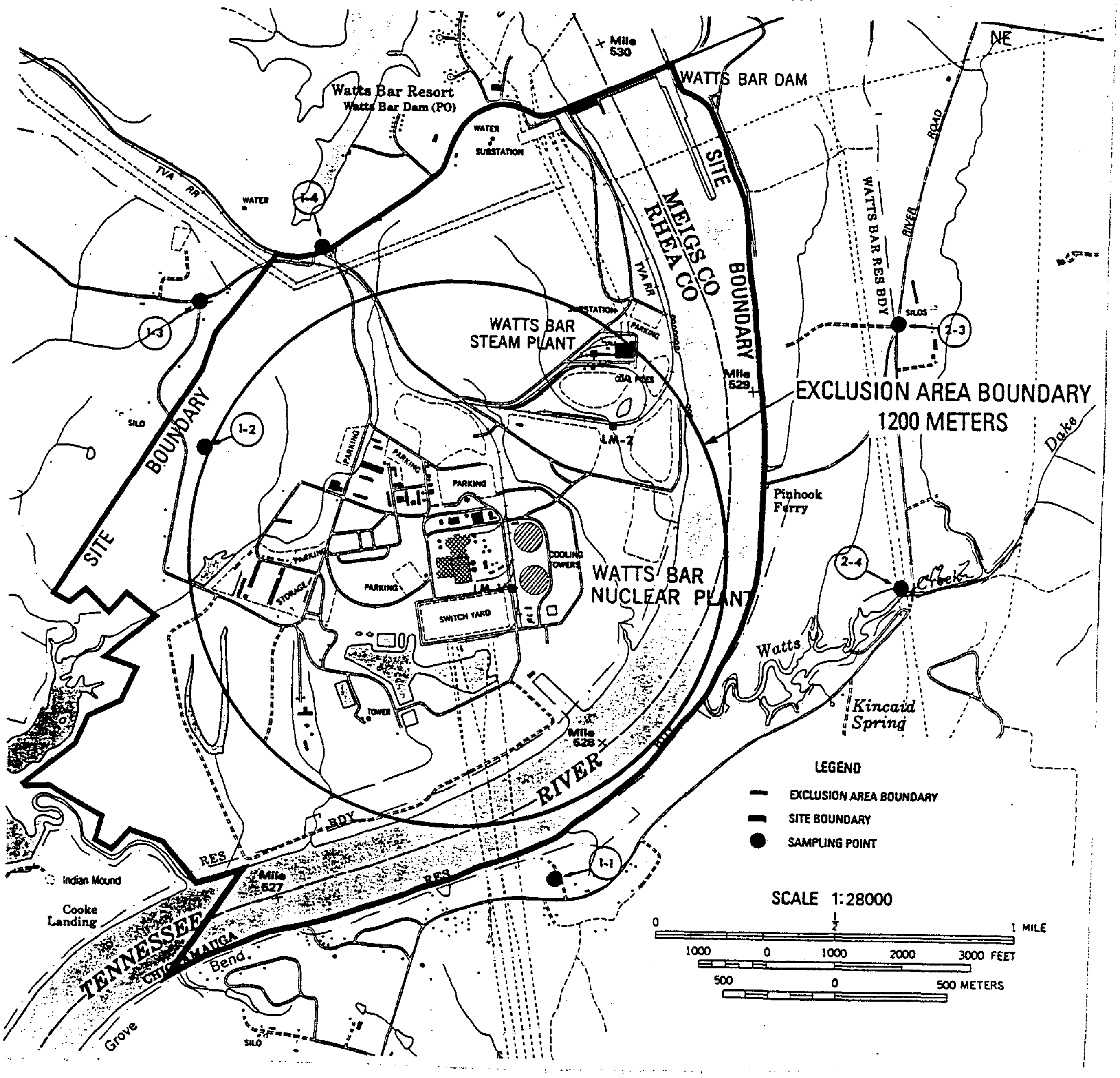
Figure 4.1-1 (page 1 of 1)
Site and Exclusion Area Boundaries

~~(continued)~~

(4)

INSERT

WATTS BAR NUCLEAR PLANT
SITE BOUNDARY/EXCLUSION AREA BOUNDARY



4

~~This figure shall consist of [a map of] the site area showing the LPZ boundary. Features such as towns, roads, and recreational areas shall be indicated in sufficient detail to allow identification of significant shifts in population distribution within the LPZ.~~

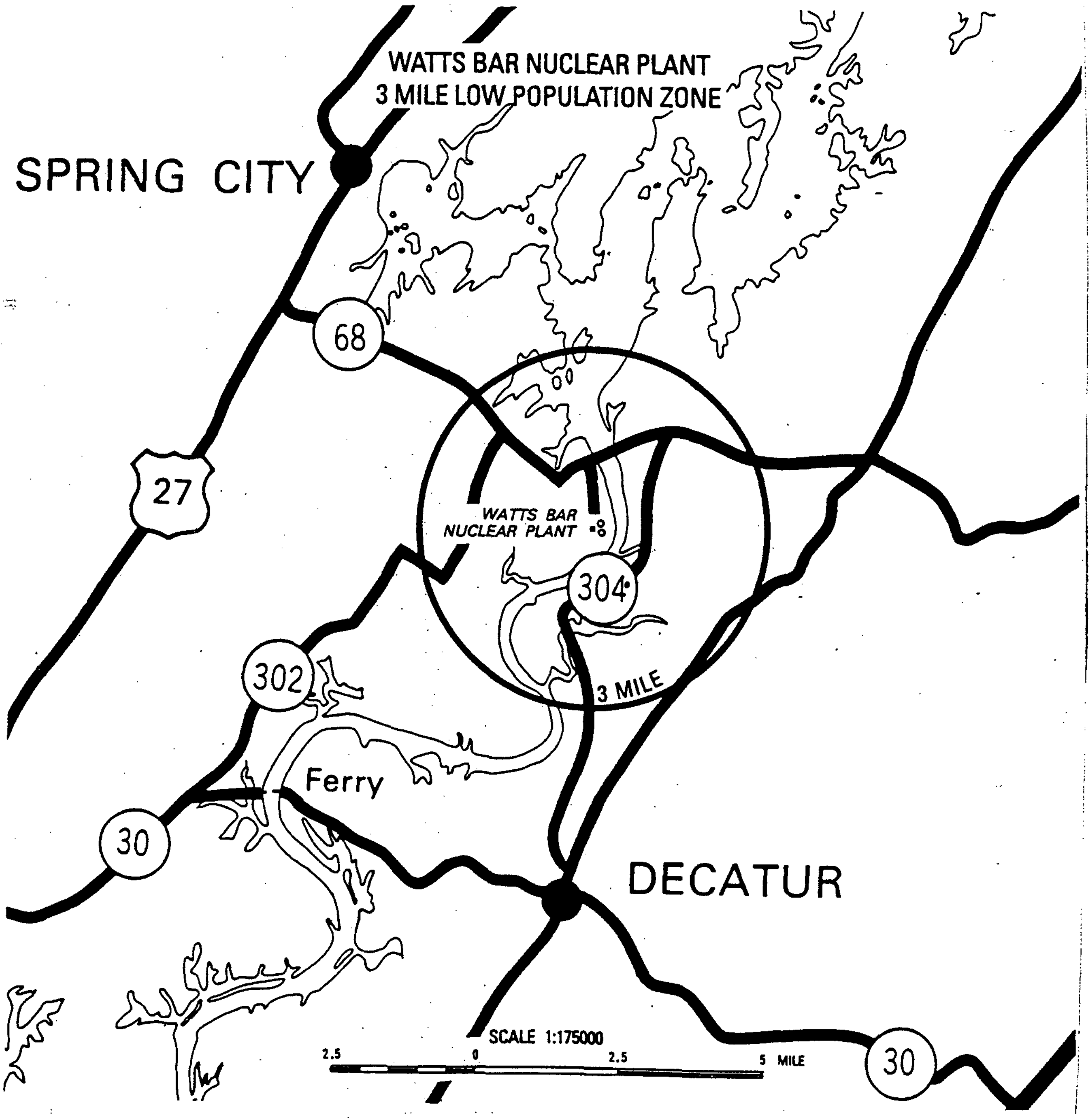
INPUT INSERT FROM
PAGE 4.0-5A HERE

Figure 4.1-2 (page 1 of 1)
Low Population Zone

~~(continued)~~

(4)

INSERT



4.0-5A

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 4.0

1. Values changed for Watts Bar specific design.

2. Center to center spacing value changed to agree with nominal value for Watts Bar design.

3. Deleted non - Watts Bar wording.

4. Change to provide WBN specific figures for Site and Exclusion Area Boundary, and Low Population Zone.

5. Change to add clarifying phrase that specifies only the 3-mile circle from a new "10-mile" figure encompasses the low population zone.

6. Change to reflect Watts Bar specific terminology.

7. Editorial change for enhanced clarity.

8. Value changed for Watts Bar specific design and to account for uncertainties in the analysis described in IE Notice 92-21, "Spent Fuel Pool Reactivity Calculations," dated March 24, 1992.

9. Change reflects comments made by the industry to the NRC at the RSTS Proof and Review meeting held in Irvine, CA in July, 1992.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 3 ~~The [Plant Superintendent] shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.~~

5.1.2 2 The [Shift Operations 1 Supervisor (SOS)] 2 shall be responsible for the control room command function. A management directive to this effect, signed by the ~~[highest level of corporate or site management]~~ Site Vice-President 1 shall be issued annually to all station personnel. During any absence of the ~~[SS]~~ SOS 1 from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with a valid Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the ~~[SS]~~ SOS 1 from the control room while the unit is in MODE 5 or 6, an individual with a valid SRO license or Reactor Operator license shall be designated to assume the control room command function.

2

The [Site Vice President] shall be responsible for overall activities of the site, while the Plant Manager shall be responsible for overall unit operation. The Site Vice President and Plant Manager shall delegate in writing the succession to this responsibility during their absence.

8 The Plant Manager, or his designee, in accordance with approved Administrative procedures, shall approve each proposed test or experiment and proposed changes and modifications to unit systems or equipment that affect nuclear safety prior to implementation.

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 5.1

1. Change to reflect Watts Bar specific management titles/terminology.

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

3. Change to reflect Watts Bar's utilization of both a Site Vice President and a Plant Manager, and their respective responsibilities.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

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

(8) (8)

a. (8) Lines of authority, responsibility, and communication shall be defined and established ~~for the~~ throughout highest management levels, ~~through~~ intermediate levels, ~~to~~ and ~~including~~ all operating organization positions. These relationships shall be documented and updated, as appropriate, in ~~the form of~~ (8) organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in (1) the ~~[FSAR];~~ Nuclear Power Organization Topical Report (TVA-NPOD89-A);

b. (2) The ~~[Plant Superintendent]~~ PLANT MANAGER shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant;

c. (2) The ~~[a specified corporate executive position]~~ Site Vice President shall have ~~corporate~~ responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety; and

d. (2) The individuals who train the operating staff, carry out ~~health physics~~ Radiological controls, or perform quality assurance functions may report to the appropriate onsite manager; however, ~~they~~ these individuals (8) shall have sufficient organizational freedom to ensure their independence from operating pressures.

(continued)

(continued)

5.2.2 Unit Staff

The unit staff organization shall be as follows:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 5.2.2-1.

b. At least one licensed Reactor Operator (RO) shall be present in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room.

(2) Radiological Controls (3)

c. A ~~[Health Physics~~ Technician] shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

d. Either a licensed SRO or licensed SRO limited to fuel handling who has no concurrent responsibilities during this operation shall be present during fuel handling and shall directly supervise all CORE ALTERATIONS. (3)

(2) radiological controls technicians

e. [Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions (e.g., licensed SROs, licensed ROs, ~~health physicists~~, auxiliary operators, and key maintenance personnel).

(3) (4) 8, 10 OR 12.

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work an [~~8 or 12~~] hour day, nominal 40-hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;]

(3)

(continued)

5.2.2 Unit Staff (continued)

3

3

2. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period, all excluding shift turnover time;

3. A break of at least 8 hours should be allowed between work periods, including shift turnover time;

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

3 MANAGER 2

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

MANAGER 2 3

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

5

OR

[The amount of overtime worked by unit staff members performing safety related functions shall be limited and controlled in accordance with the NRC Policy Statement on working hours (generic Letter 82-12).]

f. 3 The [~~Operations Manager or Assistant Operations Manager]~~ 2 shall hold an SRO license.

Operations 2

g. The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Operations Supervisor (~~SS~~ SOS) in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit.

SOS 2

individual directly supervising the Shift Operations Supervisor

3

Table 5.2.2-1 (page 1 of 1)
Minimum Shift Crew Composition[a]
~~[Single Unit Facility]~~ 6

POSITION[b]	MINIMUM CREW NUMBER	
	UNIT IN MODE 1, 2, 3, OR 4	UNIT IN MODE 5 OR 6
~~SS~~ SOS	1	1
SRO	1	None
RO	2	1
~~AO~~ AUO	~~2~~ 4	1
STA[c]	1	None

2 7

(a) The shift crew composition may be one less than the minimum requirements of Table 5.2.2-1 for not more than 2 hours to accommodate unexpected absences of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 5.2.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

(b) Table Notation: 2

SOS ~~SS~~ - Shift Operations Supervisor with a Senior Reactor Operator license; 3

SRO - Individual with a Senior Reactor Operator license;

RO - Individual Unit with a Reactor Operator license; 2

AUO ~~AO~~ - Auxiliary Operator;

STA - Shift Technical Advisor.

(c) The STA position may be filled by an on-shift SOS ~~SS~~ or SRO provided the individual meets the Commission Policy Statement on Engineering Expertise on Shift. 2

Table 5.2.2-1 (page 1 of 1)
Minimum Shift Crew Composition[(a)]
[Two Units With a Common Control Room]
(Totals for Both Units)

POSITION[(b)]	MINIMUM CREW NUMBER		
	EACH UNIT IN MODE 1, 2, 3, OR 4	ONE UNIT IN MODE 1, 2, 3, OR 4, AND ONE UNIT IN MODE 5, MODE 6, OR DEFUELED	EACH UNIT IN MODE 5 OR 6 OR DEFUELED
SS	1	1	1
SRO	1	1	None
RO	3	3	2
AO	3	3	3
STA[(c)]	1	1	None

(a) The shift crew composition may be one less than the minimum requirements of Table 5.2.2-1 for not more than 2 hours to accommodate unexpected absence of on-duty shift crew members, provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 5.2.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

(b) Table Notation:

SS - [Shift Supervisor] with a Senior Reactor Operator license for each unit whose reactor contains fuel.
SRO - Individual with a Senior Reactor Operator license for each unit whose reactor contains fuel. Otherwise, provide an individual for each unit who holds a Senior Reactor Operator license for the unit assigned. During CORE ALTERATIONS on either unit at least one licensed SRO or licensed SRO limited to fuel handling, who has no other concurrent responsibilities, must be present.
RO - Individual with a Reactor Operator license or a Senior Reactor Operator license for unit assigned. At least one RO shall be assigned to each unit whose reactor contains fuel and one RO shall be assigned as relief operator for unit(s) in MODE 1, 2, or 3. Individuals acting as relief operators shall hold a license for both units. Otherwise, for each unit, provide a relief operator who holds a license for the unit assigned.
AO - At least one auxiliary operator shall be assigned to each unit whose reactor contains fuel.
STA - Shift Technical Advisor.

(c) The STA position may be filled by an on-shift SS or SRO provided the individual meets the Commission Policy Statement on Engineering Expertise on Shift.

(6)

Table 5.2.2-1 (page 1 of 1)
Minimum Shift Crew Composition[a]
[Two Units With Two Control Rooms]
(Numbers for Each Unit)

POSITION[b]	MINIMUM CREW NUMBER			
	UNIT IN MODE 1, 2, 3, OR 4	UNIT IN MODE 5 OR 6	UNIT IN MODE 1, 2, 3, OR 4, OTHER UNIT IN MODE 5 OR 6 OR DEFUELED	UNIT IN MODE 5 OR 6, OTHER UNIT IN MODE 5 OR 6 OR DEFUELED
SS	1[d]	1[d]	1[d]	1[d]
SRO	1	None	1	None
RO	2	1	2	1
AO	2	1	2	2[e]
STA[c]	1[d]	None	1	None

(a) The shift crew composition may be one less than the minimum requirements of Table 5.2.2-1 for not more than 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 5.2.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

(b) Table Notation:

SS - [Shift Supervisor] with a Senior Reactor Operator license;
SRO - Individual with a Senior Reactor Operator license;
RO - Individual with a Reactor Operator license;
AO - Auxiliary Operator;
STA - Shift Technical Advisor.

(c) The STA position may be filled by an on-shift SS or SRO provided the individual meets the Commission Policy Statement on Engineering Expertise on Shift.

(d) Individual may fill the same position on the other unit if licensed for both.

(e) One of the two required individuals may fill the same position on the other unit.

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 5.2

1. TVA utilizes the Nuclear Power Organization Topical Report, instead of the FSAR, to describe the management "chain-of-command", responsibilities, etc.

2. Change to reflect Watts Bar specific management titles/terminology.

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

4. Change to reflect WBN specific parameter value.

5. Change to delete "either/or" text since WBN chooses to specify overtime policy requirements in the Administrative Controls using the first bracketed option.

6. WBN will utilize the "single unit facility" minimum shift crew composition table for these technical specifications.

7. WBN will require a minimum of four AUOs on shift to satisfy manpower requirements for an Appendix R shutdown.

8. Change reflects comments made by the industry to the NRC at the RSTS Proof and Review meeting held in Irvine, CA in July, 1992.

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

---

(1) ~~5.3.1 Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.~~

for comparable positions as specified in the TVA Nuclear Quality Assurance Plan

(2) ~~[5.3.2]~~ 5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications ~~of [Regulatory Guide 1.8, Revision 2, 1987, or more recent revisions, or ANSI Standard acceptable to the NRC staff]. The staff not covered by [Regulatory Guide 1.8] shall meet or exceed the minimum qualifications of [Regulations, Regulatory Guides, or ANSI standards acceptable to NRC staff].~~ (3) In addition, the Shift Technical Advisor shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

---

(3) (TVA-NQA-PLN89-A).

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 5.3

1. Delete bracketed information intended to be followed as a guide only, not technical specification text.

2. Change to reflect deletion of original Step 5.3.1.

3. Change to reflect personnel qualification requirements specified in the Nuclear Quality Assurance Plan.

5.0 ADMINISTRATIVE CONTROLS

5.4 Training

PLANT MANAGER
1

5.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the ~~[position title]~~ and shall meet or exceed the requirements ~~and recommendations of Section [ ] of [an ANSI standard acceptable to the NRC staff] and 10 CFR 55, and, for appropriate designated positions,~~ shall include familiarization with relevant industry operational experience.
2

of 10 CFR 55 and the TVA Nuclear Quality Assurance Plan (TVA-NQA-PLN89-A), and
2

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 5.4

1. Change to reflect WBN specific management titles/terminology.

2. Change to reflect personnel training requirements specified in the Nuclear Quality Assurance Plan.

5.0 ADMINISTRATIVE CONTROLS

5.5 Reviews and Audits

2

Input Insert from page 5.0-10C

1

~~The licensee shall describe the method(s) established to conduct independent reviews and audits. The methods may take a range of forms acceptable to the NRC. These methods may include creating an organizational unit or a standing or ad hoc committee, or assigning individuals capable of conducting these reviews and audits. When an individual performs a review function, a cross-disciplinary review determination is necessary. If deemed necessary, such reviews shall be performed by the review personnel of the appropriate discipline. Individual reviewers shall not review their own work. Regardless of the method used, the licensee shall specify the functions, organizational arrangement, responsibilities, appropriate ANSI/ANS 3.1-1981 qualifications, and reporting requirements of each functional element or unit that contributes to these processes.~~

~~Reviews and audits of activities affecting plant safety have two distinct elements. The first element is the reviews performed by plant staff personnel to ensure that day-to-day activities are conducted in a safe manner. These reviews are described in Section 5.5.1. The second element, described in Section 5.5.2, is the [offsite] reviews and audits of unit activities and programs affecting nuclear safety that are performed independent of the plant staff. The [offsite] reviews and audits should provide integration of the reviews and audits into a cohesive program that provides senior level utility management with an assessment of facility operation and recommends actions to improve nuclear safety and plant reliability. It should include an assessment of the effectiveness of reviews conducted according to Section 5.5.1.~~

5.5.1 Plant Reviews

Input Insert from Pages 5.0-10A and -10B

~~The licensee shall describe provisions for plant reviews (organization, reporting, records) and the appropriate ANSI/ANS standard for personnel qualification.~~

2 6

5.5.1.1 Functions

3

a. The PORC ~~[plant review method specified in Specification 5.5.1]~~ shall; ~~as a minimum, incorporate functions that:~~

1 ~~a~~. Advise the ~~[~~Plant MANAGER ~~Superintendent]~~ on all matters related to nuclear safety; 4 3

2

3 ~~b~~. Recommend to the ~~[~~Plant MANAGER, or his designee, ~~Superintendent]~~ approval or disapproval of items considered under 4 3 2

2. Recommend to the Plant Manager, or his designee, approval or disapproval of procedures that delegate review responsibilities of items considered under ~~PORC Responsibilities~~ Specifications 5.5.1.2.a and 5.7.1.3;

INSERT

5.5.1 Plant Reviews

knowledgeable in the subject matter

The items in Specification 5.5.1.2 receive varying degrees of review, before final approval or use, as specified in approved administrative procedures. The degree of review shall be commensurate with the potential to affect nuclear safety. As a minimum each item shall be reviewed by at least one ~~knowledgeable~~ individual and a ~~determination of the need for~~ cross-disciplinary review(s) shall be ~~made and~~ obtained before approval.

AS needed

The ~~plant~~ staff shall provide technical and cross-disciplinary reviews. These reviews shall be governed by administrative procedures for items considered under Specification 5.5.1.2. The sponsor of each item shall be responsible for the conduct of all reviews. All new procedures shall be reviewed by the Plant Operations Review Committee (PORC). PORC shall, during its review, make the determination as to the delegation of review requirements for changes to each procedure.

a. Plant ~~Designated~~ Review Process

Technical

~~Designated~~ Reviewers shall be chosen by the discipline supervisors to perform technical reviews based on the individual's training, experience, and knowledge level. ~~Designated~~ Reviewers assigned the responsibility for reviewing for 10 CFR 50.59 requirements shall receive training in this process. ~~Designated~~ Reviewers shall not review their own work. The minimum qualification requirements shall be as recommended in Section 4 of ANSI N18.1-1971.

1. Organization

Each supervisor is responsible for ensuring that technical ~~designated~~ reviewers are available for reviews for the equipment, systems, programs, procedures and other areas under their supervision, AND that reviews ARE Adequate to detect safety questions.

2. Reporting

Technical ~~Designated~~ Reviewers report to their supervisors or PORC on all activities and findings. The signed document processing form(s) shall serve as the reviewer's approval recommendation to the designated Approval Authority.

INSERT

b. Plant Operations Review Committee (PORC)

The PORC shall be ~~chartered, by plant procedure, as~~ the onsite review committee. The committee shall function as a multi-disciplinary review body for items which affect plant nuclear safety. PORC shall be organized and shall conduct business as described below:

1. Composition

Chairman:	Plant Manager
Member:	Operations Manager
Member:	Maintenance Manager
Member:	Technical Support Manager
Member:	Quality Assurance Representative
Member:	Site Radiological Control Manager
Member:	Site Nuclear Engineering Representative

Individuals performing the duties and serving in the official capacity of the above titled member positions may be considered, if qualified, the PORC Member for quorum purposes.

The qualifications required to serve as a member or alternate member shall be specified with the minimum qualifications as recommended in Section 4 of ANSI N18.1-1971, except for the Site Radiological Control Manager who must meet the qualifications of Regulatory Guide 1.8, Rev. 2.

2. Alternates

All alternate members shall be appointed in writing by the PORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PORC activities at any one time.

3. Meeting Frequency

The PORC shall meet on an as needed basis as convened by the PORC Chairman or his designated alternate.

4. Quorum

The PORC quorum shall consist of the Chairman or his designated alternate and four members of which two may be alternates.

5. Reporting

The PORC reports to the Plant Manager on all activities and findings. The meeting minutes shall serve as the official correspondence from PORC to the Plant Manager. PORC recommendations shall be recorded in the minutes and submitted to the Plant Manager by the PORC Chairman.

INSERT

The plant staff organization provides reviews of day-to-day activities to ensure they are conducted in a safe manner. The Plant Operations Review Committee (PORC) is a multi-disciplined committee responsible for providing an oversight review of documents required for the safe operation of the plant. The PORC advises the Plant Manager on all matters related to nuclear safety. Also, Technical Reviewers provide for reviews of procedure changes and proposed changes to structures, systems, and components that affect nuclear safety in their area of expertise. These technical reviews determine the need for a cross-disciplinary review and whether or not an unreviewed safety question is involved.

The Nuclear Assurance organization provides independent technical reviews and audits of activities effecting nuclear plant safety. The Nuclear Safety Review Board (NSRB) is an off site committee which provides senior level oversight of TVA's nuclear program with respect to nuclear safety. The NSRB reviews include the activities of the line organizations, as well as other review, audit and verification organizations. The NSRB advises the Senior Vice President, Nuclear Power (SVP,NP), on the adequacy and implementation of TVAs nuclear safety policies and programs. The NSRB also provides senior level management with an assessment of facility operations and recommendations to improve nuclear safety and plant reliability.

5.5.1.1 Functions (continued)

5 8

a.4

Specifications 5.5.1.2.~~a~~.1 through 5.5.1.2.~~e~~ prior to their implementation, except as provided in Specification 5.7.1.3;

8

~~c.~~ Obtain approval from the [Plant ~~Superintendent]~~, or his designee, in accordance with approved administrative procedures, for each proposed test or experiment and proposed changes and modifications to unit systems or equipment that affect nuclear safety prior to implementation;

5 8

a.3

4 ~~d~~. Determine whether each item considered under Specifications 5.5.1.2.a.1 through 5.5.1.2.~~d~~ constitutes an unreviewed safety question as defined in 10 CFR 50.59; and 2

Input Insert From Page 5.0-11A

6

4 Site 3

And the Nuclear Safety Review Board (NSRB)

5 ~~e~~. Notify the Site [Vice President ~~[Nuclear Operations]~~ of any safety-significant disagreement between the ~~[review organization or individual specified in Specification 5.5.1]~~ PORC 3 and the 4 [Plant ~~Superintendent]~~ within 24 hours. However, the [Plant ~~Superintendent]~~ shall have responsibility for resolution of such disagreements pursuant to Specification 5.1.1. Manager 3

5.5.1.2 Responsibilities

PORC 3

a. The ~~[plant review method specified in Specification 5.5.1]~~ shall be ~~used to conduct, as a minimum, reviews of the following:~~ Responsible for the review of:

2

1 ~~a~~. Station Administrative ~~All proposed~~ procedures required by Specification 5.7.1.1 and changes thereto;

Program descriptions for

2 ~~b~~. ~~All proposed~~ programs required by Specification 5.7.2 and changes thereto;

3 ~~c~~. All proposed changes and modifications to unit systems or equipment that affect nuclear safety;

4 ~~d~~. All proposed tests and experiments that affect nuclear safety; and

5 ~~e~~. All proposed changes to these Technical Specifications (TS), their Bases, and the Operating License.

6

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(continued)

INSERT

b. The ~~Designated~~ Technical Reviewer(s) shall:

1. Advise his supervisor and/or PORC on all matters related to nuclear safety;

2. Determine the need for additional reviews by other disciplines and ensure that identified reviews are conducted for items considered under Specification 5.5.1.2.b prior to their implementation, except as provided in Specification 5.7.1.3;

3. Recommend to the designated Approval Authority, approval or disapproval of items considered under Specification 5.5.1.2.b prior to their implementation, except as provided in Specification 5.7.1.3; AND

4. Ensure PORC reviews those items designated by procedures as requiring a PORC review.;

8 4~~5~~. Determine whether each item considered under Specifications 5.5.1.2.b.1 through 5.5.1.2.b.~~4~~ 3 constitutes an unreviewed safety question as defined in 10 CFR 50.59~~;~~. ~~and~~

6. Notify the PORC of any safety-significant disagreement between reviewing organizations.

INSERT

Technical

b. The ~~Designated~~ Reviewers shall be responsible for the technical review of:

1. Proposed procedures required by Specification 5.7.1.1 and changes thereto;

2. Proposed programs required by Specification 5.7.2 and changes thereto;

3. Proposed changes and modifications to unit systems or equipment that affect nuclear safety;

4. Proposed tests and experiments that affect nuclear safety; and

5. Proposed changes to these Technical Specifications (TS), their Bases, and the Operating License.

(continued) 4

5.5.2 [Offsite] Review and Audit

Input Insert From Page 5.0-12A 2

~~The licensee shall describe the provisions for reviews and audits independent of the plant's staff (organization, reporting, and records) and the appropriate ANSI/ANS standards for personnel qualifications. These individuals may be located onsite or offsite provided organizational independence from plant staff is maintained. The [technical] review responsibilities, Specification 5.5.2.4, shall include several individuals located onsite.~~

5.5.2.1 Functions 3

The NSRB ~~[offsite review and audit provisions specified in Specification 5.5.2]~~ shall, as a minimum, incorporate the following functions that: 4

3 Senior 4 Power 3

a. Advise the Senior [Vice President, Nuclear Power ~~Operations]~~ on all matters related to nuclear safety;

7

b. Advise the management of the audited organization, and [its Corporate Management and Vice President - Nuclear Operations], of the audit results as they relate to nuclear safety;

3 Senior Vice President, Nuclear Power,

b ~~c~~. Recommend to the Senior Vice President, Nuclear Power, ~~management of the audited organization, and its management,~~ any corrective action to improve nuclear safety and plant operation; and

3 Senior 4 Power 3

c ~~d~~. Notify the Senior [Vice President, Nuclear Power ~~Operations]~~ of any safety significant disagreement between the NSRB ~~[review organization or individual specified in Specification 5.5.2]~~ 3 and the [organization or function being reviewed] within 24 hours. 4 4

3 NSRB

5.5.2.2 ~~[Offsite]~~ NSRB Review Responsibilities

The NSRB 3 ~~[review method specified in Specification 5.5.2]~~ shall be responsible for the review of:

a. ~~The safety evaluations for changes to procedures, equipment or systems, and tests or experiments completed under the provisions of 10 CFR 50.59, to verify that such actions do not constitute an unreviewed safety question as defined in 10 CFR 50.59;~~ 7

2 The 10 CFR 50.59 Safety Evaluation Program;

(continued)

INSERT

Offsite Review and Audit

The Nuclear Safety Review Board (NSRB) shall function to provide for independent review as specified in Specification 5.5.2.2 and oversight of the audits and technical reviews as specified in 5.5.2.3 and 5.5.2.4.

The Chairman, members, and alternate members of the NSRB shall meet the qualification requirements of ANSI 18.7, 1976/ANS 3.2 Section 4.3.1.

Nuclear Assurance (NA) shall function to provide audits and onsite technical reviews as specified in Specifications 5.5.2.3 and 5.5.2.4.

NA audit personnel shall meet the qualification requirements as committed to the NRC in TVA's Nuclear Quality Assurance Plan (TVA-NQA-PLN89-A).

NA onsite technical review personnel shall have a bachelor's degree in engineering or equivalent and two to four years experience in their field, including one to two years nuclear experience.

3

5.5.2.2 NSRB ~~[Offsite]~~ Review Responsibilities (continued)

b. Proposed changes to procedures, equipment, or systems that involve an unreviewed safety question as defined in 10 CFR 50.59;

c. Proposed tests or experiments that involve an unreviewed safety question as defined in 10 CFR 50.59;

d. Proposed changes to TS and the Operating License;

e. Violations of codes, regulations, orders, license requirements, and internal procedures or instructions having nuclear safety significance;

f. All Licensee Event Reports required by 10 CFR 50.73;

g. Plant staff performance;

h. Indications of unanticipated deficiencies in any aspect of design or operation of structures, systems, or components that could affect nuclear safety;

i. Significant accidental, unplanned, or uncontrolled radioactive releases, including corrective action to prevent recurrence;

j. Significant operating abnormalities or deviations from normal and expected performance of equipment that affect nuclear safety; and

k. Implementation ~~The performance~~ of the corrective action PROGRAM ~~system~~. 7 2

2 Minutes of each NSRB meeting and reports ~~or records~~ 2 of these reviews shall be forwarded to the Senior 3 ~~[~~Vice President, Nuclear ~~Operations]~~ Power 3 within 30 days following completion of the review.

5.5.2.3 Nuclear Assurance Audit Responsibilities

2 Nuclear Assurance is responsible for the audit program whose ~~The~~ audit responsibilities shall encompass:

a. The conformance of unit operation to provisions contained within the TS and applicable license conditions;

b. The training and qualifications of the unit staff;

(continued)

⑦ Input Insert from Page 5.0-14A here

5.5.2.3 Nuclear Assurance ② Audit Responsibilities (continued)

e. The implementation of all programs required by Specification 5.7.2;

f. Actions taken to correct deficiencies occurring in equipment, structures, systems, components, or method of operation that affect nuclear safety; and

g. Other activities and documents as requested by the Senior ③ ~~[~~Vice President, Nuclear ~~Operations]~~ Power ③.

Reports or records of these audits shall be forwarded to the Senior ③ ~~[~~Vice President, Nuclear ~~Operations] within 30 days following completion of the review.~~ Power and Audited organization as committed to the NRC in the Nuclear Quality Assurance Plan. ⑦ ② ④

5.5.2.4 Nuclear Assurance ② ~~[Technical]~~ ④ Review Responsibilities

② Nuclear Assurance is responsible for ~~The~~ ~~[technical]~~ ④ review whose responsibilities shall encompass:

a. Plant operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources that may indicate areas for improving plant safety;

b. Plant operations, modifications, maintenance, and surveillance to verify independently that these activities are performed safely and correctly and that human errors are reduced as much as practical;

c. Internal and external operational experience information that may indicate areas for improving plant safety; and

d. Making detailed recommendations through the Site ③ ~~[~~Vice President ~~[Nuclear Operations]~~ for revising procedures, equipment modifications or other means of improving nuclear safety and plant reliability.

5.5.3 Records

Written records of reviews and audits shall be maintained. As a minimum these records shall include:

a. Results of the activities conducted under the provisions of Section 5.5;

(continued)

INSERT

c. Safety evaluations for changes to procedures, equipment or systems, and tests or experiments completed under the provisions of 10 CFR 50.59, to verify that such actions do not constitute an unreviewed safety question;

d. The advisement to management of the audited organization, and appropriate senior management, of the audit results as they relate to nuclear safety;

5.5.3 Records (continued)

b. Recommendations to the management of the organization being audited;

c. An assessment of the safety significance of the review or audit findings;

d. Recommended approval or disapproval of items considered under Specifications 5.5.1.2.a .1 through 5.5.1.2.~~e~~ b.5; and

5

e. Determination whether each item considered under Specifications 5.5.1.2.a through 5.5.1.2.~~d~~ constitutes an unreviewed safety question as defined in 10 CFR 50.59.
a.4

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 5.5

1. Change to delete general guidelines for text development.

2. Change to reflect WBN specific plant information.

3. Change to reflect WBN specific organizations or management titles.

4. Format change to delete brackets that identify plant specific information.

5. Change to referenced specification numbers due to renumbering of specifications.

6. WBN utilizes a Technical Reviewer in addition to the PORC for plant reviews.

7. At WBN, this function/responsibility is performed by the organization responsible for auditing, not the organization responsible for independent reviews.

8. Change reflects comments made by the industry to the NRC at the RSTS Proof and Review meeting held in Irvine, CA in July, 1992.

5.0 ADMINISTRATIVE CONTROLS

5.6 Technical Specifications (TS) Bases Control

5.6.1 Changes to the Bases of the TS shall be made under appropriate administrative controls and reviewed according to Specification 5.5.1.

5.6.2 Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

a. A change in the TS incorporated in the license; or

b. A change to the updated FSAR (UFSAR) or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.

5.6.3 The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

5.6.4 Proposed changes that meet the criteria of Specification 5.6.2 ~~(a) or (b) above~~ shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC ~~at least annually.~~ on a frequency consistent with 10 CFR 50.71.

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 5.6

1. Change to correct STS editorial error.

2. Change reflects comments made by the industry to the NRC at the RSTS Proof and Review meeting held in Irvine, CA in July, 1992.

5.0 ADMINISTRATIVE CONTROLS

5.7 Procedures, Programs, and Manuals

5.7.1 Procedures

5.7.1.1 Scope

Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33; (15) (4)

c. Security plan implementation;

d. (2) Site Radiological Emergency plan implementation;

e. Quality assurance for effluent and environmental monitoring;

f. Fire Protection Program implementation; and

g. All programs specified in Specification 5.7.2.

5.7.1.2 Review and Approval

(1) Each procedure of Specification 5.7.1.1, and changes thereto, shall be reviewed MANAGER (2) in accordance with Specification 5.5.1, approved by the [Plant ~~Superintendent]~~ or his designee in accordance with approved administrative procedures prior to implementation (12) , except as specified in Specification 5.7.1.3 and reviewed periodically as set forth in administrative procedures.

5.7.1.3 ~~Temporary~~ Temporarily Approved (2) Changes

~~Temporary~~ changes to procedures of Specification 5.7.1.1 may be made provided: (3)

a. The intent of the existing procedure is not altered;

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator license on the unit affected; and

(continued)

2

Temporarily Approved

5.7.1.3 ~~Temporary~~ Changes (continued)

1

c. The change is documented and reviewed in accordance with Specification 5.5.1 and approved by the [Plant MANAGER ~~Superintendent]~~ or his designee in accordance with approved administrative procedures within 14 days of implementation.

2

5.7.2 Programs and Manuals

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

5.7.2.1 Radiation Protection Program

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

5.7.2.2 Process Control Program (PCP)

13

The PCP shall describe the program ~~contain the current formulas, sampling, analyses, tests, and determinations to be made~~ to ensure that processing and packaging of solid radioactive wastes will be accomplished ~~to~~ ← STET

STET → ~~ensure compliance~~ with 10 CFR 20, 10 CFR 61, and 10 CFR 71; state regulations; burial ground requirements; and other requirements governing the disposal of solid radioactive waste.

Licensee-initiated changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

1. Sufficient information to support the change(s) and appropriate analyses or evaluations justifying the change(s); and

2. A determination that the change(s) maintain the overall conformance of the solidified waste product to the existing requirements of Federal, State, or other applicable regulations.

4

b. Shall be effective after review and acceptance by the PORC ~~[review method of Specification 5.5.1]~~ and the approval of the [Plant ~~Superintendent]~~.

MANAGER 2

1

(continued)

5.7.2 Programs and Manuals (continued)

5.7.2.3 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program; 14

b. The ODCM shall also contain the Radioactive Effluent Controls and Radiological Environmental Monitoring programs required by Specifications 5.7.2.7 AND 5.7.2.8 ~~5.7.2~~, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Semiannual 15 Radioactive Effluent Release Reports required by Specification [[5.9.1.3]] 1 and Specification [[5.9.1.4]]. 1

Licensee-initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s),

2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.106, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

4 b. Shall become effective after review and acceptance by the PORC ~~[review method of Specification 5.5.1]~~ and the approval of the [[Plant MANAGER ~~Superintendent]]~~. 2

c. 1 Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Semiannual Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of 15

(continued)

Chemical and Volume Control,

5.7.2.3 Offsite Dose Calculation Manual (ODCM) (continued)

the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.7.2.4 Primary Coolant Sources Outside Containment

Safety Injection 5

5 This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include the ~~Low~~ Containment ~~Pressure Core~~ Spray, ~~High Pressure Core Spray,~~ Residual Heat Removal, ~~Reactor Core Isolation Cooling, hydrogen recombiner,~~ RCS ~~process~~ sampling, and ~~Standby~~ Gas ~~Treatment~~. The program shall include the following: 5 1

Waste 5

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.7.2.5 In-Plant Radiation Monitoring

This program provides controls to ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

a. Training of personnel;

b. Procedures for monitoring; and

c. Provisions for maintenance of sampling and analysis equipment.

5.7.2.6 Post-Accident Sampling

16

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

a. Training of personnel;

(continued)

5.7.2.6 Post-Accident Sampling (continued)

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

5.7.2.7 Radioactive Effluent Controls Program

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

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 CFR 20, Appendix B, Table II, Column 2;

c. Monitoring, sampling, and analysis of radioactive liquid and
10 gaseous effluents in accordance with 10 CFR 20.~~106~~ and with the methodology and parameters in the ODCM; 1302

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the

(continued)

5.7.2.7 Radioactive Effluent Controls Program (continued)

annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with 10 CFR 20, Appendix B, Table II, Column 1;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

5.7.2.8 Radiological Environmental Monitoring Program

This program is for monitoring the radiation and radionuclides in the environs of the plant. The program shall provide representative measurements of radioactivity in the highest potential exposure pathways and verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall be contained in the ODCM, shall conform to the guidance of 10 CFR 50, Appendix I, and shall include the following:

a. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM;

b. A Land Use Census to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the monitoring program are made if required by the results of this census; and

(continued)

5.7.2.8 Radiological Environmental Monitoring Program (continued)

c. Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

5.7.2.9 Component Cyclic or Transient Limit

This program provides controls to track the FSAR Section ~~[ ]~~ 5.2.1.5 cyclic and transient occurrences to ensure that components are maintained within the design limits. 6

5.7.2.10 Containment Leakage Rate Test Program

15

This program provides controls to ensure that the containment leakage rate tests are performed to maintain containment OPERABILITY. The program shall include the following surveillances required by 10 CFR 50, Appendix J:

a. Type A tests (overall integrated containment leakage rate);

b. Type B tests (local penetration leak rates);

c. Type C tests (containment isolation valve leakage rates);

d. Air lock seal leakage and air lock overall leakage rates;

e. Isolation valve and channel weld pressurization system pressure verifications; and

f. [ ]-inch purge supply and exhaust leakage rates.

5.7.2.11 Pre-stressed Concrete Containment Tendon Surveillance Program

7

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program shall include at least the following:

a. Tendon lift-off to check tendon force;

b. The number of tendons inspected for each tendon group;

(continued)

5.7.2.11 Pre-stressed Concrete Containment Tendon Surveillance Program (continued)

7

c. Tendon wire samples taken to check physical condition, tensile strength, and elongation;

d. Grease samples taken to check chemical properties, physical appearance, and presence of free water;

e. Measurement of grease voids;

f. Visual inspection of end anchorage and containment exterior surface for cracking and grease leakage;

g. Procedures for establishing inspection frequencies;

h. Acceptance criteria;

i. The content and frequency of reporting; and

j. Remedial actions when one or more of the acceptance criteria are not met.

[Key elements to be provided.]

5.7.2.1~~2~~0 Inservice Inspection Program

This program provides controls for inservice inspection of ASME Code Class 1, 2, and 3 components, including applicable supports. The program shall include the following:

a. Provisions that inservice inspection of ASME Code Class 1, 2, and 3 components shall be performed in accordance with ASME Boiler and Pressure Vessel Code and Addenda, Section XI, as required by 10 CFR 50.55a(g), except where relief has been requested from the NRC pursuant to 10 CFR 50.55a(g)(6)(i) and 10 CFR 50.55a(a)(3);

b. The provisions of SR 3.0.2 are applicable to the frequencies for performing inservice inspection activities;

15

c. ~~An inservice inspection program for piping identified in NRC Generic Letter 88-01 in accordance with the NRC staff positions on schedule, methods, personnel, and sample~~

Inspection of each reactor coolant pump flywheel per the recomendations of Regulation Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975; And

(continued)

5.7.2.1~~2~~0 Inservice Inspection Program (continued)

15

~~expansion included in Generic Letter 88-01, or in accordance with alternate measures approved by the NRC staff; and~~

d. Nothing in the ASME Boiler and Pressure Vessel code shall be construed to supersede the requirements of any TS.

5.7.2.1~~3~~1 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following:

a. Provisions that inservice testing of ASME Code Class 1, 2, and 3 pumps, valves, and snubbers shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g), except where relief has been requested from the ~~Commission~~ NRC pursuant to 10 CFR 50.55a(g)(6)(i) and 10 CFR 50.55a(a)(3), or as provided in GL89-04; 15

b. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

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

(continued)

5.7.2.1~~3~~1 Inservice Testing Program (continued)

c. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

d. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.7.2.1~~4~~2 Steam Generator (SG) Tube Surveillance Program (9)

(15) Input Insert From Page 5.0-26A

~~This program provides controls for monitoring SG tube degradation.~~ Each SG shall be demonstrated OPERABLE by its meeting the requirements of Specification 5.7.2.11 and by an approved augmented inservice inspection program that includes at least the following:

a. SG sample selection and inspection;

b. SG tube sample selection and inspection;

c. The establishment of inspection frequencies;

d. Acceptance criteria; and

~~e. The content and frequency of reports.~~

(19) ~~[Key elements to be provided.]~~

5.7.2.1~~5~~3 Secondary Water Chemistry Program (9)

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

(continued)

(15)

INSERT

Each SG shall be demonstrated OPERABLE by performance of an inservice inspection program. The program shall include the following:

a. SG tube sample size selection, sample size expansion, and inspection result classification criteria. Sample selection and testing shall be in accordance with Regulatory Guide 1.83, Revision ~~[ ], date]~~. 1, July 1975.

b. The establishment of SG tube inspection frequency dependent upon inspection result classification. Inspection frequency shall be in accordance with Regulatory Guide 1,83, Revision ~~[ ], date]~~. 1, July 1975.

c. SG tube plugging/repair limits. These limits shall be [40%] of the nominal tube wall thickness consistent with [Regulatory Guide] 1,83, Revision ~~[ ], date]~~. 1, July 1975.

d. Specific definitions and limits for steam generator tube inservice inspection acceptance criteria consistent with Regulatory Guide 1,83, Revision ~~[ ], date]~~ 1, July 1975.

(6) e. The minimum type testing to determine tube integrity.

The content and frequency of written reports shall be in accordance with Specification 5.9.2.

The provisions of SR 3.0.2 are applicable to Steam Generator Tube Surveillance Program inspection frequencies except those established by Category C-3 inspection results.

~~[Key elements to be discussed and provided].~~

5.7.2.1~~5~~3 Secondary Water Chemistry (continued)

c. Identification of process sampling points which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off-control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.7.2.1~~6~~4 Ventilation Filter Testing Program

6 8 Input Insert From Pages 5.0-27A to -27C

~~A program shall be established to implement the following required testing of filters in accordance with [Regulatory Guide 1.52, Revision 2, or ANSI N510-1980].~~

~~a. In place penetration and bypass dioctyl phthalate (DOP) test;~~

~~b. In place penetration and bypass hydrocarbon refrigerant gas test;~~

~~c. Methyl iodide penetration test of a charcoal sample;~~

~~d. Flow rate and pressure drop test; and~~

~~[e. Heater power test.]~~

~~[Key elements to be provided.]~~

1 1

[5.7.2.1~~7~~5 Explosive Gas and Storage Tank Radioactivity Monitoring Program

1 1 quantity 9

This program provides controls for potentially explosive gas mixtures contained in the waste gas holdup system, the ~~quality~~ quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure." The liquid radwaste quantities shall be determined in accordance with Standard Review Plan Section 15.7.3, "Postulated Radioactive Release due to Tank Failures."]

8 1

(continued)

INSERT

5.7.2.14 Ventilation Filter Testing Program

A program shall be established to implement the following required testing of ESF filter ventilation systems at the frequencies specified in, and in accordance with, Regulatory Guide 1.52, Revision 2, and ASME N510-1989.

1. Demonstrate for each of the ESF systems that an inplace test of the HEPA filters shows a penetration and system bypass within acceptance criterion when tested in accordance with RG 1.52, Revision 2 and ASME N510-1989 at the system flowrate specififed below.

ESF VENTILATION SYSTEM	ACCEPTANCE CRITERIA	FLOW RATE
Reactor Building Purge (2 fans together)	Less than 1.00%	14,000 ~~22,949~~ cfm ± 10% minimum
Emergency Gas Treatment	Less than 0.05%	4,000 cfm ± 10%
Auxiliary Building Gas Treatment	Less than 0.05%	9,000 cfm ± 10%
Control Room Emergency	Less than 1.00%	4,000 cfm ± 10%

2. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass within acceptance criterion when tested in accordance with Regulatory Guide 1.52, Revision 2 and ASME N510-1989 at the system flowrate specified below.

ESF VENTILATION SYSTEM	ACCEPTANCE CRITERIA	FLOW RATE
Reactor Building Purge (2 fans together)	Less than 1.00%	14,000 ~~22,949~~ cfm ± 10% minimum
Emergency Gas Treatment	Less than 0.05%	4,000 cfm ± 10%
Auxiliary Building Gas Treatment	Less than 0.05%	9,000 cfm ± 10%
Control Room Emergency	Less than 1.00%	4,000 cfm ± 10%

INSERT

3. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of ≤ 30°C and greater than or equal to the relative humidity specified below.

ESF VENTILATION SYSTEM	METHYL IODIDE PENETRATION	RELATIVE HUMIDITY
Reactor Building Purge	Less than 10%	95%
Emergency Gas Treatment	Less than 0.175%	70%
Auxiliary Building Gas Treatment	Less than 0.175%	70%
Control Room Emergency	Less than 1%	70%

filtration unit

4. Demonstrate for each of the ESF systems that the pressure drop across the ~~combined HEPA filters, the prefilters, and the charcoal adsorbers~~ entire is less than the value specified below when tested in accordance with Regulatory 1.52, Revision 2 and ASME N510-1989 at the system flow rate specified below.

ESF VENTILATION SYSTEM	PRESSURE DROP	FLOW RATE
Reactor Building Purge (2 fans together)	Less than ~~4.7~~ [6.0] inches water	14,000 ~~22,949~~ cfm ± 10% minimum
Emergency Gas Treatment	Less than ~~5.5~~ [8.0] inches water	4,000 cfm ± 10%
Auxiliary Building Gas Treatment	Less than ~~5.2~~ [8.0] inches water	9,000 cfm ± 10%
Control Room Emergency	Less than ~~3.0~~ [8.0] inches water	4,000 cfm ± 10%

INSERT

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

ESF VENTILATION SYSTEM	AMOUNT OF HEAT
Emergency Gas Treatment	20 ± 2.0 kW
Auxiliary Building Gas Treatment	50 ± 5.0 kW

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Ventilation Filter Testing Program test frequencies.

(1) 5.7.2.1~~1~~5 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued) (1)

The program shall include:

(1) (1)

a. The limits for the concentrations of hydrogen and oxygen in the [Waste Gas Holdup System] and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion) (8)

(8)

Input Insert From Page 5.0-28A

b. ~~The limits for the quantity of radioactive gas contained in each gas storage tank and a surveillance program to ensure the limits are maintained; and~~

c. ~~The limits for the quantity of radioactive material contained in unprotected outdoor tanks and a surveillance program to ensure the limits are maintained.~~

~~[Key elements to be provided.]~~ (6) (17)

5.7.2.1~~8~~6 Diesel Fuel Oil Testing Program

in, or prior to transfer to, the 7-day storage tanks

(17) (6)

A diesel fuel oil testing program to implement required testing of ~~both new~~ fuel oil ~~and~~ stored ~~fuel oil~~ shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM standards. The purpose of the program is to establish the following: (6) (17)

the 7-day

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. An API gravity or an absolute specific gravity within limits;

   2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil;

   3. A clear and bright appearance with proper color.

b. Other properties for ASTM 2D fuel oil are within limits within 3~~0~~1 days following sampling and addition to storage tanks.

(18) the 7-day (6) (17)

(continued)

8 6

INSERT

b. A surveillance program to ensure that the quantity of radioactivity contained in [each gas storage tank ~~and fed into the offgas treatment system]~~ is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of [an uncontrolled release of the tank's contents]; and

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the [Liquid Radwaste Treatment System] is less than the amount that would result in concentrations ~~less than~~ equaling or exceeding the limits of 10 CFR Part 20, Appendix B, Table II, Column 2 at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tank's contents.

16

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

(6) in each of the four interconnected tanks which constitute a 7-day storage tank

5.7.2.1~~8~~6 Diesel Fuel Oil Testing Program (continued)

c. Total particulate concentration of the fuel oil is within limits when tested every 31 days in accordance with ASTM D-2276, Method A-2 or A-3.

(15)

5.7.2.17 Fire Protection Program

This program provides controls to ensure that appropriate fire protection measures are maintained to protect the plant from fire and to ensure the capability to achieve and maintain safe shutdown in the event of a fire is maintained.

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 5.7

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

2. Change to reflect WBN specific management titles/terminology.

3. Change to correct vague specification number reference.

4. Change to reflect WBN specific review organization.

5. Change to specify WBN specific systems.

6. Change to specify WBN specific information.

7. WBN does not have pre-stressed concrete containment tendons due to its annular containment/shield building interspace design.

8. Format change to incorporate latest NRC - approved text of program description.

9. Change to correct editorial error in STS.

10. Change reflects most recent 10 CFR 20 references.

(NOTE: Justification Number 11 was not utilized.)

12. Temporarily approved changes are not approved by the Plant Manager prior to implementation.

13. At WBN, the PCP does not contain the detail specified by the STS.

14. Editorial change to more accurately specify the exact Specification 5.7.2 steps requiring the Radioactive Effluent Controls and Radiological Environmental Monitoring programs.

15. Change reflects comments made by the industry to the NRC at the RSTS Proof and Review meeting held in Irvine, CA in July, 1992.

16. Change to enhance editorial clarity.

17. At WBN, fuel oil is tested prior to transfer to the 7-day EDG storage tanks, not prior to fuel truck unloading of new fuel.

18. Change to correct inconsistency with time limit described in Specification 3.8.3; SR 3.8.3.3 Bases.

19. Due to ongoing discussions between the NRC and the industry concerning deletion of the requirement to provide details of the Steam Generator Tube Surveillance Program, WBN elects to delete this bracketed sentence at this time.

5.0 ADMINISTRATIVE CONTROLS

5.8 Safety Function Determination Program (SFDP)

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

5.8.2 The SFDP shall contain the following:

a. Provisions for cross-train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected.

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists.

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities.

d. Other appropriate limitations and remedial or compensatory actions.

5.8.3 A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or (CASE A) 1

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or (CASE B)

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable. (CASE C)

(continued)

5.8.3 (continued)

Generic Example:

Train A		Train B	
System i		System i	←Case c
↓		↓	
System ii	←(Support System Inoperable)	System ii	
↓		↓	
System iii		System iii	←Case a
↓		↓	
System iv		System iv	←Case b

5.8.4 The Safety Function Determination Program identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 5.8

1. Change reflects comments made by the industry to the NRC at the RSTS Proof and Review meeting held in Irvine, CA in July, 1992.

5.0 ADMINISTRATIVE CONTROLS

5.9 Reporting Requirements

5.9.1 Routine Reports

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.9.1.1 Startup Report

A summary report of plant startup and power escalation testing shall be submitted following:

a. Receipt of an Operating License;

b. Amendment to the license involving a planned increase in power level;

c. Installation of fuel that has a different design or has been manufactured by a different fuel supplier; and

d. Modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit. (2)

(1) POWER ASCENSION

The initial Startup Report shall address each of the ~~startup~~ tests identified in FSAR, Chapter [14], and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report. Subsequent Startup Reports shall address startup tests that are necessary to demonstrate the acceptability of changes and modifications.

Power Ascension (2)

Startup Reports shall be submitted within 90 days following completion of the ~~Startup~~ Test Program; 90 days following resumption or commencement of commercial power operation; or 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of ~~Startup~~ Test Program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.

(continued)

5.9.1 Routine Reports (continued)

5.9.1.2 Annual Reports

1

1 [------------------------------NOTE------------------------------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.
--]

the year in which 9

Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted by March 31 of each year. [The initial report shall be submitted by March 31 of the year following initial criticality.]

1

Reports required on an annual basis include:

9

a. Occupational Radiation Exposure Report

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special

7 maintenance ~~[describe maintenance]~~, waste processing, and refueling). This tabulation supplements the requirements of

13 10 CFR 20.~~407~~. The dose assignments to various duty functions may be estimated based on pocket dosimeter,

2206 thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole-body dose received from external sources should be assigned to specific major work functions; and.

6 ~~[b. Any other unit unique reports required on an annual basis.]~~

9 was achieved. If six months have not passed since initial criticality, the initial report will be submitted one year later

(continued)

5.9.1 Routine Reports (continued)

5.9.1.3 Annual Radiological Environmental Operating Report 1

1

-----------------------------NOTE-----------------------------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. [The report shall identify the thermoluminescent dosimeter (TLD) results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result.] In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

1

5.9.1.4 8 ~~Semiannual~~ Radioactive Effluent Release Report 1

1

-----------------------------NOTE-----------------------------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

(continued)

The initial report will be submitted by JANUARY 1 OR July 1 of the year following at least six months of operation AFTER initial criticality.

8

IN ACCORDANCE with 10 CFR 50.36a.

9

5.9.1.4 ~~Semiannual~~ Radioactive Effluent Release Report (continued)

8

The ~~Semiannual~~ Radioactive Effluent Release Report covering the operation of the unit during the previous 6 months of operation shall be submitted ~~within 60 days after January 1 and July 1 of each year.~~ 8 The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program (PCP) and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.I.

5.9.1.5 Monthly Operating Reports

pressurizer 8 1

Routine reports of operating statistics and shutdown experience[, including documentation of all challenges to the power-operated relief valves (PORVs) or pressurizer safety valves,] shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

1

5.9.1.6 CORE OPERATING LIMITS REPORT (COLR)

10 changing

a. Core operating limits shall be established prior to each reload cycle, or prior to changing any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

Input Insert From Page 5.0-35A 2

~~1. [The individual specifications that address core operating limits must be referenced here.]~~

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

Input Insert From Page 5.0-35B 2

~~1. [Identify the Topical Report(s) by number, title, date, and NRC staff approval document, or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.]~~

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

(continued)

②

INSERT

LCO 3.1.4	Moderator Temperature Coefficient
LCO 3.1.6	Shutdown Bank Insertion Limit
LCO 3.1.7	Control Bank Insertion Limits
LCO 3.2.1	Heat Flux Hot Channel Factor
LCO 3.2.2	Nuclear Enthalpy Rise Hot Channel Factor
LCO 3.2.3	Axial Flux Difference
LCO 3.9.1	Boron Concentration

INSERT

(2)

1. WCAP-9272-P-A, WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary).
   (Methodology for Specifications 3.1.4 - Moderator Temperature Coefficient, 3.1.6 - Shutdown Bank Insertion Limit, 3.1.7 - Control Bank Insertion Limits, 3.2.1 - Heat Flux Hot Channel Factor, 3.2.2 - Nuclear Enthalphy Rise Hot Channel Factor, and 3.9.1 - Boron Concentration.

2. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary).
   (Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor).

3. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F(Q) SURVEILLANCE TECHNICAL SPECIFICATION", June 1983 (W Proprietary).
   (Methodology for Specifications 3.2.1 - Heat Flux Hot Channel Factor (W(Z) Surveillance Requirements For F(Q) Methodology) and 3.2.3 - Axial Flux Difference (Relaxed Axial Offset Control).)

WCAP-7924-A, "WESTINGHOUSE BASIS FOR HEATUP AND COOLDOWN LIMIT CURVES", April 1975, or Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials", May 1988.

5.9.1.6 CORE OPERATING LIMITS REPORT (COLR) (continued)

d. The COLR, including any mid-cycle revisions or supplements, shall be provided ~~upon~~ issuance for each reload cycle to the NRC. (12) within 30 days of

5.9.1.7 RCS PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

(4) And

RCS pressure ~~the~~ temperature limits, including heatup and cooldown rates, criticality, and hydrostatic and leak test limits, shall be established and documented in the PTLR. (3) ~~[The individual Specifications that address the reactor vessel pressure and temperature limits and the heatup and cooldown rates may be referenced.]~~ The analytical methods used to determine the pressure and temperature limits including the heatup and cooldown rates shall be those previously reviewed and approved by the NRC in (2) ~~[Topical Report(s), number, title, date, and NRC staff approval document, or staff safety evaluation report for a plant specific methodology by NRC letter and date].~~ The reactor vessel pressure and temperature limits, including those for heatup and cooldown rates, shall be determined so that all applicable limits (e.g., (4) heatup ~~and~~ limits, cooldown limits, and inservice leak and hydrostatic testing limits) of the analysis are met. The PTLR, including revisions or supplements thereto, shall be provided ~~upon~~ issuance~~. for each reactor vessel fluency period~~.

(12) within 30 days of

5.9.2 Special Reports

(5) Special Reports may be required covering inspection, test, and maintenance activities. These special reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.

Special Reports shall be submitted in accordance with 10 CFR 50.4 within the time period specified for each report.

The following Special Reports shall be submitted:

a. (8) In the event an ECCS is actuated and injects water into the RCS in MODE ~~1 or 2~~ 1,2 or 3, a Special Report shall be prepared and submitted within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70;

(continued)

5.9.2 Special Reports (continued) (11)

(1) g

b. [If an individual emergency diesel generator (EDG) ~~25~~ 20 (11)
experiences ~~1~~ three or more valid failures in the last 20 demands, these failures and any non-valid failures experienced by that EDG in that time period shall be reported within 30 days. Reports on EDG failures shall include the information recommended in Regulatory Guide 1.9, Revision 3, Regulatory Position C.5, or existing Regulatory Guide 1.108 reporting requirement; and] (1) g

Post Accident Monitoring Instrumentation,

(2) (2) D or H

c. (8) When a Special Report is required by Condition [redacted] of LCO 3.3.~~[X]~~ 3, a report shall be submitted within the following 14 days. ~~from the time the action is required.~~ The report shall outline the preplanned alternate method of monitoring the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the function to OPERABLE status. (8)

(8)

d. Following each inservice inspection of steam generator tubes, in accordance with the Steam Generator Tube Surveillance Program, the number of tubes plugged and tubes sleeved in each steam generator shall be reported to the NRC within 15 days.

The complete results of the steam generator tube inservice inspection shall be submitted to the NRC within 12 months following the completion of the inspection. The report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged and tubes sleeved.

Results of steam generator tube inspections which fall into Category C-3 shall be reported to the NRC prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 5.9

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

2. Change to reflect WBN specific information.

3. Deleted bracketed text since WBN options not to reference individual specifications that address reactor vessel pressure and temperature limits or RCS heatup and cooldown rates.

4. Change to correct editorial error in STS.

5. Deleted unnecessary bracketed text that only provides guidance on what information should be included under Special Reports.

6. Deleted bracketed text since WBN has no other unique reports required on an annual basis.

7. Deleted bracketed text since special maintenance could encompass any non-routine maintenance activity.

8. Change reflects comments made by the industry to the NRC at the RSTS Proof and Review meeting held in Irvine, CA in July, 1992.

9. Change reflects WBN unique position of an unlicensed plant requesting relief from initial reporting requirements.

10. Change to enhance editorial clarity.

11. Change reflects latest draft Reg. Guide 1.9 failure criterion.

12. Change provides a defined time limit which to submit the COLR and PTLR to the NRC. WBN believes this time limit removes any questions concerning when these reports must be submitted. Additionally, the last phrase was deleted because the PTLR is prepared to cover multiple fluency periods and is not revised each cycle as the wording implies.

13. Change reflects new 10 CFR 20 paragraph that will become effective in January, 1994.

5.0 ADMINISTRATIVE CONTROLS

5.10 Record Retention

5.10.1 The following records shall be retained for at least 3 years:

a. All License Event Reports required by 10 CFR 50.73; AND

7 b. Records of changes made to the procedures required by Specification 5.7.1.1; and

b ~~c~~. Records of radioactive shipments.

5.10.2 The following records shall be retained for at least 5 years:

a. Official Records and logs of unit operation covering time intervals at each power level; 1

b. Official Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety; 4

c. Records of surveillance activities, inspections, and calibrations required by the Technical Specifications (TS) 2 [and the Fire Protection Program];

d. Records of sealed source and fission detector leak tests and results; and

e. Records of annual physical inventory of all sealed source material of record.

5.10.3 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the FSAR;

b. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories;

c. Records of radiation exposure for all individuals entering radiation control areas;

(continued)

1

Technical Requirement (TR) 3.7.3, "Snubbers",

5.10.3 (continued)

d. Records of gaseous and liquid radioactive material released to the environs;

e. Records of transient or operational cycles for those unit components identified in [FSAR, Section X]; 1 5.2.1.5 2

f. Records of reactor tests and experiments; 4

g. Records of training and qualification for current members of the unit staff;

h. Records of inservice inspections performed pursuant to the TS;

1 | PLAN 2

i. Records of quality assurance activities required by the Operational Quality Assurance (QA) ~~Manual~~ [not listed in Specification 5.10.1 and which are classified as permanent records by applicable regulations, codes, and standards]; 2

AND FOR 6

j. 6 Records of reviews performed for changes made to procedures, equipment, ~~or~~ reviews of tests and experiments pursuant to 10 CFR 50.59;

k. Records of the reviews and audits required by Specification 5.5.1 and Specification 5.5.2;

l. Records of the service lives of all hydraulic and mechanical snubbers required by ~~[document where snubber requirements relocated to],~~ including the date at which the service life commences, and associated installation and maintenance records; 1

steam generator 5 water

2 [m. Records of ~~secondary water~~ sampling and water quality;] 2

n. Records of analyses required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analysis at a later date.

These records should include procedures effective at specified times and QA records showing that these procedures were followed;

o. Records of reviews performed for changes made to the Offsite Dose Calculation Manual and the Process Control Program; AND

(continued)

5.10.3 (continued)

3 ~~[p. Records of pre-stressed concrete containment tendon surveillances;] and~~

p. ~~[q.~~ Records of steam generator tube surveillances.~~]~~ e 2

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 5.10

1. Change to reflect WBN specific information.

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

3. WBN does not have pre-stressed concrete containment tendons due to its annular containment/shield building interspace design.

4. Change reflects comments made by the industry to the NRC at the RSTS Proof and Review meeting held in Irvine, CA in July, 1992.

5. WBN believes that "secondary water" is too broad. Steam generator records are applicable to pressure boundary integrity which WBN believes are adequate.

6. Change to enhance editorial clarity.

7. At WBN, procedure changes are controlled by Specifications 5.10.3j and k, which have a longer record retention requirement than Specification 5.10.1.b.

5.0 ADMINISTRATIVE CONTROLS

1 [5.11 High Radiation Area] 1

1

1611 5

5.11.1 Pursuant to 10 CFR 20, paragraph ~~20.203(c)(5)~~ 20.1601(c), in lieu of the requirements of 10 CFR 20.~~203(c)~~ 1601, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., ~~[Health Physics~~ Radiological Control (2) Technicians]) or personnel continuously escorted (1) by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the (1) [~~Radiation Protection~~ (2) Radiological Controls Manager] (1) in the RWP.

5.11.2 In addition to the requirements of Specification 5.11.1, areas with radiation levels ≥ 1000 mrem/hr shall be provided with locked or (3) be continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift ~~Foreman~~ (2) Operations Supervisor on duty or ~~health physics~~ Radiological controls (2) supervision. Doors

(continued)

1

1

1

5.11.2 (continued)

shall remain locked except during periods of access by personnel under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

5.11.3 For individual high radiation areas with radiation levels of > 1000 mrem/hr, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted. ~~and a flashing light shall be activated as a warning device.~~ However, barricades will not be left in the reactor containment during power operation.

4

JUSTIFICATIONS FOR DEVIATIONS FROM NRC STS 5.11

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

2. Change to reflect WBN specific terminology.

3. Change to enhance editorial clarity.

4. Change to reflect methodolgy utilized at WBN for warning personnel of large-area, high radiation areas.

5. Change to reflect new 10CFR20.

ENCLOSURE 5

SAMPLE CORE OPERATING LIMITS REPORT

WATTS BAR NUCLEAR PLANT

CORE OPERATING LIMITS REPORT

UNIT 1

SAMPLE DRAFT

AUGUST 1992

SAMPLE COLR FOR WATTS BAR UNIT 1 CYCLE 1

1.0 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for Watts Bar Unit 1 Cycle 1 has been prepared in accordance with the requirements of Technical Specification (TS) 5.8.1.6.

The TSs affected by this report are listed below:

3.1.4 Moderator Temperature Coefficient
3.1.6 Shutdown Rod Insertion Limit
3.1.7 Control Rod Insertion Limits
3.2.1 Heat Flux Hot Channel Factor
3.2.2 Nuclear Enthalpy Hot Channel Factor
3.2.3 Axial Flux Difference
3.9.1 Boron Concentration

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. These limits have been developed using the NRC-approved methodologies specified in TS 5.8.1.6.

2.1 Moderator Temperature Coefficient (LCO 3.1.4)

2.1.1 The Moderator Temperature Coefficient (MTC) limits are:

The BOL/ARO/HZP-MTC shall be less positive than 0 $\Delta k/k/°F$.

The EOL/ARO/RTP-MTC shall be less negative than -4.0×10^{-4} $\Delta k/k/°F$.

2.1.2 The 300 ppm surveillance limit is:

The measured 300 ppm/ARO/RTP-MTC should be less negative than or equal to 3.1×10^{-4} $\Delta k/k/°F$.

2.1.3 The 60 PPM surveillance limit is:

The measured 60 ppm/ARO/RTP-MTC should be less negative than or equal to [ ] $\Delta k/k/°F$.

where: BOL stands for Beginning of Cycle Life
ARO stands for ALL Rods Out
HZP stands for Hot Zero THERMAL POWER
EOL stands for End of Cycle Life
RTP stands for RATED THERMAL POWER

2.2 Shutdown Rod Insertion Limit (LCO 3.1.6)

2.2.1 The shutdown rods shall be withdrawn to a position as defined below:

Cycle Burnup (MWD/MTU)	Steps Withdrawn
$\leq$ 2,000	$\geq$ 226 TO $\leq$ 231
> 2,000 TO < 14,000	$\geq$ 222 TO $\leq$ 231
$\geq$ 14,000	$\geq$ 226 TO $\leq$ 231

2.3 Control Rod Insertion Limit (LCO 3.1.7)

2.3.1 The control rod banks shall be limited in physical insertion as shown in Figure 1.

2.4 Heat Flux Hot Channel Factor - $F_Q(Z)$ (LCO 3.2.1)

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

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

$$\text{where } P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

2.4.1 CFQ = 2.303

2.4.2 K(Z) is provided in Figure 2.

2.4.3 $F_Q^w(Z) = F_Q^c(Z) + W(Z)$

where W(Z) values are obtained for Figures 4 through 7.

2.5 <u>Nuclear Enthalpy Rise Hot Channel Factor</u> - $F^N_{\Delta H}$ (LCO 3.2.2)

$$F^N_{\Delta H} \leq F^{RTP}_{\Delta H} * (1 + PF_{\Delta H} * [1 - P])$$

$$\text{where } P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

$$F^{RTP}_{\Delta H} = 1.49$$

$$PF_{\Delta H} = 0.2$$

2.6 <u>Axial flux difference</u> - (LCO 3.2.3)

2.6.1 The Axial Flux Difference (AFD) Limits are provided in Figure 3.

2.7 <u>Refueling Boron Concentration</u> - (LCO 3.9.1)

2.7.1 The refueling boron concentration shall be $\geq$ [2000] ppm.

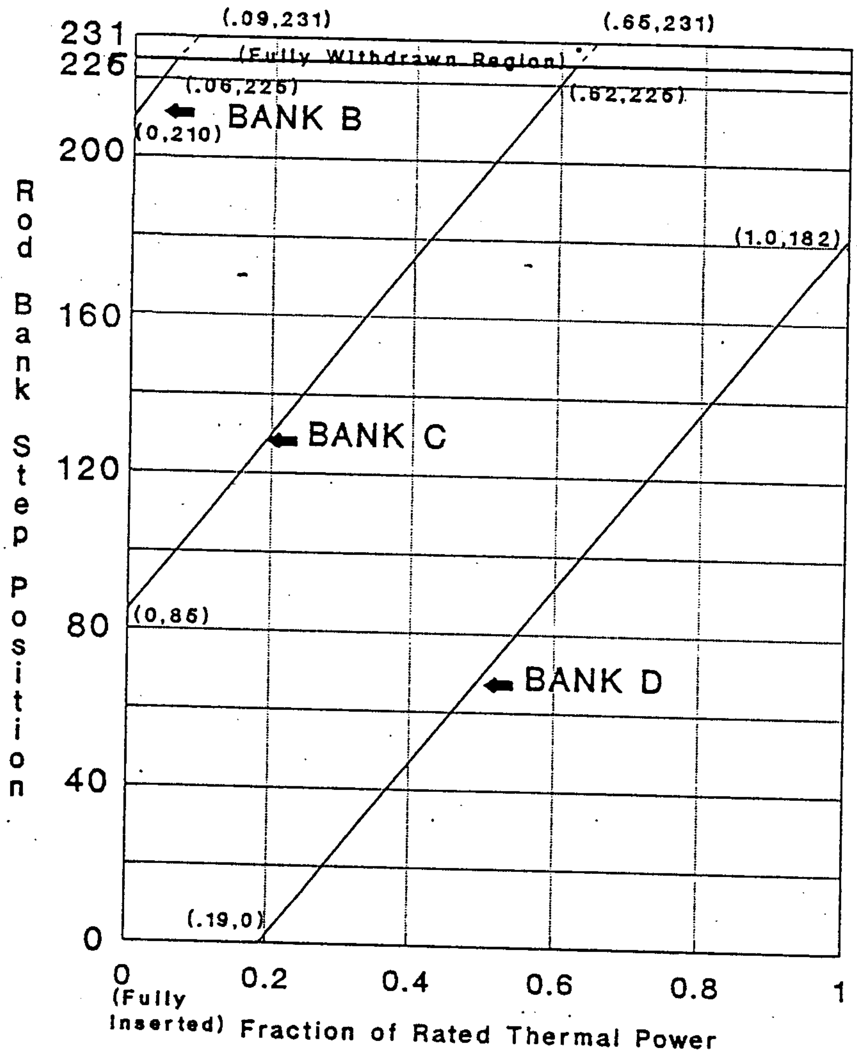


FIGURE 1: Rod Bank Insertion Limits Versus Thermal Power
Four Loop Operation

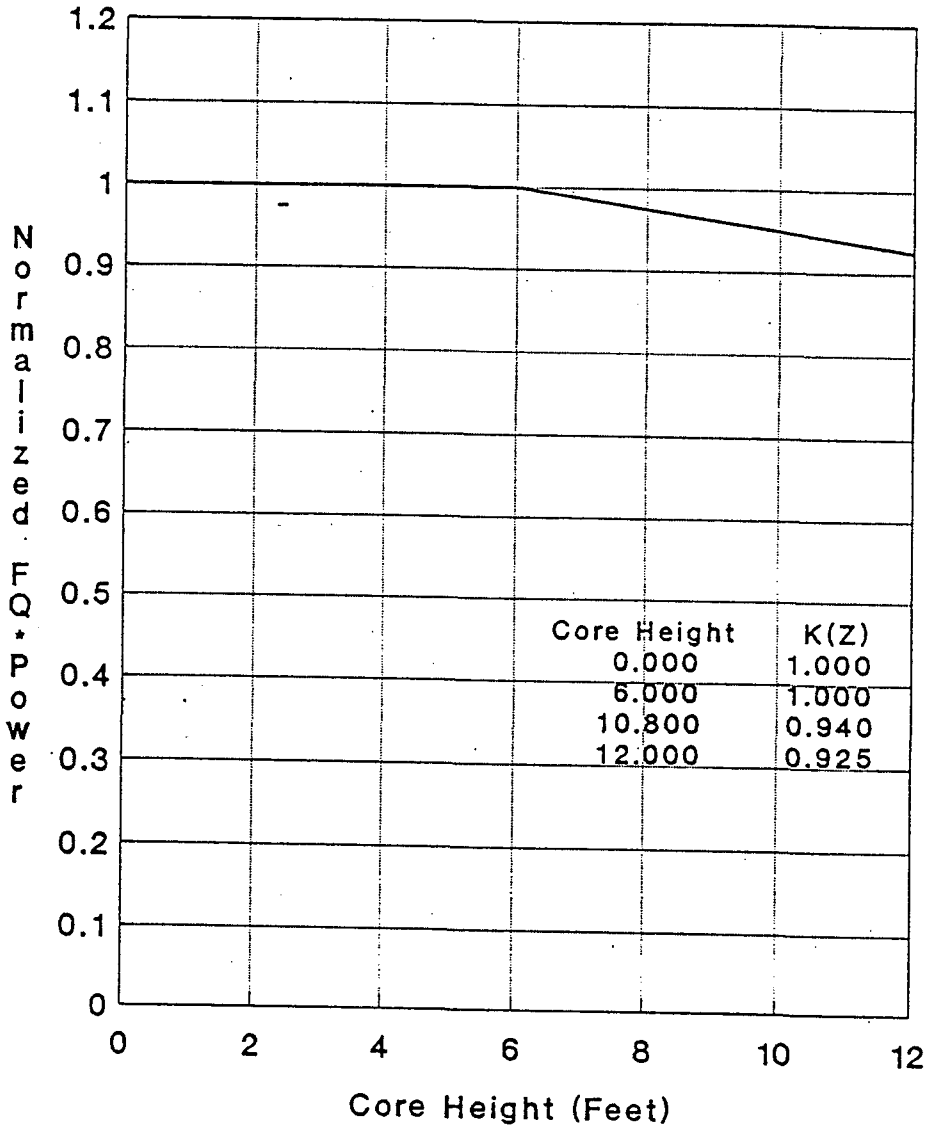


FIGURE 2: K(Z) - Normalized $F_Q(Z)$ as a Function of Core Height

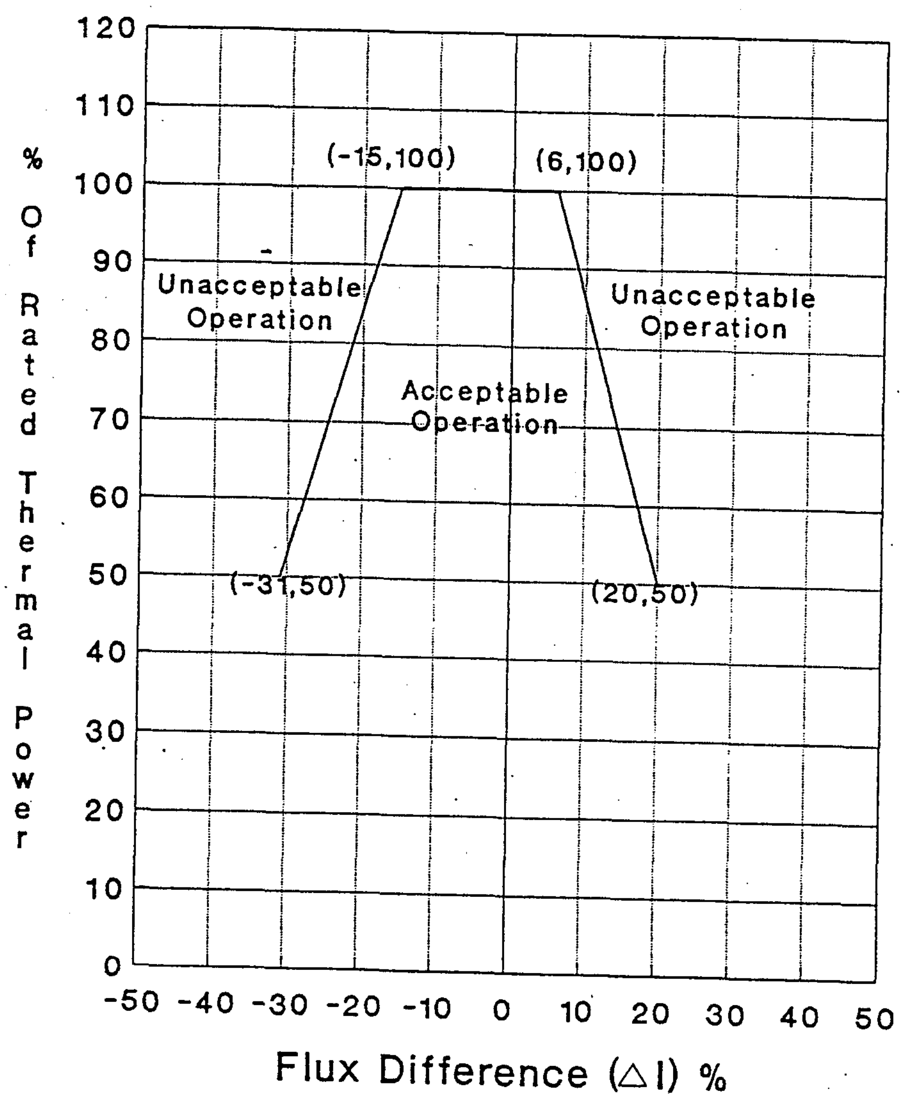
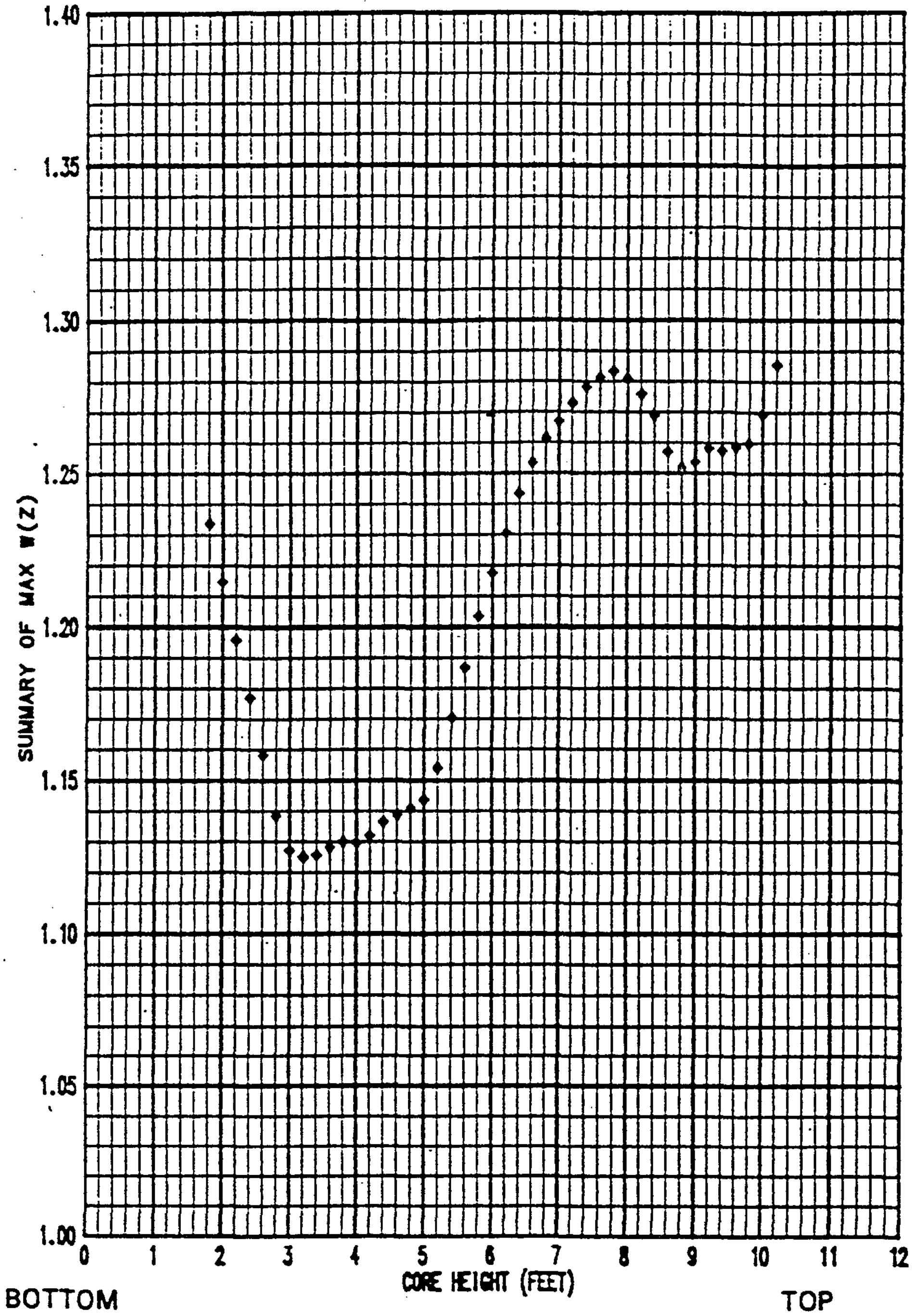


FIGURE 3: AXIAL FLUX DIFFERENCE Acceptable Operation Limits as a Function of RATED THERMAL POWER (RAOC)

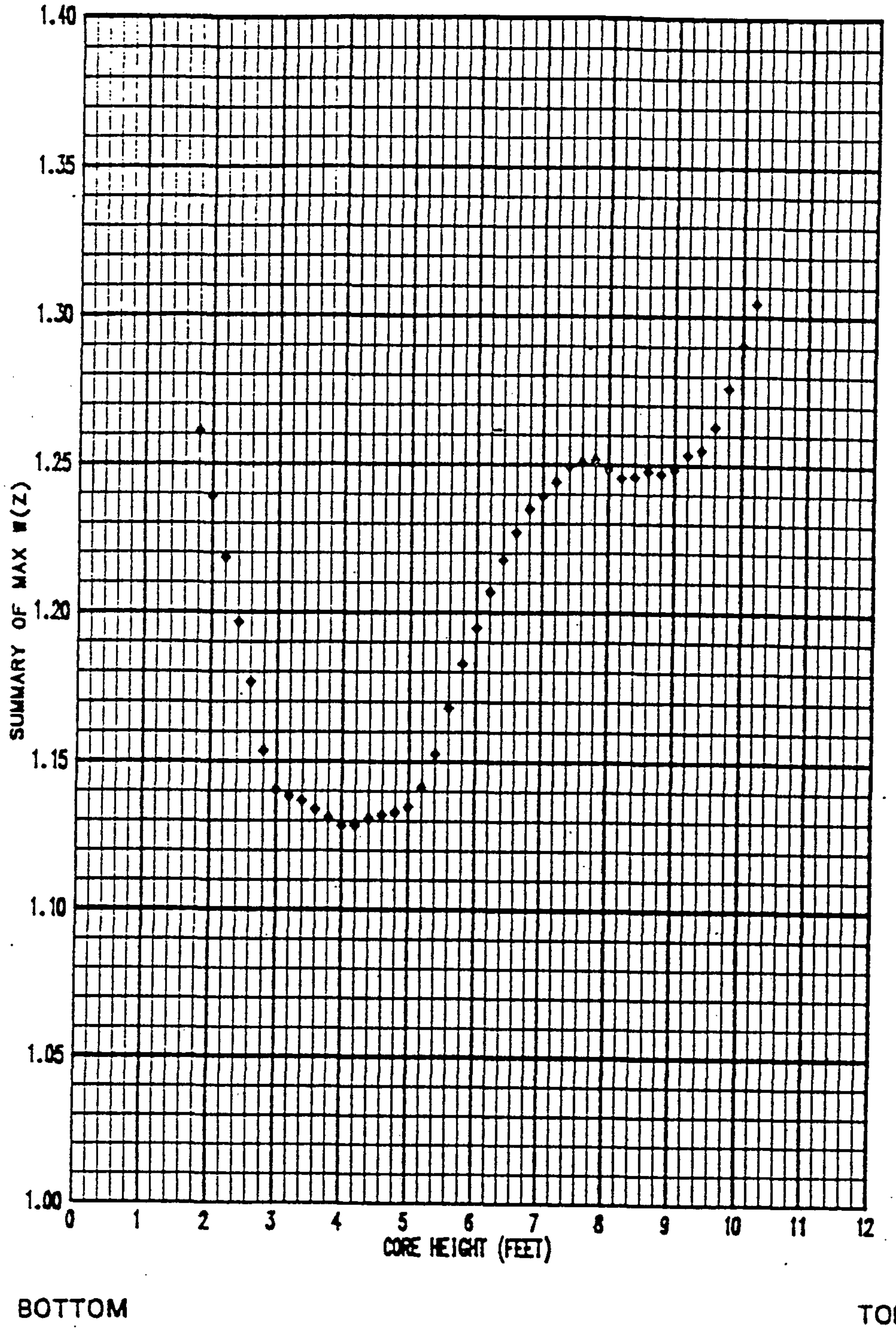
SAMPLE COLR FOR WATTS BAR UNIT 1 CYCLE 1



Height (Feet)	MAX W(Z)
0.000	1.0000
0.200	1.0000
0.400	1.0000
0.600	1.0000
0.800	1.0000
1.000	1.0000
1.200	1.0000
1.400	1.0000
1.600	1.0000
1.800	1.2337
2.000	1.2148
2.200	1.1958
2.400	1.1771
2.600	1.1584
2.800	1.1385
3.000	1.1272
3.200	1.1250
3.400	1.1257
3.600	1.1282
3.800	1.1301
4.000	1.1297
4.200	1.1320
4.400	1.1365
4.600	1.1388
4.800	1.1411
5.000	1.1436
5.200	1.1540
5.400	1.1704
5.600	1.1868
5.800	1.2034
6.000	1.2177
6.200	1.2306
6.400	1.2434
6.600	1.2533
6.800	1.2614
7.000	1.2669
7.200	1.2729
7.400	1.2782
7.600	1.2814
7.800	1.2833
8.000	1.2809
8.200	1.2759
8.400	1.2688
8.600	1.2570
8.800	1.2518
9.000	1.2536
9.200	1.2581
9.400	1.2574
9.600	1.2587
9.800	1.2597
10.000	1.2693
10.200	1.2855
10.400	1.0000
10.600	1.0000
10.800	1.0000
11.000	1.0000
11.200	1.0000
11.400	1.0000
11.600	1.0000
11.800	1.0000
12.000	1.0000

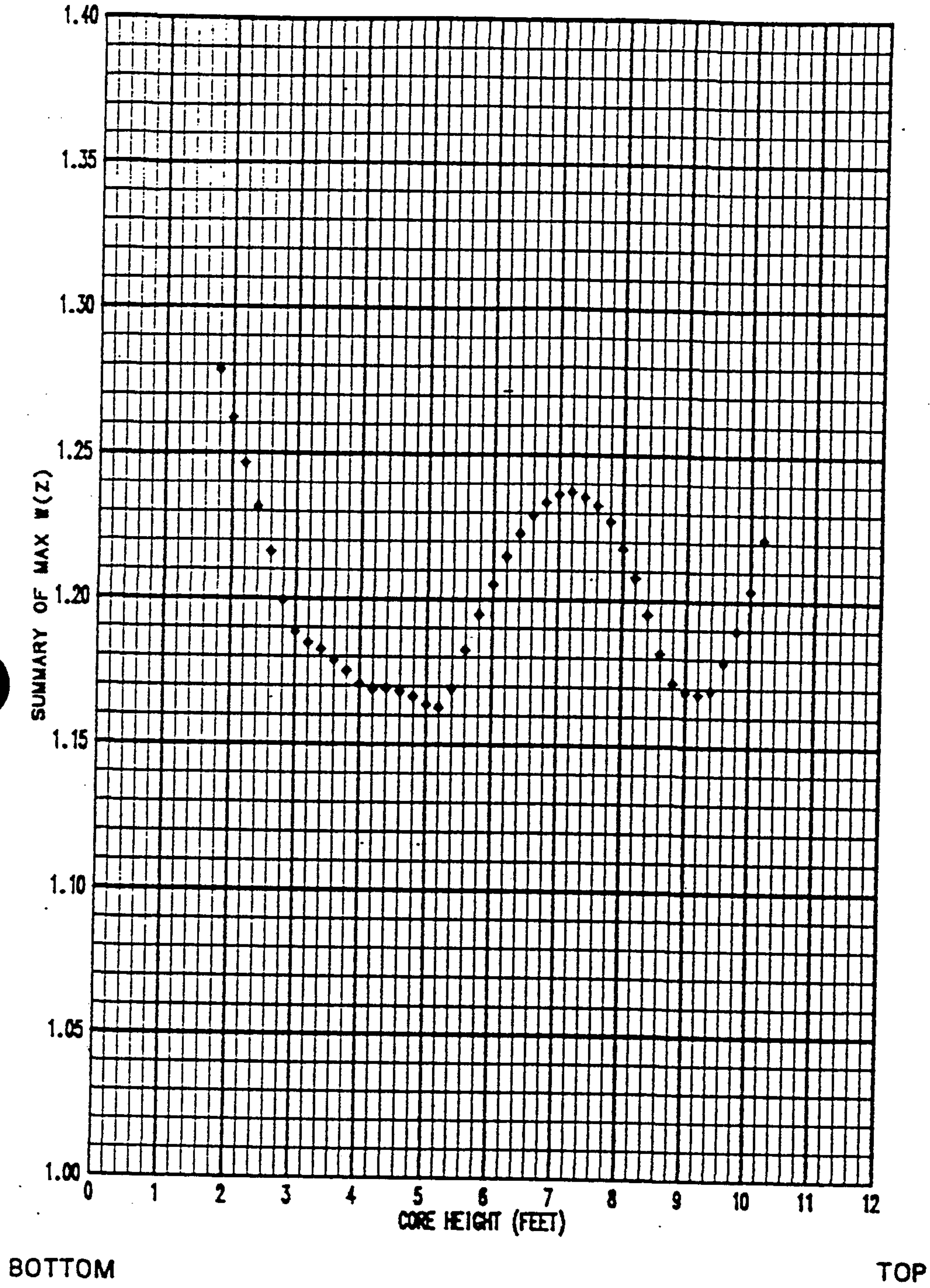
FIGURE 4: RAOC SUMMARY OF MAX W(Z) AT 150 MWD/MTU

SAMPLE COLR FOR WATTS BAR UNIT 1 CYCLE 1



Height (Feet)	MAX W(Z)
0.000	1.0000
0.200	1.0000
0.400	1.0000
0.600	1.0000
0.800	1.0000
1.000	1.0000
1.200	1.0000
1.400	1.0000
1.600	1.0000
1.800	1.2609
2.000	1.2391
2.200	1.2184
2.400	1.1967
2.600	1.1765
2.800	1.1535
3.000	1.1405
3.200	1.1384
3.400	1.1369
3.600	1.1339
3.800	1.1311
4.000	1.1285
4.200	1.1288
4.400	1.1309
4.600	1.1320
4.800	1.1330
5.000	1.1350
5.200	1.1414
5.400	1.1525
5.600	1.1680
5.800	1.1826
6.000	1.1947
6.200	1.2069
6.400	1.2177
6.600	1.2270
6.800	1.2349
7.000	1.2392
7.200	1.2441
7.400	1.2493
7.600	1.2509
7.800	1.2518
8.000	1.2484
8.200	1.2455
8.400	1.2459
8.600	1.2479
8.800	1.2469
9.000	1.2485
9.200	1.2534
9.400	1.2551
9.600	1.2630
9.800	1.2760
10.000	1.2906
10.200	1.3045
10.400	1.0000
10.600	1.0000
10.800	1.0000
11.000	1.0000
11.200	1.0000
11.400	1.0000
11.600	1.0000
11.800	1.0000
12.000	1.0000

FIGURE 5: RAOC SUMMARY OF MAX W(Z) AT 4,000 MWD/MTU



Height (Feet)	MAX W(Z)
0.000	1.0000
0.200	1.0000
0.400	1.0000
0.600	1.0000
0.800	1.0000
1.000	1.0000
1.200	1.0000
1.400	1.0000
1.600	1.0000
1.800	1.2785
2.000	1.2619
2.200	1.2465
2.400	1.2314
2.600	1.2160
2.800	1.1993
3.000	1.1885
3.200	1.1847
3.400	1.1824
3.600	1.1786
3.800	1.1750
4.000	1.1706
4.200	1.1689
4.400	1.1692
4.600	1.1683
4.800	1.1664
5.000	1.1636
5.200	1.1626
5.400	1.1691
5.600	1.1824
5.800	1.1948
6.000	1.2052
6.200	1.2147
6.400	1.2226
6.600	1.2290
6.800	1.2336
7.000	1.2363
7.200	1.2373
7.400	1.2354
7.600	1.2325
7.800	1.2271
8.000	1.2178
8.200	1.2078
8.400	1.1952
8.600	1.1817
8.800	1.1716
9.000	1.1687
9.200	1.1678
9.400	1.1693
9.600	1.1789
9.800	1.1898
10.000	1.2036
10.200	1.2213
10.400	1.0000
10.600	1.0000
10.800	1.0000
11.000	1.0000
11.200	1.0000
11.400	1.0000
11.600	1.0000
11.800	1.0000
12.000	1.0000

FIGURE 6: RAOC SUMMARY OF MAX W(Z) AT 7,000 MWD/MTU

4.0 Reactor Vessel Material Surveillance Program

The reactor vessel material irradiation surveillance specimens shall be removed and examined to determine changes in material properties. The removal schedule is provided in Table 4.0-1. The results of these examinations shall be used to update Figures 2.1-1, 2.1-2, and 3.1-1.

The pressure vessel steel surveillance program (Ref. 4) is in compliance with Appendix H to 10 CFR 50, entitled "Reactor Vessel Material Surveillance Program Requirements". The material test requirements and the acceptance standard utilize the reference nil-ductility temperature, RT_{NDT}, which is determined in accordance with ASTM E208. The empirical relationship between RT_{NDT} and the fracture toughness of the reactor vessel steel is developed in accordance with Appendix G, "Protection Against Non-Ductile Failure", to section III of the ASME Boiler and Pressure Vessel Code. The surveillance capsule removal schedule meets the requirements of ASTM E185-73. The removal schedule is provided in Table 4.0-1.

REFERENCES

1. TVA Calculation WBN-MTB-027, "Pressure-Temperature Limits Based on Regulatory Guide 1.99 Revision 2 for Submittal to NRC," (B46890501559).

2. Westinghouse letter to TVA, WAT-D-8376, "Reactor Coolant System Accelerated Cooldown", November 5, 1990.

3. WCAP-13461, "Summary Report Process Protection System EAGLE 21™ Upgrade Watts Bar."

4. Davidson, J.A., "Tennessee Valley Authority Watts Bar Unit No. 1 Reactor Vessel Radiation Surveillance Program", WCAP-9298, July, 1978.

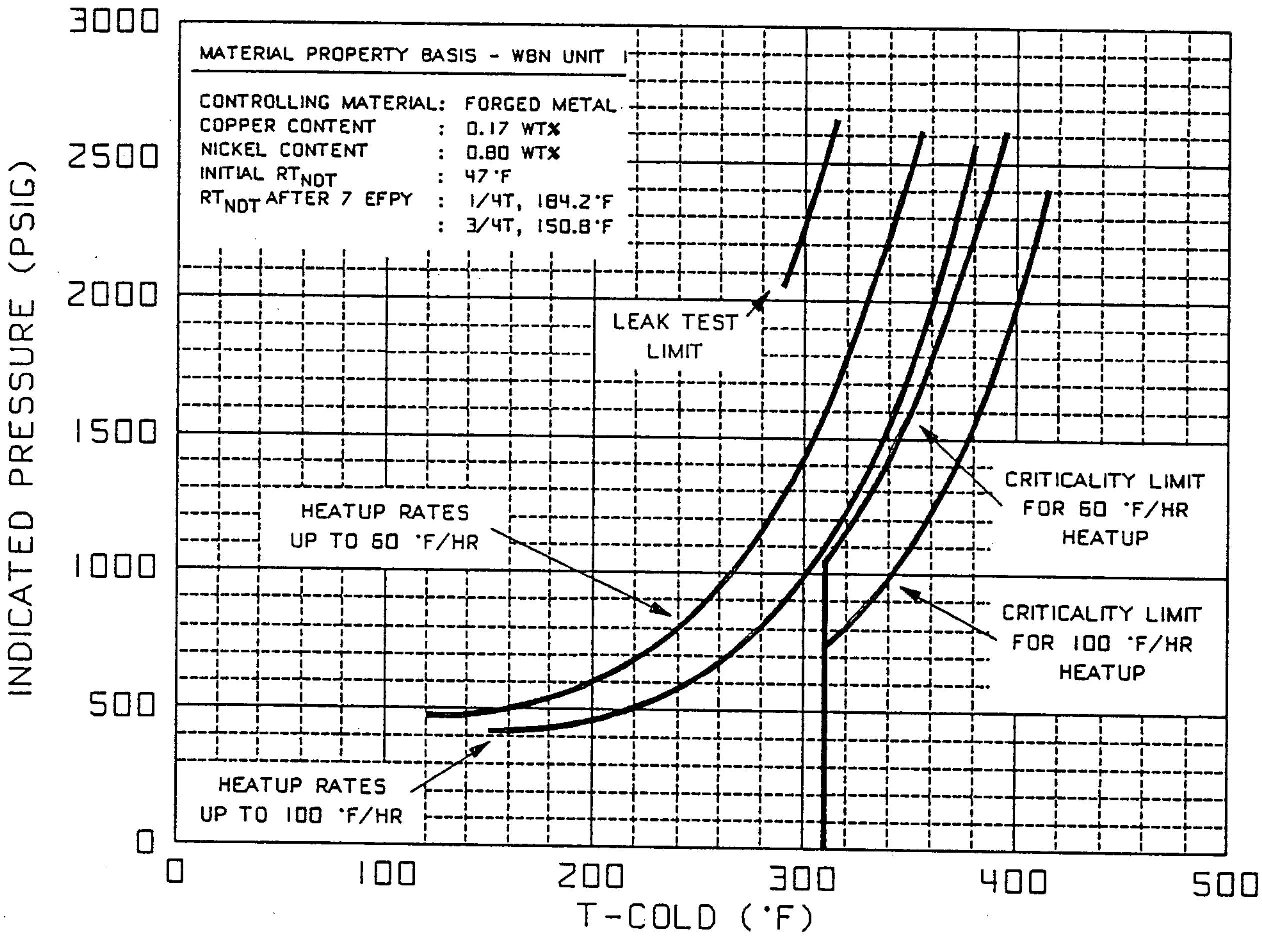


Figure 2.1-1

Reactor Coolant System
Heatup Limitations
Applicable Up to 7 EFPY

(Plotted data (Ref. 1) provided on next pages)

UNIT 1 HEATUP LIMITS
(Data (Ref. 1) plotted on Fig 2.1-1)

	INDICATED PRESSURE, PSIG				
RCS FLUID TEMPERATURE (DEG °F)	60 (°F/HR)	100 (°F/HR)	LEAK TEST LIMITS	CRITICALITY 60 (°F/HR)	CRITICALITY 100 (°F/HR)
100					
105					
110					
115					
120	472				
125	472				
130	472				
135	474				
140	478				
145	482				
150	487				
155	494	419			
160	501	420			
165	509	422			
170	519	424			
175	530	428			
180	541	433			
185	554	438			
190	568	445			
195	584	453			
200	601	461			
205	619	471			
210	638	483			
215	660	495			
220	683	509			
225	708	524			
230	735	540			

UNIT 1 HEATUP LIMITS
(Data (Ref. 1) plotted on Fig 2.1-1)

INDICATED PRESSURE, PSIG					
RCS FLUID TEMPERATURE (DEG °F)	60 (°F/HR)	100 (°F/HR)	LEAK TEST LIMITS	CRITICALITY 60 (°F/HR)	CRITICALITY 100 (°F/HR)
235	764	558			
240	795	578			
245	829	599			
250	866	622			
255	906	647			
260	948	674			
265	994	704			
270	1043	736			
275	1096	770			
280	1153	807			
285	1214	847			
290	1280	890	2050		
295	1350	936	2155		
300	1426	986	2268		
305	1508	1040	2390		
310	1596	1098	2521	1043	734
315	1690	1161	2661	1096	770
320	1792	1228		1153	807
325	1901	1300		1214	847
330	2018	1378		1280	890
335	2129	1462		1350	936
340	2239	1552		1426	986
345	2357	1649		1516	1040
350	2484	1754		1596	1098
355	2620	1866		1690	1161
360	2767	1987		1792	1228
365		2116		1901	1300

UNIT 1 HEATUP LIMITS
(Data (Ref. 1) plotted on Fig 2.1-1)

	INDICATED PRESSURE, PSIG				
RCS FLUID TEMPERATURE (DEG °F)	60 (°F/HR)	100 (°F/HR)	LEAK TEST LIMITS	CRITICALITY 60 (°F/HR)	CRITICALITY 100 (°F/HR)
370		2257		2018	1378
375		2409		2129	1462
380		2572		2239	1552
385				2357	1649
390				2484	1754
395				2620	1866
400					1987
405					2116
410					2257
415					2409

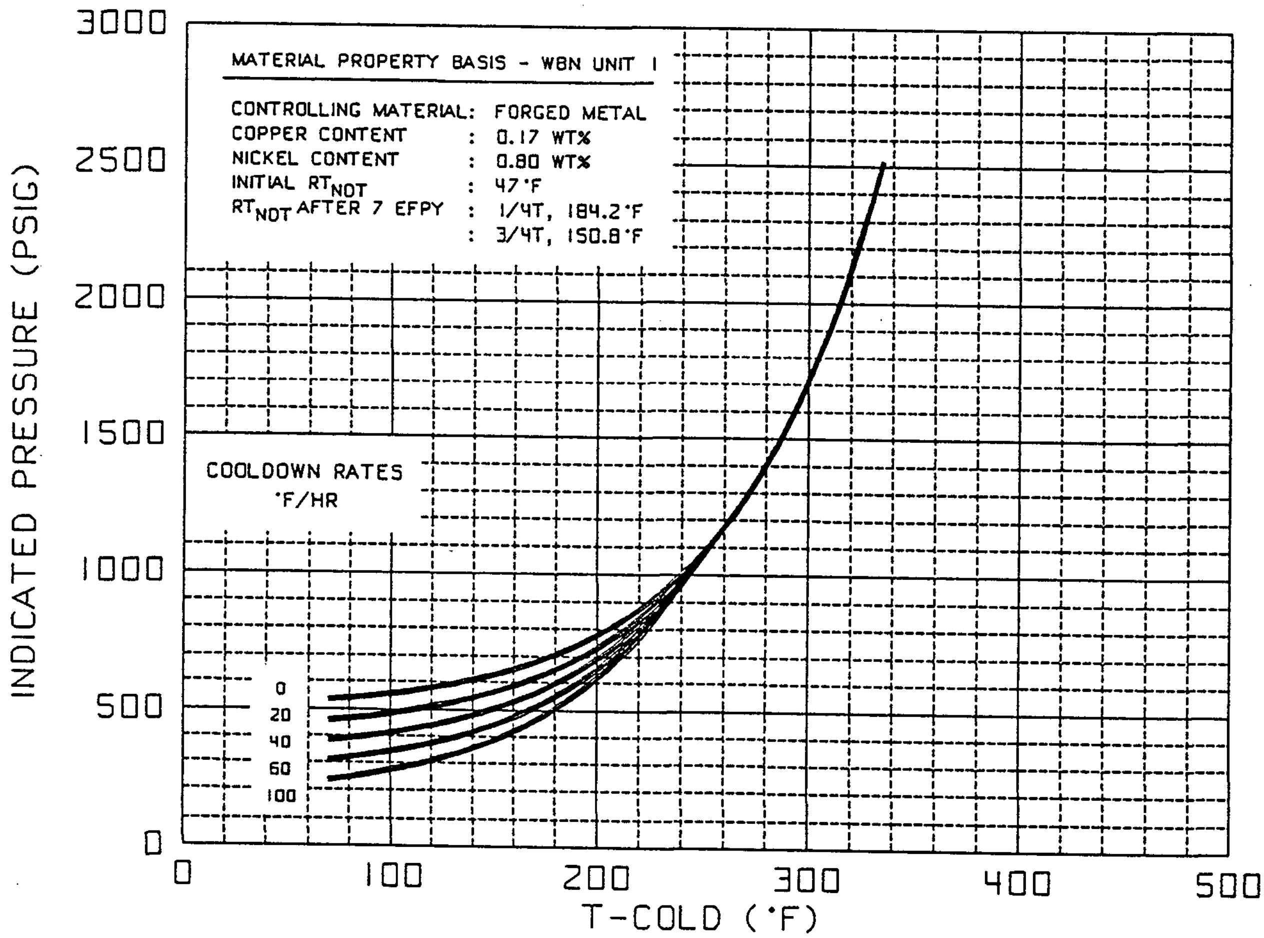


Figure 2.1-2

Reactor Coolant System
Cooldown Limitations
Applicable Up to 7 EFPY

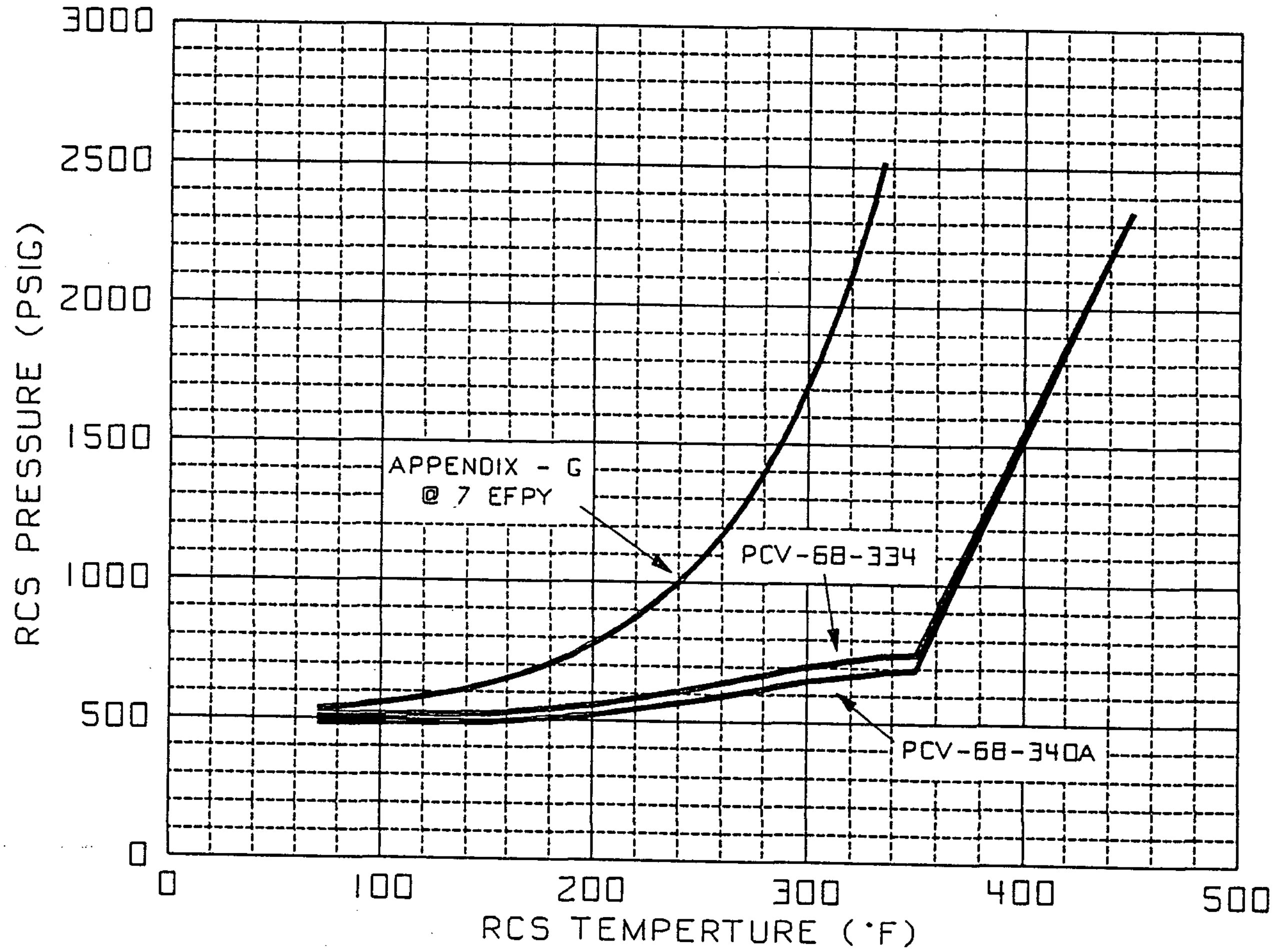
(Plotted data (Ref. 1) provided on next page)

UNIT 1 COOLDOWN LIMITS
(Data (Ref. 1) plotted on Fig 2.1-2)

RCS FLUID TEMPERATURE (DEG °F)	INDICATED PRESSURE, PSIG 100 (°F/HR)	60 (°F/HR)	40 (°F/HR)	20 (°F/HR)	0 (°F/HR)
70	241	313	386	460	535
75	246	318	390	464	538
80	251	323	395	467	541
85	257	328	399	472	545
90	264	334	404	474	549
95	271	340	410	481	554
100	278	346	416	486	558
105	286	353	422	492	563
110	294	361	429	498	568
115	303	369	436	505	575
120	313	378	444	512	581
125	324	387	452	519	588
130	335	397	462	527	595
135	347	408	471	538	603
140	360	419	482	545	611
145	374	433	493	556	620
150	389	446	505	566	629
155	406	461	518	578	640
160	423	476	532	591	652
165	442	493	547	604	664
170	462	511	562	619	677
175	484	531	581	634	691
180	507	552	600	651	706
185	532	574	620	667	722
190	559	598	641	688	739
195	588	624	665	709	758
200	620	652	690	731	778

UNIT 1 COOLDOWN LIMITS
(Data (Ref. 1) plotted on Fig 2.1-2)

RCS FLUID TEMPERATURE (DEG °F)	INDICATED PRESSURE, PSIG 100 (°F/HR)	60 (°F/HR)	40 (°F/HR)	20 (°F/HR)	0 (°F/HR)
205	653	682	716	756	799
210	689	714	745	781	822
215	728	749	778	809	847
220	770	786	810	839	874
225	815	827	846	871	903
230	863	870	884	906	934
235	915	916	926	943	967
240	970	966	971	983	1003
245	1030	1020	1019	1026	1041
250	1082	1077	1070	1072	1083
255	1127	1127	1126	1122	1127
260	1175	1175	1175	1175	1175
265	1228	1226	1220	1226	1226
270	1281	1281	1281	1281	1281
275	1341	1341	1341	1341	1341
280	1405	1405	1405	1405	1405
285	1473	1473	1473	1473	1473
290	1547	1547	1547	1547	1547
295	1626	1626	1626	1626	1626
300	1712	1712	1712	1712	1712
305	1804	1804	1804	1804	1804
310	1902	1902	1902	1902	1902
315	2008	2008	2008	2008	2008
320	2122	2122	2122	2122	2122
325	2245	2245	2245	2245	2245
330	2377	2377	2377	2377	2377
335	2519	2519	2519	2519	2519



(PCV-68-340A, Loops 1 & 2 Wide Range Temperature (Protection Set I)
(PCV-68-334, Loops 3 & 4 Wide Range Temperature (Protection Set II)

Figure 3.1-1

PORV Setpoint vs RCS Temperature

(Plotted data (Ref. 3) provided on next page)

PORV SETPOINT vs TEMPERATURE
(Data (Ref. 3) plotted on Fig 3.1-1)

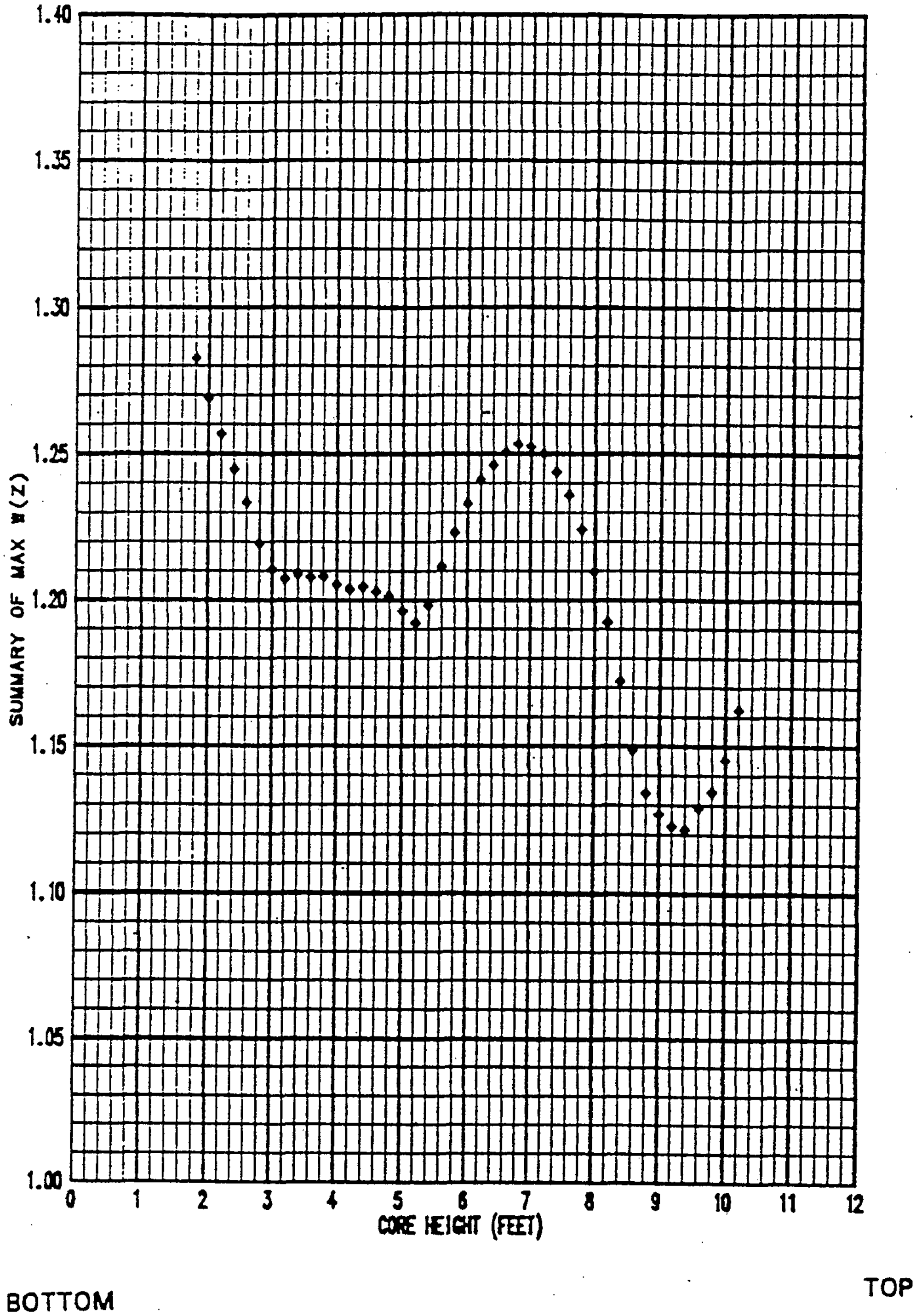
PORV Setpoint (psig)		
RCS Temperature (°F)	PCV-68-340A	PCV-68-334
70	485	515
100	485	515
150	490	520
200	520	555
250	580	625
275	615	665
300	652	702
350	693	748
450	2350	2350

Table 4.0-1
Surveillance Capsule Removal Schedule

Capsule	Vessel Location (deg.)	Capsule Lead Factor	Removal Time[a]	Estimated Fluence (n/cm^2)
U	56.0°	3.6	1.2	4.29×10^{18}
X	236.0°	3.6	4	1.43×10^{19}
V	58.5°	3.0	12	3.58×10^{19}
Y	238.5°	3.0	24	7.16×10^{19}
W	124.0°	3.6	Stand-By	----
Z	304.0°	3.6	Stand-By	----

(a) Effective Full Power Years (EFPY) from plant startup.

SAMPLE COLR FOR WATTS BAR UNIT 1 CYCLE 1



Height (Feet)	MAX W(Z)
0.000	1.0000
0.200	1.0000
0.400	1.0000
0.600	1.0000
0.800	1.0000
1.000	1.0000
1.200	1.0000
1.400	1.0000
1.600	1.0000
1.800	1.2827
2.000	1.2690
2.200	1.2568
2.400	1.2447
2.600	1.2333
2.800	1.2194
3.000	1.2105
3.200	1.2073
3.400	1.2090
3.600	1.2078
3.800	1.2082
4.000	1.2053
4.200	1.2038
4.400	1.2047
4.600	1.2029
4.800	1.2013
5.000	1.1963
5.200	1.1921
5.400	1.1983
5.600	1.2114
5.800	1.2232
6.000	1.2330
6.200	1.2411
6.400	1.2462
6.600	1.2507
6.800	1.2532
7.000	1.2523
7.200	1.2501
7.400	1.2437
7.600	1.2359
7.800	1.2242
8.000	1.2098
8.200	1.1923
8.400	1.1721
8.600	1.1486
8.800	1.1341
9.000	1.1271
9.200	1.1229
9.400	1.1217
9.600	1.1291
9.800	1.1345
10.000	1.1454
10.200	1.1624
10.400	1.0000
10.600	1.0000
10.800	1.0000
11.000	1.0000
11.200	1.0000
11.400	1.0000
11.600	1.0000
11.800	1.0000
12.000	1.0000

FIGURE 7: RAOC SUMMARY OF MAX W(Z) AT 10,000 MWD/MTU

ENCLOSURE 6

DRAFT PRESSURE AND TEMPERATURE LIMITS REPORT

WATTS BAR UNIT 1

RCS PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

REVISION 0

Prepared by : ______________________________

Approved by : ______________________________

RCS PRESSURE AND TEMPERATURE LIMITS REPORT FOR WATTS BAR UNIT 1

1.0 RCS Pressure and Temperature Limits Report (PTLR)

This PTLR for Watts Bar Unit 1 has been prepared in accordance with the requirements of Technical Specification 5.9.1.7. Revisions to the PTLR shall be provided to the NRC within 30 days of issuance.

The Technical Specifications affected by this report are listed below:

LCO 3.4.3. RCS Pressure and Temperature (P/T) Limits
LCO 3.4.12 Cold Overpressure Mitigation System (COMS)

2.0 RCS Pressure and Temperature Limits

The limits for LCO 3.4.3 are presented in the subsection which follows. These limits have been developed (Ref. 1) using the NRC-approved methodologies specified in Specification 5.9.1.7.

2.1 RCS Pressure and Temperature (P/T) Limits (LCO 3.4.3)

2.1.1 The RCS temperature rate-of-change limits are (Ref. 2):

a. A maximum heatup Rate 100°F per hour.

b. A maximum cooldown Rate 100°F per hour.

c. A maximum temperature change of $\leq$ 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

2.1.2 The RCS P/T limits for heatup, cooldown, inservice hydrostatic and leak testing, and criticality are specified by Figures 2.1-1 and 2.1-2 (Ref. 1).

3.0 Cold Overpressure Mitigation System (LCO 3.4.12)

The lift setting limits for the pressurizer Power Operated Relief Valves (PORVs) are presented in the subsection which follows. These lift setting limits have been developed using the NRC-approved methodologies specified in Specification 5.9.1.7.

3.1 Pressurizer PORV Lift Setting Limits

The pressurizer PORV lift setting limits are specified by Figure 3.1-1 (Ref. 3).