CL3.8-145

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~B. One or more DGs with lube oil inventory < [500] gal and > [425] gal.~~	~~B.1 Restore lube oil inventory to within limits.~~	~~48 hours~~
~~C~~B. ~~One or more DGs with~~Required DG fuel oil tank with stored fuel oil properties~~total particulates~~ not within limit(s).	~~C~~B.1 Restore fuel oil tank ~~total particulates~~ properties to within limit(s).	7 days CL3.8-146

(continued)

~~D~~C. ~~One or more DGs with new fuel oil properties not within limits~~Required Action and associated Completion Time of Condition B not met.	~~D.1 Restore stored fuel oil properties to within limits.~~ C.1 Isolate the associated DG fuel oil tank.	~~30 days~~ 2 hours CL3.8-146 R-12

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~E. One or more DGs with starting air receiver pressure < [225] psig and ≥ [125] psig.~~	~~E.1 Restore starting air receiver pressure to ≥ [225] psig.~~	~~48 hours~~
~~D~~F. Stored DG fuel oil supply: Unit 1 < 36,000 gal; Unit 2 < 65,000 gal. OR Required Action and associated Completion Time of Conditions A or C not met. ~~OR~~ ~~One or more DGs diesel fuel oil, lube oil, or starting air subsystem not within limits for reasons other than Condition A,B, C, D, or E.~~	-------------NOTE----------- Enter applicable Conditions and Required Actions of LCO 3.7.8, "CL System" when Condition D is entered as a result of stored fuel oil properties not within limits. ---------------------------- ~~D~~F.1 Declare ~~associated~~ DGs inoperable.	CL3.8-146 PA3.8-218 Immediately

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SURVEILLANCE REQUIREMENTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
[B. One ~~[or two]~~ batter[y][~~ies on one train~~] inoperable.	B.1	Verify associated battery charger is OPERABLE.	2 hours CL3.8-178
	AND		
	B.2	Verify other train battery is OPERABLE.	2 hours
	AND		CL3.8-179
	B.3	Verify other train battery charger is OPERABLE.	2 hours
	AND		
	B.~~2~~4	Restore batter[y][~~ies~~] to OPERABLE status	~~[2]~~ 8 hours] CL3.8-171
C. One DC electrical power subsystem inoperable for reasons other than Condition A-[or B].	C.1	Restore DC electrical power subsystem to OPERABLE status.	[2] hours
D. Required Action and Associated Completion Time not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 5.	36 hours

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SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is greater than or equal to the minimum established float voltage.	7 days
SR 3.8.4.2	Verify each battery charger supplies ≥ ~~[400]~~250 amps at greater than or equal to the minimum established float voltage for ≥ ~~[8]~~4 hours. OR Verify each battery charger can recharge the battery to the fully charged state within ~~[~~24 ~~]~~hours while supplying the ~~largest combined demands of the various continuous steady state loads~~demands of the various continuous steady state loads, after a battery discharge to the bounding design basis event discharge state.	~~[18]~~ 24 months X3.8-126 CL3.8-182
SR 3.8.4.3	- NOTES - 1. The modified performance discharge test in SR 3.8.6.6 may be performed in lieu of SR 3.8.4.3. 2. This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. 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]~~24 months X3.8-126

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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."]~~

[One DC electrical power subsystem shall be OPERABLE.]

- REVIEWER'S NOTE -

~~This second option above applies for plants having a pre-ITS licensing basis (CTS) for electrical power requirements during shutdown conditions that required only one DC electrical power subsystem to be OPERABLE. Action A the bracketed optional wording in Condition B are also eliminated for this case. The first option above is adopted for plants that have a licensing basis (CTS) requiring the same level of DC electrical power subsystem support as is required for power operating conditions.~~

NOTE

Service Building DC electrical power subsystem components may be used to replace safeguards DC electrical power subsystem components when the required safeguards DC electrical power subsystem is inoperable due to testing, maintenance, or replacement. R-12

PA3.8-212

APPLICABILITY: MODES 5 and 6,
During movement of ~~[recently]~~ irradiated fuel assemblies.

ACTIONS

- NOTE -

LCO 3.0.3 is not applicable.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
[A. One ~~[or two]~~required battery charger~~[s on one train]~~ inoperable. ~~AND~~	~~A.1~~ ~~AND~~	~~Restore battery terminal voltage to greater than or equal to the minimum established float voltage.~~	~~2 hours~~
~~The redundant train battery and charger[s] OPERABLE.~~	~~A.2~~ ~~AND~~ A.~~3~~1	~~Verify battery float current ≤ [2] amps.~~ Restore battery charger~~[s]~~ to OPERABLE status.	~~Once per [12] hours~~ PA3.8-198 ~~7 days]~~8 hours CL3.8-171

R-12

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One ~~[or more]~~ required DC electrical power subsystem~~[s]~~ inoperable [for reasons other than Condition A OR Required Action and associated Completion Time of Condition A not met].	~~B.1~~	~~Declare affected required feature(s) inoperable.~~	~~Immediately~~ PA3.8-199
	~~OR~~		
	B.~~2.~~1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	B.~~2.~~2	Suspend movement of ~~[recently]~~ irradiated fuel assemblies.	Immediately
	AND		
	B.~~2.~~3	Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND		
	B.~~2.~~4	Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	- NOTE - The following SRs are not required to be performed: SR 3.8.4.2 and SR 3.8.4.3. For DC sources required to be OPERABLE, the following SRs are applicable: SR 3.8.4.1 SR 3.8.4.2 SR 3.8.4.3	In accordance with applicable SRs

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3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Parameters

~~- REVIEWER'S NOTE -~~
~~Licensees must implement a program, as specified in Specification 5.5.17, to monitor battery parameters that is based on the recommendations of IEEE Standard 450-1995, "IEEE Recommended Practice For Maintenance, Testing, And Replacement Of Vented Lead-Acid Batteries For Stationary Applications."~~

LCO 3.8.6 Battery parameters for Train A and Train B batteries shall be within limits.

APPLICABILITY: When associated DC electrical power subsystems are required to be OPERABLE. R-12

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One ~~[or two]~~ batter[y]~~[ies on one train]~~ with one or more battery cells float voltage < ~~[~~2.07~~]~~ V.	A.1	Perform SR 3.8.4.1.	~~2~~8 hours
	AND		PA3.8-171
	A.2	Perform SR 3.8.6.1.	~~2~~8 hours
	AND		
	A.3	Restore affected cell voltage ≥ [2.07] V.	24 hours
B. One ~~[or two]~~ batter[y]~~[ies on one train]~~ with float current > ~~[~~2~~]~~ amps.	B.1	Perform SR 3.8.4.1.	~~2~~8 hours
	AND		

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PA3.8-100 CL3.8-145

BASES

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 or isolating the associated fuel oil tank. Therefore, t~~T~~he 7 day Completion Time allows for further evaluation, resampling and re-analysis of the DG fuel oil.

~~D~~C.1

CL3.8-146

~~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 for restoring the stored fuel oil properties. This period 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, or to restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the DG would still be capable of performing its intended function.~~

CL3.8-146

With a Required Action and associated Completion Time of Condition B not met, the associated fuel oil tank must be isolated within 2 hours. Isolation of a specific fuel oil tank may not make the associated DG inoperable since the DG can take suction from another fuel oil tank. Isolation of the associated fuel oil tank may cause entry into Conditions A or D which could result in the DG being inoperable.

R-12

(continued)

BASES

ACTIONS (continued)

~~battery float current is not less than or equal to [2] amps this indicates there may be additional battery problems and the battery must be declared inoperable.~~

Required Action A.3 requires, within 2 hours, that the diesel generator and safeguards equipment on the other train are verified to be OPERABLE. This verification ensures that the redundant train is OPERABLE ensuring that the plant will be able to mitigate an event as analyzed in the USAR (Ref. 3). CL3.8-179

R-12

Required Action A.~~3~~4 limits the restoration time for the inoperable battery charger to ~~7 days~~8 hours. ~~This action is applicable if an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage has been used (e.g., balance of plant non-Class 1E battery charger).~~ The ~~7 day~~8 hour Completion Time reflects a reasonable time to effect restoration of the qualified battery charger to OPERABLE status. CL3.8-171

B.1, B.2, B.3, and B.4

~~- REVIEWER'S NOTE -~~

~~The 2 hour Completion Times of Required Actions B.1 and C.1 are in brackets. Any licensee wishing to request a longer Completion Time will need to demonstrate that the longer Completion Time is appropriate for the plant in accordance with the guidance in Regulatory Guide (RG) 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications."~~

Condition B represents ~~one train with~~ one ~~[or two]~~ batter~~[~~y~~][ies]~~ inoperable. With one ~~[or two]~~ batter~~[~~y~~][ies]~~ inoperable, the DC bus is being supplied by the OPERABLE battery charger~~[s]~~. Any event that results in a loss of the AC bus supporting the battery charger~~[s]~~ will also result in loss of DC to that train. Recovery of the AC bus, especially if it is due to a loss of offsite power, will be hampered by the fact that many of the components necessary for the recovery (e.g., diesel generator control and field flash, AC load shed and diesel generator output circuit breakers, etc.) likely rely upon the batter~~[~~y~~][ies]~~. ~~In addition the energization transients of any DC loads that are beyond the capability of the battery charger[s] and normally require the assistance of the batter[y][ies] will not be able to be brought online.~~ Required Actions B.1, B.2, and B.3 verify that the associated battery charger, the other train battery and associated charger are OPERABLE

PA3.8-203

CL3.8-178

CL3.8-179

CL3.8-171

BASES

ACTIONS (continued)

within 2 hours. This time provides for either returning the inoperable battery to OPERABLE status or verifying that the associated charger and other train battery and charger are OPERABLE therefore, ensuring no loss of function exists. CL3.8-171

Required Action B.4 requires the inoperable battery to be restored to OPERABLE within 8 hours. The ~~[2]~~8 hour limit allows sufficient time to effect restoration of an inoperable battery given that the majority of the conditions that lead to battery inoperability (e.g., loss of battery charger, battery cell voltage less than [2.07] V, etc.) are identified in Specifications 3.8.4, 3.8.5, and 3.8.6 together with additional specific completion times.

C.1

Condition C 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 unit, minimizing the potential for complete loss of DC power to 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 for reasons other than Condition A or B (e.g., 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 could, however, result in the loss of minimum necessary DC electrical subsystems to mitigate a worst case accident, CL3.8-172 continued power operation should not exceed 2 hours. The 2 hour Completion Time ~~is based on Regulatory Guide 1.93 (Ref. 7) 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, to prepare to effect an orderly and safe unit shutdown.

D.1 and D.2

If the inoperable DC safeguards electrical power subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit R-12

BASES

ACTIONS (continued)

The other option requires that each battery charger be capable of recharging the battery after a ~~service~~discharge test coincident with supplying the ~~largest coincident demands of the various continuous steady state loads (irrespective of the status of the plant during which these demands occur)~~demands of the various continuous steady state loads, after the battery discharge to the bounding design basis event discharge state. ~~This level of loading may not normally be available following the battery service test and will need to be supplemented with additional loads.~~ The duration for this test may be longer than the charger sizing criteria since the battery recharge is affected by float voltage, temperature, and the exponential decay in charging current. The battery is fully recharged when the measured charging current is ≤ [2] amps. CL3.8-182

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~~24 month~~]~~ intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths. X3.8-126

SR 3.8.4.3

A battery service test is a special test of the battery 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~~2. CL3.8-172

~~The Surveillance Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 9) and Regulatory Guide 1.129 (Ref. 10), which state that t~~The battery service test should be performed during refueling operations, or at some other outage, with intervals between tests not to exceed ~~[18~~24 months~~].~~ X3.8-126

This SR is modified by two Notes. Note 1 allows the performance of a modified performance discharge test in lieu of a service test.

The reason for Note 2 is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g. post work testing following corrective maintenance, corrective modification, deficient or

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BASES

SURVEILLANCE REQUIREMENTS (continued)

incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment.

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REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 17~~ AEC "General Design Criteria for Nuclear Power Plant Construction Permits." Criterion 39, issued for comment July 10, 1976, as referenced in USAR, Section 1.2.

~~2. Regulatory Guide 1.6, March 10, 1971.~~

~~3. IEEE-308-[1978].~~ CL3.8-172

~~4~~2. ~~F~~USAR, ~~Chapter~~Section ~~[~~8~~]~~.

~~5~~3. ~~F~~USAR, ~~Chapter~~Section ~~[~~6~~]~~.

~~6. FSAR, Chapter [15].~~

~~7. Regulatory Guide 1.93, December 1974.~~

~~8. IEEE-450-[1995].~~

~~9. Regulatory Guide 1.32, February 1977.~~

~~10. Regulatory Guide 1.129, December 1974.~~

APPLICABLE SAFETY ANALYSES (continued)

shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

The DC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The DC electrical power subsystems~~, [each required] [the required] [subsystem~~ consisting of ~~two~~a battery~~ies,~~ one battery charger ~~per battery,~~ and the corresponding control equipment and interconnecting cabling within [~~one]~~ the train, ~~[are]~~ [is] required to be OPERABLE to support ~~[required]~~ [one] train~~[s]~~ of the distribution systems required OPERABLE by LCO 3.8.10, "Distribution Systems - Shutdown."] 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 ~~[involving handling recently irradiated fuel]~~).

A Note has been added to the LCO allowing the service building DC electrical power subsystem components to be used in lieu of the required safeguards DC electrical power subsystem components when the required safeguards DC electrical power subsystem is inoperable due to testing, maintenance, or replacement. The service building DC power electrical components include the battery, associated battery charger, and the interconnecting cabling. When any of the service building DC power electrical components are used in lieu of the safeguards DC electrical power subsystem components, they are required to be maintained in accordance with Specification 5.5.15 for monitoring various battery parameters that is based on the recommendations of IEEE Standard 450-1995, "IEEE Recommended Practice For Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries For Stationary Applications" (Ref. 3).

R-12

PA3.8-212

R-12

APPLICABILITY

The DC electrical power source~~s~~ required to be OPERABLE in MODES 5 and 6, and during movement of ~~[recently]~~ irradiated fuel assemblies, provide assurance that:

a. Required features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core;

APPLICABILITY (continued)

b. Required features needed to mitigate a fuel handling accident ~~[involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [ ] days)]~~ are available;

c. Required 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.

The DC electrical power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.4.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. 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 MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

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

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- REVIEWER'S NOTE -

~~ACTION A is included only when plant-specific implementation of LCO 3.8.5 includes the potential to require both trains of the DC System to be OPERABLE. If plant-specific implementation results in LCO 3.8.5 requiring only one trains of the DC System to be OPERABLE, then ACTION A is omitted and ACTION B is renumbered as ACTION A.~~

Condition A represents one train with one ~~[or two]~~required battery charger~~s~~ inoperable (e.g., the voltage limit of SR 3.8.4.1 is not maintained). ~~The ACTIONS provide a tiered response that focuses on returning the battery to the fully charged state and restoring a fully qualified charger to OPERABLE status in a reasonable time period. Required Action A.1 requires that the battery terminal voltage be restored to greater than or equal to the minimum established float voltage within 2 hours. This time provides for returning the inoperable charger to OPERABLE status or providing an alternate means of restoring battery~~

PA3.8-198

R-12

ACTIONS (continued)

~~terminal voltage to greater than or equal to the minimum established float voltage. Restoring the battery terminal voltage to greater than or equal to the minimum established float voltage provides good assurance that, within [12] hours, the battery will be restored to its fully charged condition (Required Action A.2) from any discharge that might have occurred due to the charger inoperability.~~ PA3.8-198

- REVIEWER'S NOTE -

~~A plant that cannot meet the 12-hour Completion Time due to an inherent battery charging characteristic can propose an alternate time equal to 2 hours plus the time experienced to accomplish the exponential charging current portion of the battery charge profile following the service test (SR 3.8.4.3).~~

~~A discharged battery having terminal voltage of at least the minimum established float voltage indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within [12] hours.~~

~~If established battery terminal float voltage cannot be restored to greater than or equal to the minimum established float voltage within 2 hours, and the charger is not operating in the current-limiting modes, a faulty charger is indicated. A faulty charger that is incapable of maintaining established battery terminal float voltage does not provide assurance that it can revert to and operate properly in the current limit modes that is necessary during the recovery period following a battery discharge event that the DC system is designed for.~~

~~If the charger is operating in the current limit mode after 2 hours that is an indication that the battery is partially discharged and its capacity margins will be reduced. The time to return the battery to its fully charged condition in this case is a function of the battery charger capacity, the amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within [12] hours (Required Action A.2).~~

~~Required Action A.2 requires that the battery float current be verified as less than or equal to [2] amps. This indicates that, if the battery had~~ R-12

ACTIONS (continued)

~~discharged as the result of the inoperable battery charger, it has now been fully recharged. If at the expiration of the initial [12] hour period the battery float current is not less than or equal to [2] amps this indicates there may be additional battery problems and the battery must be declared inoperable.~~ CL3.8-171

Required Action A.~~3~~1 limits the restoration time for the inoperable battery charger to ~~7 days~~8 hours. ~~This action is applicable if an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage has been used (e.g. balance of plant non-Class 1E battery charger).~~ The 8 hour~~7 day~~ Completion Time reflects a reasonable time to effect restoration of the qualified battery charger to OPERABLE status. R-12

~~B.1,~~ B.~~2~~.1., B.~~2~~.2, B.~~2~~.3, and B.~~2~~.4

~~[If two trains are required by LCO 3.8.10, the remaining train with DC power available may be capable of supporting sufficient systems to allow continuation of CORE ALTERATIONS and fuel movement] [involving handling recently irradiated fuel]. By allowing the option to declare required features inoperable with the 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 involve undesired administrative efforts.~~ Condition B represents one train with one required DC electrical power subsystem inoperable for reasons other than Condition A or if the Required Actions and associated Completion Time of Condition A are not met. In this Condition there may not be adequate DC power available to support the subsystems required by LCO 3.8.10. Therefore, ~~the allowance for sufficiently~~ conservative actions are required ~~is made~~ (i.e., to suspend CORE ALTERATIONS, movement of ~~[recently]~~ irradiated fuel assemblies, and operations involving positive reactivity additions) that assure ~~could result in failure to meet~~ the minimum SDM or boron concentration limit is met~~required~~ to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that ~~what would be~~ required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM. PA3.8-199 R-12

ACTIONS (continued)

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required DC electrical power subsystem~~[s]~~ 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 3.8.5.1 requires performance of all Surveillances required by SR 3.8.4.1 through SR 3.8.4.3. Therefore, see the corresponding Bases for LCO 3.8.4 for a discussion of each SR. R-12

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DC sources from being discharged below their capability to provide the required power supply or otherwise rendered inoperable during the performance of SRs. It is the intent that these SRs must still be capable of being met, but actual performance is not required.

REFERENCES

1. ~~F~~ USAR~~, Chapter [6]~~ Section 6.

2. ~~F~~ USAR~~, Chapter [15]~~ Section 14. PA3.8-172

3. IEEE-450-1995.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

b. A worst-case single failure.

Battery parameters satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Battery parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. Battery parameter limits are conservatively established, allowing continued DC electrical system function even with limits not met. Additional preventative maintenance, testing, and monitoring performed in accordance with the plant procedures~~[licensee controlled program]~~ is conducted as specified in Specification 5.5.1~~7~~5.

APPLICABILITY

The battery parameters are required solely for the support of the associated DC electrical power subsystems. Therefore, battery parameter limits are only required when the DC power source is required to be OPERABLE. Refer to the Applicability discussion in Bases for LCO 3.8.4 and LCO 3.8.5.

ACTIONS

A Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each battery. This is acceptable, since Required Actions for each Condition provide appropriate compensatory actions. PA3.8-158

A.1, A.2, and A.3 PA3.8-171

With one or more cells in one ~~or more~~ battery~~ies in one train~~ < ~~[~~2.07~~]~~ V, the battery cell is degraded. Within ~~2~~8 hours verification of the required battery charger OPERABILITY is made by monitoring the battery terminal voltage (SR 3.8.4.1) and of the overall battery state of charge by monitoring the battery float charge current (SR 3.8.6.1). This assures that there is still sufficient battery capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of one or more cells in one ~~or more~~ battery~~ies~~ < ~~[~~2.07~~]~~ V, and continued operation is permitted for a limited period up to 24 hours. R-12

Since the Required Actions only specify "perform," a failure of SR 3.8.4.1 or SR 3.8.6.1 acceptance criteria does not result in this Required Action not met. However, if one of the SRs is failed the appropriate Condition(s), depending on the cause of the failures, is entered. If

BASES

ACTIONS (continued)

SR 3.8.6.1 is failed then there ~~is~~may not be assurance that there is still sufficient battery capacity to perform the intended function and the battery must be declared inoperable immediately.

R-12

B.1 and B.2

One ~~or more~~ battery~~ies in one train~~ with float > {2} amps indicates that a partial discharge of the battery capacity has occurred. This may be due to a temporary loss of a battery charger or possibly due to one or more battery cells in a low voltage condition reflecting some loss of capacity. Within ~~2~~8 hours verification of the required battery charger OPERABILITY is made by monitoring the battery terminal voltage. If the terminal voltage is found to be less than the minimum established float voltage there are two possibilities, the battery charger is inoperable or is operating in the current limit mode. Condition A addresses charger inoperability. If the charger is operating in the current limit mode after ~~2~~8 hours that is an indication that the battery has been substantially discharged~~ and likely cannot perform its required design functions~~. The time to return the battery to its fully charged condition in this case is a function of the battery charger capacity, the amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within~~{~~12~~}~~hours (Required Action B.2). The battery must therefore be declared inoperable.

PA3.8-171

R-12

CL3.8-166

R-12

If the float voltage is found to be satisfactory but there are one or more battery cells with float voltage less than~~{~~2.07~~}~~V, the associated "OR" statement in Condition F is applicable and the battery must be declared inoperable immediately. If float voltage is satisfactory and there are no cells less than~~{~~2.07~~}~~V there is good assurance that, within~~{~~12~~}~~hours, the battery will be restored to its fully charged condition (Required Action B.2) from any discharge that might have occurred due to a temporary loss of the battery charger.

R-12

- REVIEWER'S NOTE -

~~A plant that cannot meet the 12-hour Completion Time due to an inherent battery charging characteristic can propose an alternate time equal to 2 hours plus the time experienced to accomplish the exponential~~

BASES

ACTIONS (continued)

~~charging current portion of the battery charge profile following the service test (SR 3.8.4.3).~~

A discharged battery with float voltage (the charger setpoint) across its terminals indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within ~~[12]~~24 hours, avoiding a premature shutdown with its own attendant risk and the battery is not inoperable. PA3.8-187

If the condition is due to one or more cells in a low voltage condition but still greater than ~~[~~2.07~~]~~ V and float voltage is found to be satisfactory, this is not indication of a substantially discharged battery and ~~[12]~~24 hours is a reasonable time prior to declaring the battery inoperable. R-12

Since Required Action B.1 only specifies "perform," a failure of SR 3.8.4.1 acceptance criteria does not result in the Required Action not met. However, if SR 3.8.4.1 is failed, the appropriate Condition(s), depending on the cause of the failure, is entered.

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

With one ~~or more~~ battery~~ies in one train~~ with one or more cells electrolyte level above the top of the plates, but below the minimum established design limits, the battery still retains sufficient capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of electrolyte level not met. Within 31 days the minimum established design limits for electrolyte level must be re-established.

With electrolyte level below the top of the plates there is a potential for dryout and plate degradation. Required Actions C.1 and C.2 address this potential (as well as provisions in Specification 5.5.1~~7~~5, Battery Monitoring and Maintenance Program). They are modified by a note that indicates they are only applicable if electrolyte level is below the top of the plates. Within 8 hours level is required to be restored to above the top of the plates. The Required Action C.2 requirement to verify that there is no leakage by visual inspection and the Specification 5.5.1~~7~~5.b item to initiate action to equalize and test in accordance with manufacturer's recommendation are taken from Annex D of IEEE Standard 450-1995. They are performed following the restoration of the electrolyte level to CL3.8-172

BASES

ACTIONS (continued)

above the top of the plates. Based on the results of the manufacturer's recommended testing the batter[y]~~[ies]~~ may have to be declared inoperable and the affected cell[s] replaced.

D.1

With one ~~or more~~ battery~~ies in one train~~ with pilot cell temperature less than the minimum established design limits, 12 hours is allowed to restore the temperature to within limits. A low electrolyte temperature limits the current and power available. Since the battery is sized with margin, while battery capacity is degraded, sufficient capacity exists to perform the intended function and the affected battery is not required to be considered inoperable solely as a result of the pilot cell temperature not met.

E.1 PA3.8-161

With ~~one or more~~ batteries in the redundant trains with battery parameters not within limits there is not sufficient assurance that battery capacity has not been affected to the degree that the batteries can still perform their required function, given that redundant batteries are involved. With redundant batteries involved this potential could result in a total loss of function on multiple systems that rely upon the batteries. The longer Completion Times specified for battery parameters on non-redundant batteries not within limits are therefore not appropriate, and the parameters must be restored to within limits on at least one train within 8~~2~~ hours. R-12

F.1 R-12

With one or more batteries with any battery parameter outside the allowances of the Required Actions for Condition A, B, C, D, or E, sufficient capacity to supply the ~~maximum expected~~design load requirement is not assured and the corresponding battery must be declared inoperable. Additionally, discovering one or more batteries in one train with one or more battery cells float voltage less than ~~[~~2.07~~]~~ V and float current greater than ~~[~~2~~]~~ amps indicates that the battery capacity may not be sufficient to perform the intended functions. The battery must therefore be declared inoperable immediately.

R-12

BASES

SURVEILLANCE REQUIREMENTS (continued)

electrolyte temperature is maintained above this temperature to assure the battery can provide~~d~~ the required current and voltage to meet the design requirements. Temperatures lower than assumed in battery sizing calculations act to inhibit or reduce battery capacity. The Frequency is consistent with IEEE-450 (Ref. 1). CL3.8-172 R-12

SR 3.8.6.6

A battery performance discharge 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.

Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.6.6 ~~; however, only the modified performance discharge test may be used to satisfy the battery service test requirements of SR 3.8.4.3.~~

A modified discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rate of the duty cycle). This will often confirm the battery's ability to meet the critical period of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test.

It may consist of just two rates; for instance the one minute rate for the battery or the largest current load of the duty cycle, followed by the test rate employed for the performance test, both of which envelope the duty cycle of the service test. Since the ampere-hours removed by a one minute discharge represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test must remain above the minimum battery terminal voltage specified in the battery service test for the duration of time equal to that of the service test.

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 1~~3~~) and IEEE-485 (Ref. 4). 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. Furthermore, the battery is sized to meet or exceed CL3.8-172

PA3.8-100

BASES (continued)

maintaining required Reactor Protection Instrument AC Panels~~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 CL3.8-163

b. A worst case single failure.

Inverters ~~are a part of the distribution system and, as such,~~ satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii)~~the NRC Policy Statement~~. PA3.8-217

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 inverters ~~[(two per train)]~~ ensure an uninterruptible supply of AC electrical power to the Reactor Protection Instrument AC Panels~~vital buses~~ even if the 4~~.16~~ kV ~~safety~~Safeguards buses are de-energized. R-12

(continued)

PA3.8-100

BASES (continued)

OPERABLE~~Operable~~ inverters require the associated ~~vital~~Reactor Protection Instrument AC Panel ~~bus~~ to be powered by the inverter with ~~output voltage and frequency within tolerances, and~~ power supply~~input~~ to the inverter from a ~~[~~125 VDC~~]~~ station battery. ~~Alternatively~~Normally, the power supply ~~may be~~is from an internal AC source via rectifier ~~as long as~~with the station battery ~~is~~ available as the uninterruptible power supply. CL3.8-180 CL3.8-183

R-12

~~This LCO is modified by a Note that allows [one/two] inverters to be disconnected from a [common] battery for ≤ 24 hours, if the vital bus(es) is powered from a [Class 1E constant voltage transformer or inverter using internal AC source] 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.~~ PA3.8-185

~~The intent of this Note is to limit the number of inverters that may be disconnected. Only those inverters associated with the single battery undergoing an equalizing charge may be disconnected. All other inverters must be aligned to their associated batteries, regardless of the number of inverters or unit design.~~ PA3.8-185

(continued)

PA3.8-100

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 CL3.8-205

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 LCO 3.8.8, "Inverters – Shutdown."

ACTIONS

A.1 and A.2

With one Reactor Protection Instrument AC inverter inoperable, Required Action A.1 and A.2 require verification, within 2 hours, the Reactor Protection Instrument AC panel with an inoperable inverter is powered from Panel 117 (Unit 2 - 217) or verify that the Reactor Protection Instrument AC panel with an inoperable inverter is powered from its inverter bypass source. CL3.8-183

Plant design provides acceptable alternate methods of powering a Reactor Protection Instrument AC panel with an inoperable inverter. Panel 117 (Unit 2 - Panel 217), by plant design, can provide reliable power to a Reactor Protection Instrument AC panel. Alternatively, a Reactor Protection Instrument AC panel may be powered by an inverter internal bypass. In the event an inverter becomes inoperable, the the inverter static transfer bypass switch will automatically bypass, thus providing power to the associated Reactor Protection Instrument AC panel and maintain OPERABILITY. Required Actions A.1 and A.2 require verification that only one Reactor Protection Instrument AC

R-12

(continued)

PA3.8-100

BASES (continued)

panel is powered from Panel 117 (Unit 2 - Panel 217) or an inverter bypass source. This verification must be completed within 2 hours.

R-12

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

With ~~a required~~two Reactor Protection Instrument AC inverters inoperable, the~~its~~ associated Reactor Protection Instrument AC panels~~vital bus becomes inoperable until it is [manually]~~ are considered to be inoperable unless they are energized from Panel 117 (Unit 2 - Panel 217) or they are automatically re-energized ~~from~~by their inverter~~its~~ static transfer switch. ~~[Class 1E constant voltage source transformer or inverter using internal AC source]~~.

R-12

CL3.8-183

For this reason a Note has been included in Condition ~~A~~B requiring the entry into the Conditions and Required Actions of LCO 3.8.9, "Distribution Systems – Operating." This ensures that the ~~vital~~Reactor Protection Instrument AC panel ~~bus~~ is re-energized within 2 hours. Plant design provides acceptable alternate methods of powering Reactor Protection Instrument AC panels with an inoperable inverter. Panel 117 (Unit 2 - Panel 217), by plant design, can provide reliable power to a Reactor Protection Instrument AC panel. Alternatively, a Reactor Protection Instrument AC panel may be powered by an inverter internal bypass. In the event an inverter becomes inoperable, the inverter static transfer bypass switch will automatically bypass, thus providing power to the associated Reactor Protection Instrument AC panel and maintain OPERABILITY. Therefore, based on plant design, Required Actions B.1 and B.2 require verification that no more than one Reactor Protection Instrument AC inverter will be powered from Panel 117 (Unit 2 - Panel 217) and one or both Reactor Protection Instrument AC panel(s) are powered from an inverter bypass source. This verification must be completed within 2 hours.

R-12

(continued)

PA3.8-100

BASES

Required Action ~~AB.1~~3 allows ~~24~~8 hours to fix the inoperable inverter and return it to service. The ~~24~~8 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 Reactor Protection Instrument AC Panel~~vital bus~~ is powered from its alternate ~~constant voltage~~ source, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the Reactor Protection Instrument AC Panel ~~vital buses~~ is the preferred source for powering instrumentation trip setpoint devices. CL3.8-183 R-12

ACTIONS (continued)

~~B~~C.1 and ~~B~~C.2 R-12

If the inoperable devices or components cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 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 plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and Reactor Protection Instrument AC panels~~vital buses~~ energized from the inverter. The verification of proper voltage ~~and frequency~~ output ensures PA3.8-102

that the required power is readily available for the instrumentation of the RPS and ESFAS connected to the Reactor Protection Instrument AC panels~~vital buses~~. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.

REFERENCES

1. ~~F~~USAR, ~~Chapter~~Section ~~[8]~~.
2. ~~F~~USAR, ~~Chapter~~Section ~~[6]~~14.
3. ~~FSAR, Chapter [15].~~

PA3.8-100

BASES (continued)

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.

Inverter requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7.

ACTIONS

LCO 3.0.3 is not applicable while in MODES 5 and 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. 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 MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3 while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily. TA3.8-140

R-12

~~A.1,~~ A.~~2.~~1, A.~~2.~~2, A.~~2.~~3, and A.~~2.~~4

If the required inverter is inoperable, the remaining OPERABLE Reactor Protection Instrument AC Panel power supplies~~two trains are~~ as required by LCO 3.8.10, "Distribution Systems – Shutdown," ~~the remaining OPERABLE Inverters~~ may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, ~~and~~or operations with a potential for positive reactivity additions. ~~By the allowance of the option to declare~~ CL3.8-177

~~ACTIONS~~ ~~A.1, A.2.1, A.2.2, A.2.3, and A.2.4 (continued)~~

(continued)

PA3.8-100

BASES

~~required features inoperable with the 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 involve undesired administrative efforts.~~ Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, 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 SDM is maintained.~~that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6)). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

PA3.8-215

R-3

TA3.8-117

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required inverter~~s~~ and to continue this action until restoration is accomplished in order to provide the necessary inverter power to the unit safety systems.

(continued)

PA3.8-100

BASES (continued)

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required inverter~~s~~ 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~~. CL3.8-177

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the required inverter~~s~~ ~~are~~is functioning properly with all required circuit breakers closed and Reactor Protection Instrument AC Panel~~ 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 connected to the Reactor Protection Instrument AC ~~vital buses~~Panel. The 7 day Frequency takes into account the ~~redundant capability~~reliability of the ~~inverters~~instrument panel power sources and other indications available in the control room that alert the operator to ~~inverter~~ malfunctions. PA3.8-102

REFERENCES

~~1. FSAR, [6].~~

~~2. FSAR, Chapter [15].~~None.

(continued)

PA3.8-100

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.9 Distribution Systems – Operating

BASES

BACKGROUND

The onsite safeguards~~Class 1E~~ AC~~,~~ and DC~~, and AC vital bus~~ electrical power distribution systems are divided by train into ~~[two]~~ redundant and independent ~~AC, DC, and AC vital bus~~ electrical power distribution subsystems. The onsite Reactor Protection Instrument AC Distribution System is divided by channels into four separate subsystems (Ref. 1). CL3.8-167

Each~~The~~ AC electrical power subsystem ~~for each train~~ consists of a ~~primary~~ safeguards~~Engineered Safety Feature (ESF)~~ 4~~.16~~ kV bus and two~~secondary~~ [480 ~~and 120]~~ V buses.~~,~~ These in turn supply power to distribution panels~~,~~ and motor control centers (MCCs)~~ and load centers~~. Each safeguards ~~[4.16~~ kV ~~ESF~~ bus~~]~~ has two~~at least [one separate and independent~~ offsite sources of power~~]~~ as well as a dedicated onsite diesel generator (DG) source. Each safeguards ~~[4.16~~ kV ~~ESF~~ bus~~]~~ is normally connected to an~~ preferred~~ offsite source. After a loss of thise ~~preferred~~ offsite power source~~ to a 4.16 kV ESF bus~~, a transfer to the alternate offsite source is accomplished by a load sequencer, initiated by~~utilizing a time delayed~~ bus undervoltage relays. If all offsite sources are unavailable, the onsite emergency DG supplies power to the safeguards 4~~.16~~ kV ~~ESF~~ bus. Control power for the 4~~.16~~ kV and 480 V bus breakers is supplied from the safeguards DC distribution~~Class 1E batteries~~ system. Additional description of the safeguards AC~~this~~ system may be found in the Bases for LCO 3.3.4, "4 kV Safeguards Bus Voltage Instrumentation," and the Bases for LCO 3.8.1, "AC Sources – Operating~~," and the Bases for LCO 3.8.4, "DC Sources – Operating~~." R-12 CL3.8-167

(continued)

The ~~secondary~~ AC electrical power distribution system for each train includes the safety related buses~~load centers, motor control centers~~ and MCCs~~, and distribution panels~~ shown in Table B 3.8.9-1. R-12

The 120 V Reactor Protection Instrument AC ~~vital buses~~Panels are arranged in four~~two~~ load groups ~~per train~~ and are normally powered from ~~the~~ inverters. An~~The~~ alternate power supply for the instrument panels is~~vital buses are Class 1E constant voltage source~~ the inverter bypass transformer~~s~~ powered from the same MCC~~train~~ as the associated inverter. Another alternate power supply is from the unit 208/120 VAC interruptable panel.~~, and its~~ U~~u~~se of these supplies is governed by LCO 3.8.7, "Inverters – Operating."~~ Each constant voltage source transformer is powered from a Class 1E AC bus.~~ CL3.8-167

There are two independent 125~~/250~~ VDC electrical power distribution subsystems (one for each train). The 125 VDC safeguards electrical power system consists of two independent and redundant safety related DC safeguards electrical power subsystems (Train A and Train B). The sources for each train are a 125 VDC battery, a battery charger, and all the associated control equipment and interconnecting cabling. CL3.8-167

The list of the~~all~~ required Reactor Protection Instrument AC and safeguards DC distribution panels~~buses~~ is presented in Table B 3.8.9-1.

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR~~, Chapter [6] (Ref. 1), and in the FSAR, Chapter [15]~~ (Ref. 2)~~,~~ assume ESF systems are OPERABLE. The safeguards AC, DC, and Reactor Protection Instrument AC~~ vital bus~~ electrical power distribution systems are designed to provide sufficient capacity,

(continued)

PA3.8-100

Maintaining the Train A and Train B safeguards AC~~,~~ and DC, and Reactor Protection Instrument 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 or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor. This does not preclude redundant safeguards 4 kV buses from being powered from the same offsite path.

CL3.8-167

LCO (continued)

OPERABLE AC electrical power distribution subsystems require the associated buses~~, load centers,~~ and MCCs~~motor control centers, and distribution panels~~ to be energized to their proper voltages. OPERABLE DC electrical power distribution subsystems require the associated panels~~buses~~ to be energized to their proper voltage from either the associated battery or charger. OPERABLE Reactor Protection Instrument AC ~~vital bus~~ electrical power distribution subsystems require the associated panels~~buses~~ to be energized to their proper voltage~~ from the associated [inverter via inverted DC voltage, inverter using internal AC source, or Class 1E constant voltage transformer]~~.

R-12

~~In addition, tie breakers between redundant safety related AC, DC, and AC vital bus power distribution subsystems, if they exist, must be open. This prevents any electrical malfunction in any power distribution subsystem from propagating to the redundant subsystem, that could cause the failure of a redundant subsystem and 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 from being powered from the same offsite circuit.~~

CL3.8-167

(continued)

PA3.8-100

BASES (continued)

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 R-12

CL3.8-205

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 LCO 3.8.10, "Distribution Systems — Shutdown."

ACTIONS

A.1 and A.2

With one or more ~~required~~ safeguards AC electrical power distribution subsystems~~buses~~, ~~load centers, motor control centers, or distribution panels, except AC vital buses~~, ~~in one train~~ inoperable, the remaining AC electrical power distribution subsystems ~~in the other train is~~are 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, there are two Required Actions that can be taken. Required Action A.1 would allow declaring the affected supported feature(s), being powered from the inoperable portion of the safeguards AC electrical power distribution system, inoperable. If Required Action A.1 is used, LCO 3.0.6 would also be entered to verify that no loss of function would exist. If LCO 3.0.6 identifies that a loss of function did exist, Condition E would be entered.

R-12 PA3.8-213 R-12 PA3.8-213 R-12

(continued)

Required Action A.2 requires~~d~~ safeguards AC electrical power ~~buses, load centers~~, ~~motor control centers, and~~ distribution subsystems ~~panels must~~ to be restored to OPERABLE status within 8 hours. R-12

~~Condition A~~The worst scenario is one train without AC power (i.e., no offsite power to the train 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 operator's attention be focused on minimizing the potential for loss of power to the remaining train by stabilizing the unit, and on restoring power to the affected train. The 8 hour time limit before requiring a unit shutdown in this Condition is acceptable because of: R-12

a. The potential for decreased safety if the unit operator's attention is diverted from the evaluations and actions necessary to restore power to the affected train, to the actions associated with taking the unit to shutdown within this time limit; and

b. The potential for an event in conjunction with a single failure of a redundant component in the train with AC power.

The second Completion Time for Required Action A.2~~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 restored 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 R-12

(continued)

~~ACTIONS~~ ~~A.1 (continued)~~

become inoperable, and AC distribution restored OPERABLE. This could continue indefinitely.

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.

Required Action A.1 and A.2 are modified by a Note that requires the applicable Conditions and Required Actions of LCO 3.8.4, "DC Sources - Operating," to be entered for DC trains made inoperable by inoperable AC power distribution subsystems. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components. Inoperability of a distribution system can result in loss of charging power to batteries and eventual loss of DC power. This Note ensures that the appropriate attention is given to restoring charging power to batteries, if necessary, after loss of distribution systems. PA3.8-213

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

With one Reactor Protection Instrument AC Panel~~vital bus~~ inoperable, the remaining OPERABLE Reactor Protection Instrument AC Panels~~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, there are two Required Actions that can be taken. Required Action C.1 would allow declaring the affected supported feature(s) PA3.8-213

R-12

R-12

(continued)

being powered from the inoperable portion of the Reactor Protection Instrument AC panel, inoperable. If Required Action C.1 is used, LCO 3.0.6 would also be entered to verify that no loss of function would exist. If LCO 3.0.6 identifies that a loss of function did exist, Condition E would be entered. Required Action C.2 ~~the~~ requires~~d~~ the Reactor Protection Instrument AC panel~~vital bus must~~ to be restored to OPERABLE status within 2 hours by powering the panel~~bus~~ from the associated ~~[~~inverter ~~via inverted DC~~, inverter ~~using internal AC source, or Class 1E constant voltage~~bypass transformer~~]~~, or interruptible panel. PA3.8-213 R-12

Condition ~~B~~C represents one Reactor Protection Instrument AC ~~vital bus~~panel without power~~; potentially both the DC source and the associated AC source are nonfunctioning~~. In this situation, the unit is significantly more vulnerable to a complete loss of all noninterruptible power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining instrument~~vital buses~~panels and restoring power to the affected instrument~~vital bus~~panel. CL3.8-167 R-12

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that are without adequate instrument~~vital~~ AC power. Taking exception to LCO 3.0.2 for components without adequate instrument~~vital~~ AC power, that would have the Required Action Completion Times shorter than 2 hours if declared inoperable, is acceptable because of:

~~ACTIONS~~ ~~B.1 (continued)~~

(continued)

a. The potential for decreased safety by requiring a change in 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 ~~A~~applicable Conditions and Required Actions for components without adequate instrument~~vital~~ AC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train; and

c. The potential for an event in conjunction with a single failure of a redundant component.

The 2 hour Completion Time takes into account the importance to safety of restoring the Reactor Protection Instrument AC ~~vital bus~~Panel to OPERABLE status, the redundant capability afforded by the other OPERABLE instrument~~vital buses~~ panels, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action C~~B~~.2~~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 C~~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 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." R-12

(continued)

PA3.8-100

This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition ~~C~~B was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely. R-12

ACTIONS (continued)

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

PA3.8-213

With one or more safeguards DC electrical power distribution subsystem panel(s)~~bus(es) in one train~~ inoperable, the remaining safeguards DC electrical power distribution subsystem~~s are~~ 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 safeguards DC electrical power distribution subsystem could result in the minimum required ESF functions not being supported. Therefore, there are two Required Actions that can be taken. Required Action B.1 would allow declaring the affected supported feature(s), being powered from the inoperable portion of the safeguards DC panel, inoperable. If Required Action B.1 is used, LCO 3.0.6 would also be entered to verify that no loss of function would exist. If LCO 3.0.6 identifies that a loss of function did exist, Conditon E would be entered. Required Action B.2 ~~the~~ ~~[~~require~~d]~~s the DC panels~~buses must~~ be restored to OPERABLE status within 2 hours by powering the bus from the associated battery, ~~or~~ charger, or portable charger. R-12 R-12

~~Condition C represents one train~~The worst case scenario is one train without ~~adequate~~ safeguards DC power; potentially with both ~~with~~ the battery significantly degraded and the associated charger nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all DC power. It is, therefore, imperative that the

(continued)

PA3.8-100

BASES

operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining trains and restoring power to the affected train.

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that 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 because of:

R-12

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while 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 for restoring power to the affected train; and

c. The potential for an event in conjunction with a single failure of a redundant component.

~~The 2 hour Completion Time for DC buses is consistent with Regulatory Guide 1.93 (Ref. 3).~~

ACTIONS

~~C.1~~ (continued)

PA3.8-169

The second Completion Time for Required Action B~~C~~.2~~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 B~~C~~ is entered while, for instance, an AC bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met

R-12

(continued)

PA3.8-100

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 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 B~~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 R-12

If the inoperable distribution subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 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 plant systems.

E.1

Condition E addresses ~~With~~ two trains with inoperable distribution subsystems that result in a loss of safety function, adequate core cooling, containment OPERABILITY and other vital functions for DBA mitigation would be compromised~~,~~. Condition E also addresses two R-12 CL3.8-214

PA3.8-100

BASES (continued)

or more Reactor Protection Instrument AC Panels inoperable. If the plant is in this Condition, an~~d~~ immediate plant shutdown in accordance with LCO 3.0.3 is required. R-12

SURVEILLANCE REQUIREMENTS

SR 3.8.9.1

This Surveillance verifies that the ~~[~~required~~]~~ safeguards AC, DC, and Reactor Protection Instrument AC ~~vital bus~~ electrical power distribution systems, presented in Table B.3.8.9-1, are functioning properly, with the correct circuit breaker and switch alignment. The correct breaker and switch alignment ensures the appropriate separation and independence of the electrical divisions is maintained, and the appropriate voltage is available to each required subsystem ~~bus~~. The verification of proper voltage ~~availability on the buses~~ ensures that the required voltage is readily available for motive as well as control functions for critical system loads ~~connected to these buses~~. Various indications are available to the operators which demonstrate correct voltage for the subsystems. The 7 day Frequency takes into account the redundant capability of the safeguards AC, DC, and Reactor Protection Instrument AC ~~vital bus~~ electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

1. U~~F~~SAR, Section 8~~Chapter [6]~~.

2. U~~F~~SAR, Section 14~~Chapter [15]~~. CL3.8-172

~~3. Regulatory Guide 1.93, December 1974.~~

PA3.8-100

~~Table B 3.8.9-1 (page 1 of 1)~~
~~AC and DC Electrical Power Distribution Systems~~

~~TYPE~~	~~VOLTAGE~~	~~TRAIN A *~~	~~TRAIN B*~~
~~AC 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]~~ ~~Bus [NK03]~~ ~~Distribution Panels [NK41, NK43, NK51]~~	~~Bus [NK02]~~ ~~Bus [NK04]~~ ~~Distribution Panels [NK42, NK44, NK52]~~
~~AC vital buses~~	~~[120 V]~~	~~Bus [NN01]~~ ~~Bus [NN03]~~	~~Bus [NN02]~~ ~~Bus [NN04]~~

~~* Each train of the AC and DC electrical power distribution systems is a subsystem.~~

Table B 3.8.9-1 (page 1 of 1)
Safeguards AC and DC Electrical Power Distribution Systems

TYPE	DISTRIBUTION EQUIPMENT	UNIT 1 TRAIN A AND B	UNIT 2 TRAIN A AND B
Safeguards AC	4 kV Buses	15, 16	25, 26
	480 V Buses	111, 112, 121, 122	211, 212, 221, 222
	Motor Control Centers	1A1, 1A2 1AB1*, 1AB2* 1AC1, 1AC2 1K1, 1K2, 1KA2 1L1, 1L2 1LA1, 1LA2 1M1, 1M2 1MA1*, 1MA2* 1R1, 1S1 1T1*, 1T2* 1TA1, 1TA2 1X1, 1X2	2A1, 2A2 1AB1*, 1AB2* 2AC1, 2AC2 2K1, 2K2, 2KA2 2L1, 2L2 2LA1, 2LA2 2M1, 2M2 1MA1*, 1MA2* 2R1, 2S1 1T1*, 1T2* 2TA1, 2TA2 2X1, 2X2
Safeguards DC	125 VDC Panels	11, 12 15, 16 14*, 19* 17*, 18* 151, 161 152, 162 153, 163 191	21, 22 25, 26 14*, 19* 17*, 18* 27, 28 251, 261 252, 262 253, 263
Reactor Protection Instrument AC	120 VAC Panels	111, 112, 113, 114	211, 212, 213, 214

* Denotes MCC's or Panels that are transferrable between units.

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Difference Category	Difference Number 3.8-	Justification for Differences
	159	Not used.
CL	160	Additional discussion of the diesel fuel oil storage system is provided since Prairie Island has a unique design that involves many tanks, different systems for the emergency diesel generators (EDGs) for each unit and sharing of the storage capacity between the Unit 1 EDGs and the diesel driven cooling water pumps.

Difference Category	Difference Number 3.8-	Justification for Differences
PA	161	NUREG-1431, Rev. 2, LCO 3.8.6 Condition E and associated Bases have been revised by deleting the verbiage of one or more batteries. PI only has two batteries, one on each train. Therefore, PI physically cannot have more than one battery on a redundant train. In addition, the associated Completion Time has also been changed to 8 hours. The 8 hours is acceptable based on PI design, and in accordance with CTS which allows the batteries to be inoperable for 8 hours before initiating a reactor shutdown. PI has installed batteries that far exceed the loads they would be required to provide in the event of an accident. Due to being oversized, if Condition E were entered, the batteries may still be capable of performing their intended safety function. Based on the above, requiring both batteries to be declared inoperable, in a time less than 8 hours, would be inappropriate.
	162	Not used.

Difference Category	Difference Number 3.8-	Justification for Differences
CL	163	NUREG-1431, Rev 1, Bases 3.8.1 and 3.8.9, and NUREG-1431, Rev. 2, Bases 3.8.4 and 3.8.6, Applicable Safety Analyses Section have been revised by deleting, "or all onsite AC power". PI Safety Analysis for this system does not assume a loss of all onsite power. Therefore, this statement is deleted to be consistent with PI Safety Analysis.
	164	Not used.

Difference Category	Difference Number 3.8-	Justification for Differences
	165	Not used.
CL	166	NUREG-1431, Rev. 2, Bases 3.8.6, Action B has been revised deleting the following phrase, " ... and likely cannot perform its required design functions." This statement is not applicable to PI. At PI, even though the battery charger is operating in the current limit mode for 2 hours, the battery will still be able to perform its intended function. Therefore, this statement is deleted.
CL	167	NUREG-1431 Bases 3.8.9, has been revised throughout providing additional information or deleting detail in the Bases to make them more applicable to PI design, operations, and testing. For example, in the LCO Section, deleted the paragraph discussing tie breakers between redundant trains since PI design does not include tie breakers between the trains.

Difference Category	Difference Number 3.8-	Justification for Differences
PA	168	NUREG-1431 SR 3.8.10.1 has been revised by adding the following, "Verify correct breaker "and switch" alignments ... ." Adding switches makes this a more accurate SR for the PI design. PI has both breakers and switches in the safeguards AC, DC, and Reactor Protection Instrument AC electrical power distribution subsystems. This change is consistent with PI design and current operating practices.
PA	169	NUREG-1431 Bases 3.8.9, Action C.1, is being revised by deleting the following sentence, "The 2 hour Completion Time for DC buses is consistent with Regulatory Guide 1.93 (Ref. 3)." This sentence is being deleted because it is not referenced in the subject Regulatory Guide; in addition, reference JFD CL3.8-172 for PI position on Regulatory Guides.
	170	Not used.

Difference Category	Difference Number 3.8-	Justification for Differences
CL	171	NUREG-1431, Rev. 2, LCO 3.8.4, Required Actions A.4 and B.4 Completion Times and their associated Bases have been revised to 8 hours to be consistent with CLB as in the PI CTS 3.7.B.7 and 3.7.B.8. In addition, LCO 3.8.5 Required Action A.3 and LCO 3.8.6 Required Actions A.1, A.2, and B.1 have been changed to be consistent with LCO 3.8.4 Required Actions A.4 and B.4. Maintaining CLB was agreed to be acceptable between the industry and NRC during the onset of the ITS conversion project. This change is consistent with that agreement.
CL	172	NUREG-1431 Bases 3.8 has been revised deleting references to specific Regulatory Guides, IEEE Standards, and 10 CFR 50 criteria that PI is either not committed or designed to. PI was designed, built, and licensed prior to 10 CFR 50 Appendix A GDC and other noted NRC/industry design criteria. Where specific Industry Standards or Regulatory Guides are referenced, within the ITS, it does not mean PI is committing to them. They are only used as reference to support the ITS Bases or NRC criteria, Frequencies, SRs, etc., that are consistent with PI CLB.
CL	173	NUREG-1431, Rev. 2, Bases 3.8.4, Background Section, has been revised by changing the information about the "spare" battery charger to be applicable to the PI "portable" charger design and usage. At PI, there is a portable battery charger that may be moved into place to be used in either unit. This portable charger has been approved in the PI initial SER.

Difference Category	Difference Number 3.8-	Justification for Differences
	174	Not used.
TA	175	NUREG-1431 LCO 3.8.5, 3.8.8, and associated Bases have been revised consistent with the guidance of TSTF-204, Rev. 3.
PA	176	NUREG-1431 Bases 3.8.8, LCO Section has been revised by deleting the first sentence. This is consistent with TSTF-204, Rev. 3, which clarifies that safety analyses for Shutdown MODES operation does not consider Operating DBA's. The sentence is not consistent with PI CLB since PI does not currently have Shutdown Technical Specifications.
CL	177	NUREG-1431 Bases 3.8.8 has been revised providing additional information and clarification consistent with PI design, terminology, and operating practices, since PI does not currently have Shutdown Technical Specifications. This clarification takes TSTF-204, Rev. 3, into account.

Difference Category	Difference Number 3.8-	Justification for Differences
	181	Not used.
CL	182	NUREG-1431, Rev. 2, SR 3.8.4.2 and associated Bases, have been revised replacing the statement, "largest combined demands of the various continuous steady state loads ... ." with the statement, " ... demands of the various continuous steady state loads, ..." PI battery charger design requirements were based on the demands of the various continuous steady state loads not the largest combined demands of the of the various continuous steady state loads. Revising this statement as proposed, brings the SR into agreement with the PI design and consistent with the PI USAR.

Difference Category	Difference Number 3.8-	Justification for Differences
CL	183	NUREG-1431 LCO 3.8.7 requires four inverters to be OPERABLE when in MODES 1, 2, 3, or 4. LCO 3.8.7, Condition A, provides Required Actions A.1 and A.2 in the event that one Reactor Protection Instrument AC inverted is inoperable. In this case, Required Action A.1 requires verifying only one Reactor Protection Instrument AC panel is powered from Panel 117. Required Action A.2 requires verification that only one Reactor Protection Instrument AC Panel is powered from its inverter bypass source. These Required Actions verify and ensure that only one Reactor Protection Instrument AC Panel is powered from Panel 117 (Unit 2 – 217) or powered from its bypass source in accordance with PI design. PI design does not allow long term operation with one Reactor Protection Instrument AC Panel powered from Panel 117 and at the same time another Reactor Protection Instrument AC Panel supplied power from its bypass source. Required Action A.1 requires verifying only one Reactor Protection Instrument AC panel is powered from Panel 117. Alternatively, Required Action A.2 requires verification that only one Reactor Protection Instrument AC Panel is powered from its inverter bypass. In the Condition when two Reactor Protection Instrument AC inverters are inoperable, PI would enter Condition B which requires verification that at most one Reactor Protection Instrument AC panel is powered from Panel 117 (217) and one or both Reactor Protection Instrument AC panel(s) are powered from their inverter bypass source. In this case, one inverter would have to be restored in 8 hours.

Difference Category	Difference Number 3.8-	Justification for Differences
CL	183	(continued) Required Action A.1 (ITS Required Action B.3) Completion Time and associated Bases have been decreased from 24 hours to 8 hours. The decrease in Completion Time to 8 hours is consistent with the CTS. Both the new Required Action A.1 and A.2 are CLB and have been incorporated based on PI design. Maintaining CLB was agreed to be acceptable between the industry and NRC during the onset of the ITS conversion project. This change is consistent with that agreement.
	184	Not used.

Difference Category	Difference Number 3.8-	Justification for Differences
PA	185	NUREG-1431 Bases 3.8.7 LCO Section contains an explanation of the Note which allows an instrument bus inverter to be disconnected from its associated DC bus for up to 24 hours while performing an equalizing charge on the battery. The inverters used at PI are not required to be disconnected during equalizing charges. Therefore, this Note has been deleted consistent with ITS.
	186	Not used.
PA	187	NUREG-1431, Rev. 2, LCO 3.8.6 and associated Bases, Completion Time for Required Action B.2 has been revised to 24 hours. PI does not have either this Required Action or Completion Time in the CTS. The 24 hours is in accordance with PI USAR 8.5.2, which states, "Each of the four battery chargers has been sized to recharge its associated partially discharged battery within 24 hours, while carrying its normal load." Therefore, the 24 hours is consistent with PI licensing basis.
	188	Not used.
	189	Not used.

Difference Category	Difference Number 3.8-	Justification for Differences
	193-196	Not used.
CL	197	NUREG-1431 LCO SR 3.8.1.5 and associated Bases have been deleted. PI day tanks are not designed with any type of drain in the tank that would allow draining any water. PI operating history has shown that the day tanks have not had any water accumulation problems. In addition, neither PI CTS or CLB require checking the day tanks for water; therefore, this SR is being deleted.

Difference Category	Difference Number 3.8-	Justification for Differences
PA	198	NUREG-1431, Rev. 2, LCO 3.8.5, Condition A, and associated Bases, have been revised to delete the Condition phrase, "The redundant train battery and charger[s] OPERABLE." Per PI design, two trains of DC power are not required to be OPERABLE to support plant DC shutdown requirements as identified by LCO 3.8.10. Therefore, this part of Condition A does not apply to PI and is being deleted. In addition, the word "required" has been added as appropriate. Since PI has two trains, with each train consisting of a battery, battery charger, and interconnecting cable, it is necessary for clarification to state the "required" battery therefore, no mistake can be made on which battery charger is being credited when the plant is in the shutdown condition. ISTS LCO 3.8.5, Required Actions A.1 and A.2 have also been deleted to be consistent with LCO 3.8.4, Required Actions A.1 and A.2. PI design has a portable battery charger which can be used to replace the inoperable battery charger. The ITS allows 8 hours for the portable charger to be installed or declare the inoperable charger to OPERABLE. In either event, since PI has 8 hours to get a charger OPERABLE, Required Actions A.1 and A.2 would not be applicable and are therefore deleted.

Difference Category	Difference Number 3.8-	Justification for Differences
PA	199	NUREG-1431,Rev. 2, LCO 3.8.5, Required Action B.1, and associated Bases, have been deleted. This Required Action requires declaring the affected required features(s) inoperable. This action is only applicable if there were more than one DC electrical power subsystems required to be OPERABLE. Since PI design and shutdown operations do not require more than one DC electrical power subsystem to be OPERABLE, this Required Action does not apply.
CL	200	NUREG-1431 Bases 3.8 has been revised to reflect current PI design and operating practices. As an example, Bases 3.8.1, Required Action B.2 states, "This includes motor driven auxiliary feedwater pumps. Single train systems, such as turbine driven auxiliary feedwater pumps, are not included." PI has two 100% capacity auxiliary feedwater pumps, a motor and a turbine driven. The turbine driven auxiliary feedwater pump is not supported by the DG. Therefore, this statement is not applicable to PI design and is deleted.

Difference Category	Difference Number 3.8-	Justification for Differences
CL	201	NUREG-1431 Bases 3.8.1, Background Section has been revised by adding the statement, "... the Unit 1 DGs meet the intent of Safety Guide 9 and Unit 2 DGs satisfy the intent of Regulatory Guide 1.9, ..." This statement was added to reflect the differences between the two unit DGs. Unit 1 DGs were installed prior to the issuance of Regulatory Guide 1.9. Therefore, Unit 1 DGs rating were consistent with Safety Guide 9. When Unit 2 DGs were installed, Regulatory Guide 1.9 has been issued; however, PI did not adopt this Regulatory Guide in its entirety as discussed in the PI USAR. This change is consistent with the PI CLB.
CL	202	NUREG-1431 Bases 3.8.1 and 3.8.2, LCO Section have been revised by replacing the statement, "This will be accomplished ... ." with "The DG will be ready to load ... following receipt of a start signal." PI design is that each DG is capable of starting, accelerating to the required speed and voltage, and ready to be loaded within 10 seconds. PI DGs are not required to be loaded within 10 seconds. In addition, Bases 3.8.2, LCO statement has been revised by deleting the statement, "This sequence must be accomplished within [10] seconds." As stated above, the PI DGs are required to be ready to load within 10 seconds upon receipt of a start signal. Therefore, the Bases is revised to reflect the PI design and CLB.

Difference Category	Difference Number 3.8-	Justification for Differences
PA	203	NUREG-1431, Rev. 2, Bases 3.8.4 Actions B Section has been revised deleting the following sentence, "In addition the energization transients of any DC loads that are beyond the capability of the battery charger[s] and normally require the assistance of the batter[y][ies] will not be able to be brought online." PI design does not have any energization transients that exceed the battery charger capacity. The PI battery chargers were designed and installed to handle any of the anticipated transients that they would experience. Therefore, this statement is not applicable to PI.
PA	204	NUREG-1431 Bases 3.8.1, LCO Section has been revised by deleting the subject paragraphs. The subject paragraphs discuss various information about the AC sources in a train and the AC offsite sources being independent and separated to the extent practical. PI USAR provides a detailed discussion about the design of the AC trains and offsite sources; therefore, this redundant information is not needed in the TS and is being deleted. Also, Bases SR 3.8.1.1 has been revised by editing the sentence discussing preferred power source. PI design does not identify a preferred power source. The correct plant terminology is offsite power source.

Difference Category	Difference Number 3.8-	Justification for Differences
PA	205	NUREG-1431 Bases 3.8.1, 3.8.4, 3.8.7, and 3.8.9 Applicability Sections have been revised by deleting the following, " ...or abnormal transients;" PI considers an abnormal transient as an AOO. Therefore, the specific reference to an abnormal transient is being deleted.
CL	206	NUREG-1431 Bases 3.8.1, Condition C is for two paths inoperable. Required Action C.1 states to declare required feature(s) inoperable when its redundant required feature(s) is inoperable with a Completion Time of 12 hours. The ISTS states that the justification for the 12 hours is Regulatory Guide 1.93. PI CTS already has a Completion Time of 12 hours. Therefore, any references in the ISTS to the Completion Time being shorter or reduced is deleted.
CL	207	NUREG-1431 Bases 3.8.1, Required Action C.1 and C.2 have been revised by deleting the subject discussions since they are referring to Regulatory Guide 1.93. Since PI is not committed to Regulatory Guide 1.93, the subject discussions are not applicable to PI.
PA	208	NUREG-1431 Bases 3.8 has been revised deleting redundant information that also appears in the USAR.
	209	Not used.

Difference Category	Difference Number 3.8-	Justification for Differences
CL	210	NUREG-1431 Bases 3.8.2, LCO statement has been revised deleting the following, "It is acceptable for trains to be cross tied during shutdown conditions, allowing a single offsite power circuit to supply all required trains." PI design does not provide a cross tie between the trains. The design, as described in the USAR, provides for each offsite source being capable of supplying both trains, but this not termed a cross tie.
PA	211	NUREG-1431 LCO 3.8.2 and associated Bases has been revised by adding a Note allowing the LCO not being applicable for a period of 8 hours during the performance of SR 3.8.1.10. Without the Note, the LCO requires that one DG capable of supplying one train of the onsite 4 kV safeguards distribution system required by LCO 3.8.10 be OPERABLE. SR 3.8.2.1 requires the SRs of Specification 3.8.1 be performed at their specified Frequencies for those AC sources that are required to be OPERABLE to support those systems operating during plant shutdown. One of these SRs requires DG testing. At PI, when a DG is being tested, and thus operating, it is considered to be inoperable since during this testing some controls must be placed in manual. SR 3.8.1.10 in particular results in considering both DGs inoperable during test performance. The 8 hour period is reasonable to allow performance of the required SR.

Difference Category	Difference Number 3.8-	Justification for Differences
PA	212	NUREG-1431, Rev. 2, LCO 3.8.5 and associated Bases Background Section has been revised by adding a NOTE stating, service building DC electrical power subsystem components may be used in lieu of a safeguards DC electrical power subsystem component when the required safeguards DC electrical power subsystem is inoperable due to testing, maintenance, or replacement. PI design comprises of one battery, battery charger, and interconnecting cabling for each train. Since PI only has two trains of safeguards DC electrical power, during an outage only one train is required to be OPERABLE to support plant operations. The other train may be inoperable. At times during the outage, one train will be inoperable with the other needing testing, or even replacement. Based on new shutdown requirements, one train must remain OPERABLE, therefore requiring an extension in the outage schedule in order to accomplish needed maintenance, testing or replacement. PI design has two service building DC electrical power subsystems from which components may be used in lieu of either safeguards DC electrical power subsystem components. This is acceptable since the service building batteries are maintained in accordance with TS 5.5.15, Battery Testing Program, and the service building DC electrical power components will be maintained the same as the safeguards DC electrical power subsystem components.

Difference Category	Difference Number 3.8-	Justification for Differences
PA	212	(continued) Plant procedures will ensure that the DC electrical power subsystem components will perform their intended safety function. The time in which the service building DC electrical subsystem components can be used in lieu of the safeguards DC electrical power will be limited to the time the safeguards DC electrical power subsystem components are inoperable due to maintenance, testing, or replacement.
CL	213	NUREG-1431 LCO 3.8.9, Required Actions A, B, and C and their associated Bases have been revised by adding the following, "Declare associated required supported feature(s) inoperable, immediately." This Action needed to be added to provide guidance for when a portion of safeguards AC, DC, and Reactor Protection Instrument AC electrical power distribution subsystems are inoperable. This Condition is not covered by the ITS. This change is consistent with PI practices. In addition, a Note was added stating to enter the applicable Condition and Required Actions of LCO 3.8.4 for DC trains made inoperable by inoperable power distribution subsystems. This Note provides clarification for the operator. This change is consistent with NUREG-1431, Rev. 2.

Difference Category	Difference Number 3.8-	Justification for Differences
CL	214	NUREG-1431 LCO 3.8.9 Required Action E has been revised by adding the following, "Two or more Reactor Protection Instrument AC Panels inoperable, Enter LCO 3.0.3, Immediately." This Required Action has been added to provide specific Actions when two or more Reactor Instrument AC panels are inoperable, since the instrument AC panels are distinct from "Two trains...". The ISTS does not currently specify this condition.
PA	215	NUREG-1431, Rev. 1 LCO 3.8.2, Required Action A.1, LCO 3.8.8, Required Action A.1 and associated Bases have been deleted. The rational for the subject Required Actions A.1 was based on NUREG-1431, Rev.1 which would, in certain conditions, require more than one safeguards bus or inverter required to be OPERABLE. With one of two or more required safeguards bus or inverter inoperable, the remaining safeguards bus(s) or inverter(s) might be able to power all necessary loads. In such a case, it is acceptable to declare inoperable required features associated with the inoperability. However, with only one safeguards bus or inverter required, the above conditions do not exist, and the option to declare required features inoperable is not appropriate. Therefore, Required Action A.1 is being deleted.

Difference Category	Difference Number 3.8-	Justification for Differences
PA	218	NUREG-1431 LCO 3.8.3, Required Action D, has been revised by a Note stating, "Enter applicable Conditions and Required Actions of LCO 3.7.8, 'CL System' when Condition D is entered as a result of stored fuel oil properties not within limits". The requirements for diesel fuel oil volume have been divided into two separate specifications. ITS 3.7.8 provides diesel fuel oil volume for the diesel driven CL pumps and ITS 3.8.3 provides the requirements for the diesel generators. In addition, ITS 5.5.11 provides the requirements for the Diesel Fuel Oil Testing Program. Both the diesel driven CL pumps and the plant diesel generators share a common storage tank and fuel oil contents. ITS 3.8.3 provides requirements for testing the tank contents and associated Required Actions if the fuel oil properties are not restored to within limits. A Note was added to Condition D instructing the operators that if the diesel fuel oil in the storage tanks is not within limits, to enter the associated Conditions and Required Actions for the diesel driven CL pumps. This change provides consistency between the two systems and is consistent with current plant design and practices.

M - More restrictive (GENERIC NSHD)
(M3.8-04, M3.8-06, M3.8-14, M3.8-18, M3.8-19, M3.8-21, M3.8-24, M3.8-27, M3.8-31, M3.8-41, M3.8-42, M3.8-47, M3.8-49, M3.8-50, M3.8-52, M3.8-55, M3.8-64, M3.8-65, M3.8-66, M3.8-67)

This proposed TS revision involves modifying the CTS to impose more stringent requirements upon plant operations to achieve consistency with the guidance of NUREG-1431, correct discrepancies or remove ambiguities from the specifications. These more restrictive TSs have been evaluated against the plant design, safety analyses, and other TS requirements to ensure the plant will continue to operate safely with these more stringent specifications.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed changes provide more stringent requirements for operation of the plant. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event.

   These more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed changes do not involve any physical alteration of the plant, that is, no new or different type of equipment will be installed, nor do they change the methods governing normal plant operation.

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
4.6.B.3			Relocated - TRM	
4.6.B.4		(Partial)	Relocated - TRM	
4.6.B.4			3.8.6.6	
4.6.B.5			Deleted	
New		SR	3.8.4.2	
New		SR	3.8.4.3	
New		SR	3.8.7.1	
4.6.C		SR	3.4.9.2	
4.6.C		(Partial)	Relocated - Bases	

Improved Technical Specification Cross-Reference

ITS Section	ITS Table Item Number	Section Type	CTS Section	CTS Table Item Number
3.8.4		LCO	3.7.B.7	
3.8.4		LCO	3.7.B.8	
3.8.4.1		SR	New	
3.8.4.2		SR	New	
3.8.4.3		SR	New	
3.8.5		LCO	New	
3.8.5.1		SR	New	
3.8.6		LCO	New	
3.8.6.1		SR	New	
3.8.6.2		SR	4.6.B.1	
3.8.6.3		SR	4.6.B.2	
3.8.6.4		SR	4.6.B.1	
3.8.6.5		SR	New	
3.8.6.6		SR	4.6.B.4	
3.8.6.6		SR	New	
3.8.7		LCO	3.7.A	
3.8.7		LCO	3.7.A.7	
3.8.7		LCO	3.7.B	
3.8.7		LCO	New	

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.4.2	----------NOTE---------- Not required to be met for containment purge (high flow) and inservice (low flow) purge valve(s) in penetrations closed to comply with LCO 3.9.4.c.1. ------------------------------ Verify each required containment purge (high flow) and inservice (low flow) purge system valve actuates to the isolation position on an actual or simulated actuation signal.	24 months

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 Initiate action to satisfy RHR loop requirements.	Immediately
	AND	
	A.4 Close equipment hatch and secure with four bolts.	4 hours
	AND	
	A.5 Close one door in each airlock.	4 hours
	AND	
	A.6.1 Close each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, or blind flange.	4 hours
	OR	
	A.6.2 Verify each penetration is capable of being closed by an OPERABLE Containment Ventilation Isolation System.	4 hours

BASES

BACKGROUND (continued)

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 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 movement of irradiated fuel assemblies within containment, containment closure or closure capability is required; therefore, the door interlock mechanism may remain disabled and both doors may be open provided one door can be closed with at least two containment fan coil unit fans capable of operating in high speed.

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will restrict fission product radioactivity release from containment to be within regulatory limits.

The Containment Purge and Exhaust System includes two subsystems, Containment Purge and Containment Inservice Purge. The containment purge subsystem includes a 36 inch purge penetration and a 36 inch exhaust penetration. The second subsystem, a minipurge system referred to as containment inservice purge, includes a 14 inch purge penetration and an 18 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, or the penetrations may be blank flanged. The two valves in each of the two containment inservice purge penetrations can be opened intermittently, but are closed automatically by the Containment Ventilation Isolation System.

BASES

BACKGROUND (continued)

In MODE 6, sufficient air flow rates are necessary to conduct refueling operations. The inservice purge system is used for this purpose, and each of the four valves is closed by the radiation monitors associated with the containment inservice purge system in accordance with LCO 3.3.5, "Containment Ventilation Isolation Instrumentation." The 36 inch subsystem is normally blank flanged, although the option for use is allowed during outages, except during movement of irradiated fuel or CORE ALTERATIONS. All four containment purge valves are also closed by the Containment Ventilation Isolation System.

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, or blind flange.

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. 1). Fuel handling accidents include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.2, "Refueling Cavity Water Level," in conjunction with the minimum decay time of 100 hours prior to irradiated fuel movement with containment closure capability 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. The acceptance limit for offsite radiation exposure is 25% of 10 CFR 100 values.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will restrict fission product release from containment to be well within regulatory limits. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.

A fuel handling accident does not cause containment pressurization; however, with an assumed single failure, the operating purge system supply fan is assumed to continue supplying air to containment. To maintain post-fuel handling accident releases well within the limits of 10 CFR 100, only the inservice purge system is allowed to be operating during fuel movement. Two fan coil unit fans are required to operate in the high speed mode following a fuel handling accident to assure that radioactive material in containment is well mixed and any releases will leave containment at a lower concentration over the duration of the accident. The provision that one air lock door is OPERABLE and under procedural control will assure that at least one door will be closed within 30 minutes as required, thus assuring radioactive releases are well within the limits of 10 CFR 100 (Ref. 1).

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36 (c) (2)(ii).

LCO

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 containment penetrations to meet the following requirements:

BASES

LCO (continued)

a. The equipment hatch is closed and held in place by at least 4 bolts;

b. One door in each air lock is closed, or both doors in each air lock may be open with:

   1. containment (high flow) purge system isolated,

   2. one air lock door OPERABLE and capable of being closed within 30 minutes, and

   3. at least two containment fan coil unit fans capable of operating in the high speed mode; and

c. At least one isolation valve in each penetration, including the containment (high flow) purge system and inservice (low flow) purge system, providing direct access from the containment atmosphere to the outside atmosphere is either:

   1. OPERABLE or closed by a manual valve, or blind flange, or

   2. capable of being closed by an OPERABLE Containment Ventilation Isolation System.

   A penetration with direct access from the containment atmosphere to the outside atmosphere includes all penetrations that have a flow path that leads anywhere outside containment.

The containment air lock doors may be open during movement of irradiated fuel in the containment provided that the LCO requirements are met. These requirements include one door

BASES

LCO (continued)

OPERABLE, under procedural control and capable of being closed within 30 minutes following a fuel handling accident in containment and at least two fan coil unit fans are capable of operating in the high speed mode following a fuel handling accident in containment. Should a fuel handling accident occur inside containment, the fan coil unit fans will be operated in high speed and one door in each air lock will be closed following an evacuation of containment.

For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the Containment Ventilation Isolation System. The OPERABILITY requirements for this LCO require that the automatic purge and exhaust valve closure 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 movement of irradiated fuel assemblies within containment because this is when there is a potential for the limiting fuel handling accident.

In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1.

In MODES 5 and 6, when movement of irradiated fuel assemblies within containment is 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.

BASES (continued)

ACTIONS

A.1

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 Ventilation Isolation System not capable of automatic actuation when the purge and exhaust valves are open, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a fuel assembly 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 that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves will function if required during a fuel handling accident. 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 Ventilation Isolation signal.

The Surveillance is performed every 7 days during 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. A surveillance is to be conducted before the start of refueling operations and then in accordance with the frequency specified. 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 significant fission product radioactivity to the environment.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 24 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. In BASES LCO 3.3.5, the Containment Ventilation Isolation instrumentation requires a CHANNEL CHECK every 12 hours and a COT every 92 days to ensure the channel OPERABILITY during refueling operations. Every 24 months a CHANNEL CALIBRATION is performed. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances, when performed, will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

The SR is modified by a Note stating that this Surveillance is not required to be met for valves in isolated penetrations. The LCO provides the option to close penetrations in lieu of requiring automatic actuation capability.

REFERENCES

1. USAR, Section 14.5.

~~TS.3.8-2~~
~~Overflow~~
~~continued~~

~~3~~. If Specification 3.8.A.1.f or 3.8.A.1.g cannot be satisfied, all fuel handling operations in the core ~~containment~~ shall be

A3.9-47

LCO3.9.5
RA A.2

suspended (3.8.A.1.f (ITS 3.9.5) ,the requirements of

A3.9-47

LCO3.9.5
RA A.4,
A.5,A.6.1
LCO3.9.6
RA B.3,
B.4, B.5.1

Specification 3.8.A.1.a.1 (Close the equipment hatch and penetrations) shall be satisfied,

A3.9-50

at least one door in each personnel air lock shall be closed,

LCO3.9.5
RA A.1
LCO3.9.6
RA B.1

and no reduction in reactor coolant boron concentration less than required to meet LCO 3.9.1 shall be made.

A3.9-48

M3.9-51

LCO3.9.5
RA A.6.2
LCO3.9.6
RA B.5.2

Verify each penetration is capable of being closed by an OPERABLE containment ventilation isolation system.

R-12

R-10

NSHD Category	Change Number 3.9-	Discussion of Change
M	16	New SRs, 3.9.4.1 and 3.9.4.2 are included which require verification of containment penetration status every 7 days and verification of containment purge and inservice purge valve actuation every 24 months. The 7 day frequency for containment penetration status verification is commensurate with the normal duration of time to complete fuel handling operations. The 24 month Frequency for verification of containment purge and inservice purge valve actuation is consistent with a 24 month refueling outage interval and will allow this verification to be performed during each refueling outage. ITS SR 3.9.4.2 is modified by a note which does not require the SR to be met when containment purge and inservice purge valves are closed in compliance with LCO 3.9.4 requirements for penetrations to be isolated. This is an acceptable exception since penetrations that are closed in compliance with the LCO do not have to be tested to assure that they can be automatically closed. These are activities which are currently performed under plant procedures; therefore this change does not adversely impact plant operations. Since these will be formal TS required surveillances, this change is considered more restrictive. This change is included to make the PI ITS complete and consistent with the guidance of NUREG-1431.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1	~~Suspend CORE ALTERATIONS.~~	Immediately TA3.9-66
	~~AND~~		
	~~A.2~~	Suspend movement of irradiated fuel assemblies within containment.	~~Immediately~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.4.2	-------------------NOTE------------------- Not required to be met for containment purge (high flow) and inservice (low flow) purge valve(s) in penetrations closed to comply with LCO 3.9.4.c.1. --- Verify each required containment purge (high flow) and inservice (low flow) purge system~~and exhaust~~ valve actuates to the isolation position on an actual or simulated actuation signal.	TA3.9-67 PA3.9-64 X3.9-61 R-12 R-12 24~~[18]~~ months

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.4	Close equipment hatch and secure with four bolts~~all containment penetrations providing direct access from containment atmosphere to outside atmosphere~~.	4 hours TA3.9-69
	AND		
	A.5	Close one door in each air lock.	4 hours
	AND		
	A.6.1	Close each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, or blind flange.	4 hours
	OR		
	A.6.2	Verify each penetration is capable of being closed by an OPERABLE Containment Ventilation Isolation System.	4 hours R-12

BASES

four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

BACKGROUND (continued)

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 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 or closure capability is required; therefore, the door interlock mechanism may remain disabled and both doors may be open provided one door can be closed with at least two containment fan coil unit fans capable of operating in high speed~~, but one air lock door must always remain closed~~. TA3.9-66 CL3.9-62

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 to be within regulatory limits~~due to a fuel handling accident during refueling~~. CL3.9-62

PA3.9-101

The Containment Purge and Exhaust System includes two subsystems, Containment Purge and Containment Inservice Purge. The containment purge~~normal~~ subsystem includes a 36~~42~~ inch purge penetration and a 36~~42~~ inch exhaust penetration. The second subsystem, a minipurge R-12

(continued)

BASES

system referred to as containment inservice purge, includes a~~n~~ 14~~8~~ inch purge penetration and an 18 inch exhaust penetration.

CL3.9-102

During MODES 1, 2, 3, and 4, the two valves in each of the containment~~normal~~ purge and exhaust penetrations are ~~secured in the closed position~~blank flanged. The two valves in each of the two containment inservice purge~~ minipurge~~ penetrations can be opened intermittently, but are closed automatically by the Containment Ventilation Isolation System.~~ Engineered Safety Features Actuation System (ESFAS). Neither of the subsystems is subject to a Specification in MODE 5.~~ CL3.9-102

In MODE 6, sufficient air flow rates ~~large air exchangers~~ are necessary to conduct refueling operations. The ~~normal 42 inch~~ inservice purge system is used for this purpose, and ~~all~~ each of the four valves is~~are~~ closed by the radiation monitors associated with the containment inservice purge system in accordance with, LCO 3.3.5, "Containment Ventilation Isolation Instrumentation." ~~ESFAS in accordance with LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."~~ The 36 inch subsystem is normally blank flanged, although the option for use is allowed during outages, except during movement of irradiated fuel or CORE ALTERATIONS. All four containment purge valves are also closed by the Containment Ventilation Isolation System. CL3.9-102 PA3.9-101

~~The minipurge system remains operational~~ PA3.9-101 R-12
~~in MODE 6, and all four valves are also closed by the ESFAS. or The minipurge system is not used in MODE 6. All four 8 inch valves are secured in the closed position.~~

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one

(continued)

BASES

a. The equipment hatch is closed and held in place by at least 4 bolts;

b. One door in each air lock is closed, or both doors in each air lock may be open with: CL3.9-62

   1. containment (high flow) purge system isolated,

   2. one air lock door OPERABLE and capable of being closed within 30 minutes, and

   3. at least two containment fan coil unit fans capable of operating in the high speed mode; and

c. At least one isolation valve in each penetration, including the containment (high flow) purge system and inservice (low flow) purge system, providing direct access from the containment atmosphere to the outside atmosphere is either:

   1. OPERABLE or closed by a manual valve, or blind flange, or

   2. capable of being closed by an OPERABLE Containment Ventilation Isolation System. PA3.9-64

A penetration with direct access from the containment atmosphere to the outside atmosphere includes all penetrations that have a flow path that leads anywhere outside containment. CL3.9-62

R-10

(continued)

BASES

The containment air lock doors may be open during movement of irradiated fuel in the containment provided that the LCO requirements are met. These requirements include one door OPERABLE, under procedural control and capable of being closed within 30 minutes following a fuel handling accident in containment and at least two fan coil unit fans are capable of operating in the high speed mode following a fuel handling accident in containment. Should a fuel handling accident occur inside containment, the fan coil unit fans will be operated in high speed and one door in each air lock will be closed following an evacuation of containment. CL3.9-62 PA3.9-64 R-10

For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the Containment Ventilation Isolation System. ~~Containment Purge and Exhaust Isolation System.~~ The OPERABILITY requirements for this LCO require~~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. R-12 R-12

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 the limiting~~a~~ fuel handling accident. TA3.9-66

In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1.

(continued)

BASES

In MODES 5 and 6, when ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment is ~~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. PA3.9-64

ACTIONS

A.1 ~~and A.2~~

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~~ Ventilation Isolation System not capable of automatic actuation when the purge and exhaust valves are open, the unit 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 completion of movement of a fuel assembly~~component~~ to a safe position. R-12

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1 PA3.9-64

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves will function if required during a fuel handling acciden~~tare not blocked from closing~~. Also the Surveillance will TA3.9-66 R-12

(continued)

BASES

SURVEILLANCE REQUIREMENTS

~~SR 3.9.4.1 (continued)~~

demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed by an OPERABLE automatic C~~c~~ontainment Ventilation~~purge and exhaust~~ I~~i~~solation signal.

TA3.9-66

PA3.9-64

The Surveillance is performed every 7 days during ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within

TA3.9-66

containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance is to be conducted before the start of refueling operation and then in accordance with the frequency specified~~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 significant fission product radioactivity to the environment.

TA3.9-66

CL3.9-105

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 24~~18~~ month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. In LCO 3.3.5~~6~~, the Containment Ventilation ~~Purge and Exhaust~~ Isolation instrumentation requires a CHANNEL CHECK every 12 hours and a COT every 92 days to ensure the channel OPERABILITY during refueling operations. Every 24~~18~~ months a CHANNEL CALIBRATION is

R-12

X3.9-61

(continued)

Difference Category	Difference Number 3.9-	Justification for Differences
	63	Not used.
PA	64	The PI name for the instrumentation system which automatically isolates containment ventilation during fuel handling is the "Containment Ventilation Isolation System" and the Specification for this system is 3.3.5, "Containment Ventilation Isolation Instrumentation". The systems which are isolated are the "containment purge (high flow) system" and containment inservice (low flow) purge system", thus these names are used in SR 3.9.4.2 and throughout the Bases as applicable. The parenthetical modifiers " (high flow)" and "(low flow)" may be included to assure that the operators do not confuse these systems.
CL	65	The "or equivalent" option in NUREG-1431 is not included in the PI ITS. The Specification and Bases have been revised. PI does not currently have this flexibility and the evaluations which support it have not been performed, thus this is not included.

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-1C	11	SR	3.3.4.1	
Table 4.1-1C	11	SR	3.3.4.2	
Table 4.1-1C	12		Deleted - Boric Acid LAR	
Table 4.1-1C	13		Relocated - TRM	
Table 4.1-1C	14		CTS Deleted	
Table 4.1-1C	15	TABLE	3.3.1-1	16.b.2
Table 4.1-1C	15		Relocated - TRM	
Table 4.1-1C	16		Relocated - TRM	
Table 4.1-1C	17		Relocated - TRM	
Table 4.1-1C	18	SR	3.3.1.12	
Table 4.1-1C	19		Relocated - TRM	
Table 4.1-1C	20		Relocated - TRM	
Table 4.1-1C	21	SR	3.3.3.1	
Table 4.1-1C	21	SR	3.3.3.2	
Table 4.1-1C	21	SR	3.3.3.3	
Table 4.1-1C	22		CTS Deleted	
Table 4.1-1C	23		CTS Deleted	
Table 4.1-1C	24		Relocated - TRM	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-1C	24	SR	3.3.6.5	
Table 4.1-1C	24	SR	3.3.6.2	
Table 4.1-1C	25	SR	3.4.12.4	
Table 4.1-1C	25	SR	3.4.12.5	
Table 4.1-1C	25	SR	3.4.13.5	
Table 4.1-1C	25	SR	3.4.13.6	
Table 4.1-1C	26		Relocated - TRM	
Table 4.1-1C	27		Relocated - TRM	
Table 4.1-1C	28		Relocated - TRM	
Table 4.1-1C	29	SR	3.3.3.1	
Table 4.1-1C	29	SR	3.3.3.2	
Table 4.1-1C	29	(Partial)	Relocated - TRM	
Table 4.1-1C	30		Relocated - Bases	
Table 4.1-1C	31		Relocated - TRM	
Table 4.1-1C	Note 30	SR	3.1.7.1	
Table 4.1-1C	Note 31		Deleted	
Table 4.1-1C	Note 32		Relocated - TRM	

5.5 Programs and Manuals (continued)

5.5.8 Steam Generator (SG) Tube Surveillance Program

Steam generator tubes in each unit shall be determined operable by the following:

a. Steam Generator Sample Selection and Inspection

Each steam generator shall be determined operable in accordance with the in-service inspection schedule in Specification 5.5.8.c. The in-service inspection may be limited to one steam generator on a rotating schedule encompassing 6% of the tubes in the single steam generator, provided the previous inspections indicated that the two steam generators are performing in a like manner.

b. Steam Generator Tube Sample Selection and Inspection

The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Tables 5.5.8-1 and 5.5.8-2. The in-service inspection of steam generator tubes shall be performed at the frequencies specified in Specification 5.5.8.c and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 5.5.8.d. The tubes selected for each in-service inspection shall include at least 3% of the total number of tubes in all steam generators and at least 20% of the total number of sleeves in service in both steam generators; the tubes selected for these inspections shall be selected on a random basis except:

1. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

2. The first sample of tubes selected for each in-service inspection (subsequent to the preservice inspection) of each steam generator shall include:

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)

(a) All tubes that previously had detectable wall penetrations (>20%) that have not been plugged or sleeve repaired in the affected area.

(b) Tubes in those areas where experience has indicated potential problems.

(c) A tube inspection (pursuant to Specification 5.5.8.d.1.(h)) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

3. In addition to the sample required in Specification 5.5.8.b.2(a) through (c), all tubes which have had the F* or EF* criteria applied will be inspected in the F* and EF* regions of the roll expanded region. The region of these tubes below the F* and EF* regions may be excluded from the requirements of Specification 5.5.8.b.2(a).

4. The tubes selected as the second and third samples (if required by Tables 5.5.8-1 or 5.5.8-2) during each in-service inspection may be subjected to a partial tube or sleeve inspection provided:

(a) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

(b) The inspections include those portions of the tubes or sleeves where imperfections were previously found.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

5. Indications left in service as a result of application of tube support plate voltage-based repair criteria shall be inspected by bobbin coil probe during all future refueling outages.

6. Implementation of the steam generator tube/tube support plate repair criteria requires a 100 percent bobbin coil inspection for hot leg and cold leg tube support plate intersections down to the lowest cold leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20 percent random sampling of tubes inspected over their full length.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)

c. Inspection Frequencies

The above required in-service inspections of steam generator tubes shall be performed at the following frequencies:

1. In-service inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

2. If the results of the in-service inspection of a steam generator conducted in accordance with Table 5.5.8-1 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.5.8.c.1; the interval may then be extended to a maximum of once per 40 months.

3. Additional, unscheduled in-service inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 5.5.8-1 during the shutdown subsequent to any of the following conditions:

   (a) Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.14.

   (b) A seismic occurrence greater than the Operating Basis Earthquake.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)

(c) A loss-of-coolant accident requiring actuation of the engineered safeguards.

(d) A main steam line or feedwater line break.

d. Acceptance Criteria

1. As used in this Specification:

(a) Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

(b) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

(c) Degraded Tube means a tube containing imperfections $\geq$ 20% of the nominal wall thickness caused by degradation.

(d) % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

(e) Defect means an imperfection of such severity that it exceeds the repair limit. A tube containing a defect is defective.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)

(f) Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving because it may become unserviceable prior to the next inspection and is equal to 50% of the nominal tube wall thickness. If significant general tube thinning occurs, this criteria will be reduced to 40% wall penetration. This definition does not apply to the portion of the tube in the tubesheet below the F* distance provided the tube is not degraded (i.e., no indications of cracks) within the F* or EF* distance for F* or EF* tubes. The repair limit for the pressure boundary region of any sleeve is 25% of the nominal sleeve wall thickness. This definition does not apply to tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to Specification 5.5.8.d.4 for the repair limit applicable to these intersections.

(g) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break.

(h) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

(i) Sleeving is the repair of degraded tube regions using a new Alloy 690 tubing sleeve inserted inside the parent tube and sealed at each end by welding or by replacing the lower weld in a full depth tubesheet sleeve with a hard rolled joint. The new sleeve becomes the pressure boundary spanning the original degraded tube region.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)

(j) F* Distance is the distance from the bottom of the hardroll transition toward the bottom of the tubesheet that has been conservatively determined to be 1.07 inches (not including eddy current uncertainty). The F* distance applies to roll expanded regions below the midplane of the tubesheet.

(k) F* Tube is a tube with degradation, below the F* distance, equal to or greater than 40%, and not degraded (i.e., no indications of cracking) within the F* distance.

(l) EF* Distance is the distance from the bottom of the upper hardroll transition toward the bottom of the tubesheet that has been conservatively determined to be 1.67 inches (not including eddy current uncertainty). EF* distance applies to roll expanded regions when the top of the additional roll expansion is 2.0 inches or greater down from the top of the tubesheet.

(m) EF* Tube is a tube with degradation, below the EF* distance, equal to or greater than 40%, and not degraded (i.e., no indications of cracking) within the EF* distance.

2. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair by sleeving all tubes exceeding the repair limit and all tubes containing through-wall cracks or classify as F* or EF* tubes) required by Tables 5.5.8-1 and 5.5.8-2.

3. Tube repair, after April 1, 1999, using Combustion Engineering welded sleeves shall be in accordance with the methods described in the following:

   CEN-629-P, Revision 03-P, "Repair of Westinghouse Series 44 and 51 Steam Generator Tubes Using Leak Tight Sleeves".

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)

4. Tube Support Plate Repair Limit is used for the disposition of a steam generator tube for continued service that is experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the repair limit is based on maintaining steam generator serviceability as described below:

   (a) Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltages less than or equal to 2.0 volts will be allowed to remain in service.

   (b) Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts, will be repaired or plugged, except as noted in Specification 5.5.8.d.4(c) below.

   (c) Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts but less than or equal to the upper voltage repair limit, may remain in service if a rotating pancake coil (or comparable examination technique) inspection does not detect degradation. Steam generator tubes, with indications of outside diameter stress corrosion cracking degradation with a bobbin voltage greater than the upper voltage repair limit will be plugged or repaired.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)

(d) If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits in Specifications 5.5.8.d.4(a), (b) and (c). The mid-cycle repair limits are determined from the following equations:

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr\left(\frac{CL - \Delta t}{CL}\right)}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - 2.0)\left(\frac{CL - \Delta t}{CL}\right)$$

Where:

V_{URL} = upper voltage repair limit

V_{LRL} = lower voltage repair limit

V_{MURL} = mid-cycle upper voltage repair limit based on time into cycle

V_{MLRL} = mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle

Δt = length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)

CL = cycle length (time between two scheduled steam generator inspections)

V_{SL} = structural limit voltage

Gr = average growth rate per cycle length

NDE = 95 percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20 percent has been approved by the NRC)

Implementation of these mid-cycle repair limits should follow the same approach as described in Specifications 5.5.8.d.4(a), (b) and (c).

Note: The upper voltage repair limit is calculated according to the methodology in Generic Letter 95-05 as supplemented.

5.5 Programs and Manuals (continued)

5.5.9 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems and the Spent Fuel Pool Special and Inservice Purge Ventilation System each operating cycle (18 months for shared systems).

Demonstrate for the Auxiliary Building Special Ventilation, Shield Building Ventilation, Control Room Special Ventilation, and Spent Fuel Pool Special and Inservice Purge Ventilation Systems that:

a. An inplace DOP test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 1% (for DOP, particles having a mean diameter of 0.7 microns);

b. A halogenated hydrocarbon test of the inplace charcoal adsorber shows a penetration and system bypass < 1% (for DOP, particles having a mean diameter of 0.7 microns);

c. 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 15% penetration (less than 5% penetration for the Control Room Special Ventilation System) when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and 95% relative humidity (RH) (or 70% RH with humidity controls if the humidity controls are capable of maintaining the humidity of the air entering the charcoal less than or equal to 70% RH under worst-case design-basis conditions); and

d. The pressure drop across the combined HEPA filters and the charcoal adsorbers is less than 6 inches of water at the system flowrate ±10%.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5 Programs and Manuals (continued)

5.5.10 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the waste gas holdup system, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

a. The limits for concentrations of 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;

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than or equal to 78,800 curies of noble gas (considered as dose equivalent Xe-133); and

c. A surveillance program to ensure that the quantity of radioactivity contained in each of the following tanks shall be limited to 10 curies, excluding tritium and dissolved or entrained noble gases:

   Condensate storage tanks
   Outside temporary tanks

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.

5.5 Programs and Manuals (continued)

5.5.11 Diesel Fuel Oil Testing Program

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 the limits specified in Table 1 of ASTM D975-77 when checked for viscosity, water, and sediment. Acceptability of new fuel oil shall be determined prior to addition to the safeguards storage tanks. Testing of diesel fuel oil stored in the safeguards storage tanks shall be performed at least every 31 days.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

5.5.12 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

    1. a change in the TS incorporated in the license; or

    2. a change to the USAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the USAR.

5.5 Programs and Manuals

5.5.12 Technical Specifications (TS) Bases Control Program (continued)

d. Proposed changes that meet the criteria of Specification 5.5.12 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 on a frequency consistent with USAR updates.

5.5.13 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

5.5 Programs and Manuals

5.5.13 Safety Function Determination Program (SFDP) (continued)

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the inoperable support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP 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. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.14 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

5.5 Programs and Manuals

5.5.14 Containment Leakage Rate Testing Program (continued)

b. The peak calculated containment internal pressure for the design basis loss of coolant accident is less than the containment internal design pressure, P_a, of 46 psig.

c. The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.25% of primary containment air weight per day. For pipes connected to systems that are in the auxiliary building special ventilation zone, the total leakage shall be less than 0.1% of primary containment air weight per day at pressure P_a. For pipes connected to systems that are exterior to both the shield building and the auxiliary building special ventilation zone, the total leakage past isolation valves shall be less than 0.01% of primary containment air weight per day at pressure P_a.

d. Leakage Rate acceptance criteria are:

1. Primary containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. Prior to unit startup, following testing in accordance with the program, the combined leakage rate acceptance criteria are $\leq 0.60\ L_a$ for all components subject to Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests.

2. Air lock testing acceptance criteria are:

   a) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at ≥ 46 psig.

   b) For each door intergasket test, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 10 psig.

5.5 Programs and Manuals

5.5.14 Containment Leakage Rate Testing Program (continued)

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

5.5.15 Battery Monitoring and Maintenance Program

This Program provides for restoration and maintenance of the 125V plant safeguards batteries and service building batteries, which may be used instead of the safeguards batteries during shutdown conditions in accordance with manufacturer's recommendations, as follows:

a. Actions to restore battery cells with float voltage < 2.13 V will be in accordance with manufacturer's recommendations, and

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the minimum established design limit.

Table 5.5.8-1
STEAM GENERATOR TUBE INSPECTION

<table>
<tr><th colspan="3">1st SAMPLE INSPECTION</th><th colspan="2">2nd SAMPLE INSPECTION</th><th colspan="2">3rd SAMPLE INSPECTION</th></tr>
<tr><th>Sample Size</th><th>Result</th><th>Action Required</th><th>Result</th><th>Action Required</th><th>Result</th><th>Action Required</th></tr>
<tr><td rowspan="9">A minimum of S Tubes per S.G.</td><td>C-1</td><td>None</td><td>N/A</td><td>N/A</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="5">C-2</td><td rowspan="5">Repair defective tubes and inspect additional 2S tubes in this S.G.</td><td>C-1</td><td>None</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="3">C-2</td><td rowspan="3">Repair defective tubes and inspect additional 4S tubes in this S.G.</td><td>C-1</td><td>None</td></tr>
<tr><td>C-2</td><td>Repair defective tubes</td></tr>
<tr><td>C-3</td><td>Perform action for C-3 result of first sample</td></tr>
<tr><td>C-3</td><td>Perform action for C-3 result of first sample</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="3">C-3</td><td rowspan="3">Inspect all tubes in this S.G., Repair defective tubes and inspect 2S tubes in each other S.G. Prompt notification to NRC</td><td>All other S.G.s are C-1</td><td>None</td><td>N/A</td><td>N/A</td></tr>
<tr><td>Some S.G.s C-2 but no additional S.G. are C-3</td><td>Perform action for C-2 result of second sample</td><td>N/A</td><td>N/A</td></tr>
<tr><td>Additional S.G. is C-3</td><td>Inspect all tubes in each S.G. and repair defective tubes. Prompt notification to NRC.</td><td>N/A</td><td>N/A</td></tr>
</table>

S=3%; When two steam generators are inspected during that outage.
S=6%; When one steam generator is inspected during that outage.

Table 5.5.8-2
STEAM GENERATOR TUBE SLEEVE INSPECTION

1st Sample Inspection			**2nd Sample Inspection**	
Sample Size	**Result**	**Action Required**	**Result**	**Action Required**
A minimum of 20% of Tube Sleeves (1)	C-1	None	N/A	N/A
	C-2	Inspect all remaining tube sleeves in this S.G. and plug or repair defective sleeved tubes.	C-1	None
			C-2	Plug or repair defective sleeved tubes
			C-3	Perform action for C-3 result of first sample
	C-3	Inspect all tube sleeves in this S.G., inspect 20% of the tube sleeves in the other S.G., and plug or repair defective sleeved tubes	The other S.G. is C-1	None
			The other S.G. is C-2	Perform action for C-2 result of first sample
			The other S.G. is C-3	Inspect all tube sleeves in each S.G. and plug or repair defective sleeved tubes

(1) Each type of sleeve is considered a separate population for determination of scope expansion

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Exposure Report

----------NOTE----------
A single submittal may be made for the plant. The submittal should combine sections common to both units.

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrem and the associated collective deep dose equivalent (reported in person-rem) according to work and job functions, e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, thermoluminescent dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totaling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

5.6 Reporting Requirements (continued)

5.6.2 Annual Radiological Environmental Monitoring Report

---NOTE---
A single submittal may be made for the plant. The submittal should combine sections common to both units.

The Annual Radiological Environmental Monitoring Report covering the operation of the plant 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 Monitoring Report shall include summarized and tabulated results, in the format of Regulatory Guide 4.8, December 1975, of all radiological environmental samples taken during the report period. 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.

The report shall also include the following: a summary description of the radiological environmental monitoring program; a map of sampling locations keyed to a table giving distances and directions from the reactor site; and the results of licensees participation in the Interlaboratory Comparison Program defined in the ODCM.

5.6 Reporting Requirements (continued)

5.6.3 Radioactive Effluent Report

---NOTE---
A single submittal may be made for the plant. The submittal shall combine sections common to both units.

The Radioactive Effluent Report covering the operation of the plant during the previous calendar year shall be submitted by May 15 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the plant. The material provided shall be consistent with the objectives outlined in the ODCM and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

   LCO 3.1.1, "SHUTDOWN MARGIN (SDM)";
   LCO 3.1.3, "Isothermal Temperature Coefficient (ITC)";
   LCO 3.1.5, "Shutdown Bank Insertion Limits";
   LCO 3.1.6, "Control Bank Insertion Limits";
   LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 2";

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)";
LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor $\left(F_{\Delta H}^{N}\right)$;
LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)";
LCO 3.4.1, "RCS Pressure, Temperature, and Flow - Departure from Nucleate Boiling (DNB) Limits"; and
LCO 3.9.1, "Boron Concentration".

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. NSPNAD-8101-PA, "Qualification of Reactor Physics Methods for Application to PI Units" (latest approved version);

2. NSPNAD-8102-PA, "Prairie Island Nuclear Power Plant Reload Safety Evaluation Methods for Application to PI Units"(latest approved version);

3. NSPNAD-97002-PA, "Northern States Power Company's "Steam Line Break Methodology", (latest approved version);

4. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology", July, 1985;

5. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code", August, 1985;

6. WCAP-10924-P-A, "Westinghouse Large Break LOCA Best-Estimate Methodology", December, 1988;

7. WCAP-10924-P-A, Volume 1, Addendum 4, "Westinghouse Large Break LOCA Best Estimate Methodology", August, 1990;

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

8. XN-NF-77-57 (A), XN-NF-77-57, Supplement 1 (A), "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors Phase II", May, 1981;

9. WCAP-13677, "10 CFR 50.46 Evaluation Model Report: W-COBRA/TRAC 2-Loop Upper Plenum Injection Model Update to Support ZIRLO$_{TM}$ Cladding Options", April 1993 (approved by NRC SE dated November 26, 1993);

10. NSPNAD-93003-A, "Transient Power Distribution Methodology", (latest approved version).

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 Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat-up, cooldown, low temperature operation, criticality, and hydrostatic testing, LTOP arming, PORV lift settings and Safety Injection Pump Disable Temperature as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.10, "Pressurizer Safety Valves";

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) – Reactor Coolant System Cold Leg Temperature (RCSCLT) > Safety Injection (SI) Pump Disable Temperature";
LCO 3.4.13, "Low Temperature Overpressure Protection (LTOP) – Reactor Coolant System Cold Leg Temperature (RCSCLT) ≤ Safety Injection (SI) Pump Disable Temperature"; and
LCO 3.5.3, "ECCS - Shutdown".

b. The analytical methods used to determine the RCS pressure and temperature limits and Cold Overpressure Mitigation System setpoints shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" (includes any exemption granted by NRC to ASME Code Case N-514).

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto. Changes to the curves, setpoints, or parameters in the PTLR resulting from new or additional analysis of beltline material properties shall be submitted to the NRC prior to issuance of an updated PTLR.

5.6 Reporting Requirements (continued)

5.6.7 Steam Generator Tube Inspection Report

1. Following each in-service inspection of steam generator tubes, if there are any tubes requiring plugging or sleeving, the number of tubes plugged or sleeved in each steam generator shall be reported to the Commission within 15 days.

2. The results of steam generator tube in-service inspections shall be included with the summary reports of ASME Code Section XI inspections submitted within 90 days of the end of each refueling outage. Results of steam generator tube in-service inspections not associated with a refueling outage shall be submitted within 90 days of the completion of the inspection. These reports shall include: (1) number and extent of tubes inspected, (2) location and percent of wall-thickness penetration for each indication of an imperfection, and (3) identification of tubes plugged or sleeved.

3. Results of steam generator tube inspections which fall into Category C-3 require notification to the Commission prior to resumption of plant operation, and reporting as a special report to the Commission within 30 days. This special report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

4. The results of inspections performed under Specification 5.5.8.b for all tubes that have defects below the F* or EF* distance, and were not plugged, shall be reported to the Commission within 15 days following the inspection. The report shall include:

   a. Identification of F* and EF* tubes, and

   b. Location and extent of degradation.

5.6 Reporting Requirements

5.6.7 Steam Generator Tube Inspection Report (continued)

5. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the NRC staff prior to returning the steam generators to service should any of the following conditions arise:

   a. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steamline break) for the next operating cycle.

   b. If circumferential crack-like indications are detected at the tube support plate intersections.

   c. If indications are identified that extend beyond the confines of the tube support plate.

   d. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

   e. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1×10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.

5.6 Reporting Requirements (continued)

5.6.8 EM Report

When a report is required by Condition C or J of LCO 3.3.3, "Event Monitoring (EM) Instrumentation," a report shall be submitted within the following 14 days. 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.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied in place of the controls required by paragraph 10 CFR 20.1601(a) and (b) of 10 CFR 20:

5.7.1 <u>High Radiation Areas accessible to personnel in which radiation levels could result in an individual receiving a deep dose equivalent **less than** 1.0 rem in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates</u>

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in each such area shall be controlled by means of a Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

   1. A radiation monitoring device that continuously displays radiation dose rates in the area; or

   2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint; or

5.7 High Radiation Area

5.7.1 High Radiation Areas accessible to personnel in which radiation levels could result in an individual receiving a deep dose equivalent **less than** 1.0 rem in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates (continued)

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area; or

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

    (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area, who is responsible for controlling personnel exposure within the area; or

    (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7 High Radiation Area (continued)

5.7.2 High Radiation Areas accessible to personnel in which radiation levels could result in an individual receiving a deep dose equivalent **in excess of** 1.0 rem in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates, but less than 500 rad in one hour at one meter from the source

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

   1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or their designee.

   2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

   1. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint; or

5.7 High Radiation Area

5.7.2 High Radiation Areas accessible to personnel in which radiation levels could result in an individual receiving a deep dose equivalent **in excess of** 1.0 rem in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates, but less than 500 rad in one hour at one meter from the source (continued)

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area; or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area, who is responsible for controlling personnel exposure within the area; or

   (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

4. In those cases where options (2) and (3), above are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device shall be used that continuously displays radiation dose rates in the area.

5.7 High Radiation Area

5.7.2 High Radiation Areas accessible to personnel in which radiation levels could result in an individual receiving a deep dose equivalent **in excess of** 1.0 rem in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates, but less than 500 rad in one hour at one meter from the source (continued)

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are located within a larger area where no enclosure exists for the purpose of locking and where no enclosure can be reasonably constructed around the individual area, that individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a flashing light shall be activated at the area as a warning device.

~~4.12 STEAM GENERATOR TUBE SURVEILLANCE~~ A5.0-04

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~~Applicability~~

~~Applies to inservice surveillance of the steam generator tubes.~~

~~Objective~~

~~To assure the continued integrity of the steam generator tubes that are a part of the primary coolant pressure boundary.~~

~~Specification~~

5.5.8 Steam generator tubes in each unit shall be determined operable by the following:

a~~A~~. Steam Generator Sample Selection and Inspection-Each steam generator shall be determined operable in accordance with the in-service inspection schedule in Specification ~~4.12.~~5.5.8.~~C~~c. The in-service inspection may be limited to one steam generator on a rotating schedule encompassing 6% of the tubes in the single steam generator, provided the previous inspections indicated that the two steam generators are performing in a like manner.

b~~B~~. Steam Generator Tube Sample Selection and Inspection-The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Tables ~~TS.4.12~~5.5.8.-1 and ~~TS.4.12~~5.5.8.-2. The in-service inspection of steam generator tubes shall be performed at the frequencies specified in Specification ~~4.12.~~ 5.5.8.~~C~~c and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification ~~4.12.~~ 5.5.8.~~D~~d. The tubes selected for each in-service inspection shall include at least 3% of the total number of tubes in all steam generators and at least 20% of the total number of sleeves in service in both steam generators; the tubes selected for these inspections shall be selected on a random basis except:

1. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

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2. The first sample of tubes selected for each in-service inspection (subsequent to the preservice inspection) of each steam generator shall include:

   (a) All tubes that previously had detectable wall penetrations (>20%) that have not been plugged or sleeve repaired in the affected area.

   (b) Tubes in those areas where experience has indicated potential problems.

   (c) A tube inspection (pursuant to Specification 5.5.8.d~~4.12.D~~.1.(h)) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

3. In addition to the sample required in Specification 5.5.8.b~~4.12.B~~.2.a through c, all tubes which have had the F* or EF* criteria applied will be inspected in the F* and EF* regions of the roll expanded region. The region of these tubes below the F* and EF* regions may be excluded from the requirements of 5.5.8.b ~~4.12.B~~.2.a.

4. The tubes selected as the second and third samples (if required by Tables ~~TS.4.12~~5.5.8-1 or ~~TS.4.12~~ 5.5.8-2) during each inservice inspection may be subjected to a partial tube or sleeve inspection provided:

   (a) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

   (b) The inspections include those portions of the tubes or sleeves where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

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5.5.8 5. Indications left in service as a result of application of tube support plate voltage-based repair criteria shall be inspected by bobbin coil probe during all future refueling outages.

6. Implementation of the steam generator tube/tube support plate repair criteria requires a 100 percent bobbin coil inspection for hot leg and cold leg tube support plate intersections down to the lowest cold leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20 percent random sampling of tubes inspected over their full length.

c~~C~~. Inspection Frequencies-The above required in-service inspections of steam generator tubes shall be performed at the following frequencies:

1. In-service inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

2. If the results of the inservice inspection of a steam generator conducted in accordance with Table 5.5.8.~~TS.4.12~~-1 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.5.8.c~~4.12.C~~.1; the interval may then be extended to a maximum of once per 40 months.

3. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 5.5.8~~TS.4.12~~-1 during the shutdown subsequent to any of the following conditions.

   (a) Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.14~~3.1.C.6~~.

   (b) A seismic occurrence greater than the Operating Basis Earthquake.

   (c) A loss-of-coolant accident requiring actuation of the engineered safeguards.

   (d) A main steam line or feedwater line break.

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d~~D~~. Acceptance Criteria

5.5.8

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1. As used in this Specification:

   (a) Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

   (b) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

   (c) Degraded Tube means a tube containing imperfections ≥20% of the nominal wall thickness caused by degradation.

   (d) % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

   (e) Defect means an imperfection of such severity that it exceeds the repair limit. A tube containing a defect is defective.

   (f) Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving because it may become unserviceable prior to the next inspection and is equal to 50% of the nominal tube wall thickness. If significant general tube thinning occurs, this criteria will be reduced to 40% wall penetration. This definition does not apply to the portion of the tube in the tubesheet below the F* or EF* distance provided the tube is not degraded (i.e., no indications of cracks) within the F* or EF* distance for F* or EF* tubes. The repair limit for the pressure boundary region of any sleeve is 25% of the nominal sleeve wall thickness. This definition does not apply to tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to Specification 5.5.8.d~~4.12.D~~.4 for the repair limit applicable to these intersections.

   (g) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break.

   (h) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

   (i) Sleeving is the repair of degraded tube regions using a new Alloy 690 tubing sleeve inserted inside the parent tube and sealed at each end by welding or by replacing the lower weld in a full depth tubesheet sleeve with a hard rolled joint. The new sleeve becomes the pressure boundary spanning the original degraded tube region.

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(j) F* Distance is the distance from the bottom of the hardroll trans A5.0-04 toward the bottom of the tubesheet that has been conservatively determined to be 1.07 inches (not including eddy current uncertainty). The F* distance applies to roll expanded regions below the midplane of the tubesheet

(k) F* Tube is a tube with degradation, below the F* distance, equal to or greater than 40%, and not degraded (i.e., no indications of cracking) within the F* distance.

(l) EF* Distance is the distance from the bottom of the upper hardroll transition toward the bottom of the tubesheet that has been conservatively determined to be 1.67 inches (not including eddy current uncertainty). EF* distance applies to roll expanded regions when the top of the additional roll expansion is 2.0 inches or greater down from the top of the tubesheet

(m) EF* Tube is a tube with degradation, below the EF* distance, equal to or greater than 40%, and not degraded (i.e., no indications of cracking) within the EF* distance.

2. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair by sleeving all tubes exceeding the repair limit and all tubes containing through-wall cracks or classify as F* or EF* tubes) required by Tables 5.5.8~~TS.4.12~~-1 and 5.5.8~~TS.4.12~~-2.

3. Tube repair, after April 1, 1999, using Combustion Engineering welded sleeves shall be in accordance with the methods described in the following:

CEN-629-P, Revision 03-P, "Repair of Westinghouse Series 44 and 51 Steam Generator Tubes Using Leak Tight Sleeves";

4. Tube Support Plate Repair Limit is used for the disposition of a steam generator tube for continued service that is experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the repair limit is based on maintaining steam generator serviceability as described below:

a. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltages less than or equal to 2.0 volts will be allowed to remain in service.

b. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts, will be repaired or plugged, except as noted in Specification ~~4.12.D~~5.5.8.d.4.c below.

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c. Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts but less than or equal to the upper voltage repair limit, may remain in service if a rotating pancake coil (or comparable examination technique) inspection does not detect degradation. Steam generator tubes, with indications of outside diameter stress corrosion cracking degradation with a bobbin voltage greater than the upper voltage repair limit will be plugged or repaired.

d. If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits in Specifications 5.5.8.d.~~4.12.D~~.4.(a), (b) and (c). The mid-cycle repair limits are determined from the following equations:

$$V_{murl} = \frac{V_{SL}}{1.0 + NDE + Gr\left(\frac{CL - \Delta T}{CL}\right)}$$

$$V_{mlrl} = V_{murl} - (V_{url} - 2.0)\left(\frac{CL - \Delta T}{CL}\right)$$

where:

V_{URL} = upper voltage repair limit

V_{LRL} = lower voltage repair limit

V_{MURL} = mid-cycle upper voltage repair limit based on time into cycle

V_{MLRL} = mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle

Δt = length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented

CL = cycle length (time between two scheduled steam generator inspections)

V_{SL} = structural limit voltage

Gr = average growth rate per cycle length

NDE = 95 percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20 percent has been approved by the NRC)

Implementation of these mid-cycle repair limits should follow the same approach as described in Specifications 5.5.8.d.~~4.12.D~~.4.(a), (b) and (c)..

Note: The upper voltage repair limit is calculated according to the methodology in Generic Letter 95-05 as supplemented.

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~~E~~. Steam Generator Tube Inspection Report

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5.6.7 1. Following each in-service inspection of steam generator tubes, if there are any tubes requiring plugging or sleeving, the number of tubes plugged or sleeved in each steam generator shall be reported to the Commission within 15 days.

2. The results of steam generator tube inservice inspections shall be included with the summary reports of ASME Code Section XI inspections submitted within 90 days of the end of each refueling outage. Results of steam generator tube inservice inspections not associated with a refueling outage shall be submitted within 90 days of the completion of the inspection. These reports shall include: (1) number and extent of tubes inspected, (2) location and percent of wall-thickness penetration for each indication of an imperfection and (3) identification of tubes plugged or sleeved.

3. Results of steam generator tube inspections which fall into Category C-3 require notification to the Commission prior to resumption of plant operation, and reporting as a special report to the Commission within 30 days. This special report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

4. The results of inspections performed under Specification 5.5.8.b ~~4.12.B~~ for all tubes that have defects below the F* or EF* distance, and were not plugged, shall be reported to the Commission within 15 days following the inspection. The report shall include:

   a. Identification of F* and EF* tubes, and

   b. Location and extent of degradation.

5. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the NRC staff prior to returning the steam generators to service should any of the following conditions arise:

   a. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steamline break) for the next operating cycle.

   b. If circumferential crack-like indications are detected at the tube support plate intersections.

   c. If indications are identified that extend beyond the confines of the tube support plate.

   d. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

   e. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1 x 10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.

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TABLE 5.5.8~~TS.4.12~~-1

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STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Repair defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Repair defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Repair defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., Repair defective tubes and inspect 2S tubes in each other S.G. Prompt notification to NRC.	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and repair defective tubes. Prompt notification to NRC.	N/A	N/A

S=3%; When two steam generators are inspected during that outage.
S=6%; When one steam generator is inspected during that outage.

~~TABLE TS.4.12-1~~
~~REV 137 8/13/98~~

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TABLE 5.5.8~~TS.4.12~~-2

Steam Generator Tube Sleeve Inspection

1st Sample Inspection			2nd Sample Inspection	
Sample Size	Result	Action Required	Result	Action Required
A minimum of 20% of Tube Sleeves (1)	C-1	None	N/A	N/A
	C-2	Inspect all remaining tube sleeves in this S.G. and plug or repair defective sleeved tubes.	C-1	None
			C-2	Plug or repair defective sleeved tubes
			C-3	Perform action for C-3 result of first sample
	C-3	Inspect all tube sleeves in this S.G., inspect 20% of the tube sleeves in the other S.G., and plug or repair defective sleeved tubes	The other S.G. is C-1	None
			The other S.G. is C-2	Perform action for C-2 results of first sample
			The other S.G. is C-3	Inspect all tube sleeves in each S.G. and plug or repair defective sleeved tubes

(1) Each type of sleeve is considered a separate population for determination of scope expansion

~~TABLE TS.4.12-2~~
~~REV 132 11/4/97~~

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5.6.6

LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP)- Reactor Coolant System Cold Leg Temperature (RCSCLT) > Safety Injection (SI) Pump Disable Temperature"; R-12

LCO 3.4.13, "Low Temperature Overpressure Protection (LTOP) - Reactor Coolant System Cold Leg Temperature (RCSCLT) ≤ Safety Injection (SI) Pump Disable Temperature"; and R-12

LCO 3.5.3, "ECCS - Shutdown.

b~~2~~. The analytical methods used to determine the RCS pressure and temperature limits and Cold Overpressure Mitigation System setpoints shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" (~~I~~includes any exemption granted by NRC to ASME Code Case N-514)

5.6.8

When a report is required by Condition or of LCO 3.3.3, "Event Monitoring (EM) Instrumentation." a report shall be submitted within the following 14 days. 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.

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NSHD category	Change number 5.0-	Discussion Of Change
LR	02	4.2.A.2. The CTS requirements for inservice testing have been relocated to the Inservice Testing (IST) Program in accordance with the guidance of NUREG-1431. This change is acceptable since the IST is required by the Administrative Controls Section 5.5. Since the program definition has been moved to the Administrative Controls section of the ITS this is a less restrictive change.
LR	03	Table 4.2-1 and 6.5.F. The CTS requirements for reactor coolant pump flywheel inspection have been relocated to the Reactor Coolant Pump Flywheel Inspection Program which is required by ITS Administrative Controls Section 5.5. This change is acceptable since reactor coolant pump flywheel inspection continues to be required by ITS Section 5.5. Since the program definition has been moved to the Administrative Controls section of the ITS, this is a less restrictive change. This change is consistent with the guidance of NUREG-1431.
A	04	CTS 4.12. The CTS requirements for Steam Generator (SG) tube surveillance in CTS 4.12.A through D have been included in the SG Tube Surveillance Program in the ITS Administrative Controls Section 5.5.8. CTS 4.12.E has been included in the Steam Generator Tube Inspection Report in ITS Administrative Controls Section 5.6.7. This change is acceptable since SG tube surveillance will continue to be required in accordance with the new program and report. Since there are no changes in technical requirements, this is an administrative change.

NSHD category	Change number 5.0-	Discussion Of Change
A	05	CTS 4.12.E. The CTS requirements for steam generator (SG) tube surveillance reports in CTS 4.12.E have been included in the SG Tube Inspection Report which is required by the ITS Administrative Controls Section 5.6.7. This change is acceptable since SG tube surveillance reports will continue to be required in accordance with the new program and report. Since there are no changes in technical requirements, this is an administrative change. This change is consistent with the guidance of NUREG-1431.
A	06	Throughout the CTS 6.0 markup, the section and paragraph numbering, punctuation and paragraph references have been revised to correspond to the NUREG-1431 format, numbering and punctuation. Since these changes do not introduce any substantive requirement changes, these changes are administrative.
A	07	CTS 6.5.L.2.b The use of "involves an unreviewed safety question as defined in" has been replaced by "requires NRC approval pursuant to" to be consistent with the most recent issuance of 10CFR50.59. Since this does not involve any substantive changes, this is an administrative change. This change is consistent with NUREG-1431 as modified by approved TSTF-364.
	08	Not used.
	09	Not used.
	10	Not used.

NSHD category	Change number 5.0-	Discussion Of Change
A	26	6.6.A, B and C. In conformance with the guidance of NUREG-1431 as modified by TSTF-152, a note is included to clarify the ITS reporting requirements. Since this does not change the reporting requirements for PI this change is administrative.
A	27	CTS 6.6.D. The CTS requirement for monthly reporting of challenges to the pressurizer power operated relief valves or pressurizer safety valves is not included in the ITS. In accordance with NRC GL 97-02, "Revised Contents of the Monthly Operating Report", the NRC has requested less information in the monthly operating report. This generic leter identifies what needs to be reported to support the NRC Performance Indicator Program, and availability and capacity statistics. The generic letter does not identify the need to report these valve challenges and thus in conformance with the guidance of NUREG-1431 as modified by TSTF-258, Rev. 4, these challenges are not included in the ITS. Since these changes only involve reporting requirements and do not affect the safe operation of the plant this is an administrative change.
A	28	CTS 6.6.E. A new COLR reference to the latest Prairie Island approved steam line break methodology is included. Since the change is just a new reference which was previously reviewed and approved by the NRC in a letter dated January 21, 2000, this is an administrative change in this submittal. The status designator for reports NSPNAD-8101 and 8102 have been corrected to "PA" to indicate that these are proprietary documents. These changes do not materially change these reports, thus these are also administrative changes.

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Weekly	At least once per 7 days
Monthly	At least once per 31 days
Semiquarterly	At least once per 46 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

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b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5.8~~9~~ Steam Generator (SG) Tube Surveillance Program

CL5.0-64

~~Reviewer's Note: The Licensee's current licensing basis steam generator tube surveillance requirements shall be relocated from the LCO and included here. An appropriate administrative controls program format should be used.~~

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Steam generator tubes in each unit shall be determined operable by the following:

a. Steam Generator Sample Selection and Inspection

Each steam generator shall be determined operable in

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(continued)

5.5 Programs and Manuals (continued)

accordance with the in-service inspection schedule in Specification 5.5.8.c. The in-service

inspection may be limited to one steam generator on a rotating schedule encompassing 6% of the tubes in the single steam generator, provided the previous inspections indicated that the two steam generators are performing in a like manner.

b. Steam Generator Tube Sample Selection and Inspection

The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Tables 5.5.8-1 and 5.5.8-2. The in-service inspection of steam generator tubes shall be performed at the frequencies specified in Specification 5.5.8.c and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 5.5.8.d. The tubes selected for each in-service inspection shall include at least 3% of the total number of tubes in all steam generators and at least 20% of the total number of sleeves in service in both steam generators; the tubes selected for these inspections shall be selected on a random basis except:

1. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

2. The first sample of tubes selected for each in-service inspection (subsequent to the preservice inspection) of each steam generator shall include:

   (a) All tubes that previously had detectable wall penetrations (>20%) that have not been plugged or sleeve repaired in the affected area.

   (b) Tubes in those areas where experience has indicated potential problems.

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5.5 Programs and Manuals (continued)

(c) A tube inspection (pursuant to Specification 5.5.8.d.1(h)) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

3. In addition to the sample required in Specification 5.5.8.b.2(a) through (c), all tubes which have had the F* or EF* criteria applied will be inspected in the F* and EF* regions of the roll expanded region. The region of these tubes below the F* and EF* regions may be excluded from the requirements of Specification 5.5.8.b.2(a).

4. The tubes selected as the second and third samples (if required by Tables 5.5.8-1 or 5.5.8-2) during each in-service inspection may be subjected to a partial tube or sleeve inspection provided:

(a) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

(b) The inspections include those portions of the tubes or sleeves where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.

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5.5 Programs and Manuals (continued)

C-2 One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.

C-3 More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

5. Indications left in service as a result of application of tube support plate voltage-based repair criteria shall be inspected by bobbin coil probe during all future refueling outages.

6. Implementation of the steam generator tube/tube support plate repair criteria requires a 100 percent bobbin coil inspection for hot leg and cold leg tube support plate intersections down to the lowest cold leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20 percent random sampling of tubes inspected over their full length.

c. Inspection Frequencies

The above required in-service inspections of steam generator tubes shall be performed at the following frequencies:

1. In-service inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection.

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If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

2. If the results of the in-service inspection of a steam generator conducted in accordance with Table 5.5.8-1 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.5.8.c.1; the interval may then be extended to a maximum of once per 40 months.

3. Additional, unscheduled in-service inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 5.5.8-1 during the shutdown subsequent to any of the following conditions:

(a) Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.14.

(b) A seismic occurrence greater than the Operating Basis Earthquake.

(c) A loss-of-coolant accident requiring actuation of the engineered safeguards.

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(d) A main steam line or feedwater line break.

d. Acceptance Criteria

1. As used in this Specification:

(a) Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

(b) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

(c) Degraded Tube means a tube containing imperfections ≥20% of the nominal wall thickness caused by degradation.

(d) % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

(e) Defect means an imperfection of such severity that it exceeds the repair limit. A tube containing a defect is defective.

(f) Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving because it may become unserviceable prior to the next inspection and is equal to 50% of the nominal tube wall thickness. If significant general

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(continued)

5.5 Programs and Manuals (continued)

tube thinning occurs, this criteria will be reduced to 40% wall penetration. This definition does not apply to the portion of the tube in the tubesheet below the F* distance provided the tube is not degraded (i.e., no indications of cracks) within the F* or EF* distance for F* or EF* tubes. The repair limit for the pressure boundary region of any sleeve is 25% of the nominal sleeve wall thickness. This definition does not apply to tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to Specification 5.5.8.d.4 for the repair limit applicable to these intersections.

(g) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break.

(h) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

(i) Sleeving is the repair of degraded tube regions using a new Alloy 690 tubing sleeve inserted inside the parent tube and sealed at each end by welding or by replacing the lower weld in a full depth tubesheet sleeve with a hard rolled joint. The new sleeve becomes the pressure boundary spanning the original degraded tube region.

(j) F* Distance is the distance from the bottom

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of the hardroll transition toward the bottom of the tubesheet that has been conservatively determined to be 1.07 inches (not including eddy current uncertainty). The F* distance applies to roll expanded regions below the midplane of the tubesheet.

(k) F* Tube is a tube with degradation, below the F* distance, equal to or greater than 40%, and not degraded (i.e., no indications of cracking) within the F* distance.

(l) EF* Distance is the distance from the bottom of the upper hardroll transition toward the bottom of the tubesheet that has been conservatively determined to be 1.67 inches (not including eddy current uncertainty). EF* distance applies to roll expanded regions when the top of the additional roll expansion is 2.0 inches or greater down from the top of the tubesheet.

(m) EF* Tube is a tube with degradation, below the EF* distance, equal to or greater than 40%, and not degraded (i.e., no indications of cracking) within the EF* distance.

2. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair by sleeving all tubes exceeding the repair limit and all tubes containing through-wall cracks or classify as F* or EF* tubes) required by Tables 5.5.8-1 and 5.5.8-2.

3. Tube repair, after April 1, 1999, using Combustion Engineering welded sleeves shall be in accordance with the methods described in the following:

CEN-629-P, Revision 03-P, "Repair of Westinghouse

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(continued)

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Series 44 and 51 Steam Generator Tubes Using Leak Tight Sleeves";

4. Tube Support Plate Repair Limit is used for the disposition of a steam generator tube for continued service that is experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the repair limit is based on maintaining steam generator serviceability as described below:

(a) Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltages less than or equal to 2.0 volts will be allowed to remain in service.

(b) Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts, will be repaired or plugged, except as noted in Specification 5.5.8.d.4(c) below.

(c) Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts but less than or equal to the upper voltage repair limit, may remain in service if a rotating pancake coil (or comparable examination

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(continued)

5.5 Programs and Manuals (continued)

technique) inspection does not detect degradation. Steam generator tubes, with indications of outside diameter stress corrosion cracking degradation with a bobbin voltage greater than the upper voltage repair limit will be plugged or repaired.

(d) If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits in Specifications 5.5.8.d.4(a), (b) and (c). The mid-cycle repair limits are determined from the following equations:

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr\left(\frac{CL - \Delta t}{CL}\right)}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - 2.0)\left(\frac{CL - \Delta t}{CL}\right)$$

where:

V_{URL} = upper voltage repair limit

V_{LRL} = lower voltage repair limit

V_{MURL} = mid-cycle upper voltage repair limit based on time into cycle

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(continued)

5.5 Programs and Manuals (continued)

V_{MLRL} = mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle

Δt = length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented

CL = cycle length (time between two scheduled steam generator inspections)

V_{SL} = structural limit voltage

Gr = average growth rate per cycle length

NDE = 95 percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20 percent has been approved by the NRC)

Implementation of these mid-cycle repair limits should follow the same approach as described in Specifications 5.5.8.d.4(a), (b) and (c).

Note: The upper voltage repair limit is calculated according to the methodology in Generic Letter 95-05 as supplemented.

~~5.5.10 Secondary Water Chemistry Program~~ R-12

~~This program provides controls for monitoring secondary water~~ CL5.0-56

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5.5 Programs and Manuals (continued)

~~chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:~~

~~a. Identification of a sampling schedule for the critical variables and control points for these variables;~~

~~b. Identification of the procedures used to measure the values of the critical variables;~~

~~c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;~~

~~d. Procedures for the recording and management of data;~~

~~e. Procedures defining corrective actions for all off control point chemistry conditions; and~~

~~f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.~~

5.5.9~~11~~ Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems and the Spent Fuel Pool Special and Inservice Purge Ventilation System each operating cycle (18 months for shared systems)~~at the frequencies specified in [Regulatory Guide], and in accordance with [Regulatory Guide 1.52, Revision 2, ASME N510-1989, and AG-1]~~. CL5.0-66

(continued)

5.5 Programs and Manuals (continued)

Demonstrate for the Auxiliary Building Special Ventilation, Shield Building Ventilation, Control Room Special Ventilation, and Spent Fuel Pool Special and Inservice Purge Ventilation Systems that:

a. ~~Demonstrate for each of the ESF systems thatA~~an inplace DOP test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 1~~[0.05]~~% (for DOP, particles having a mean diameter of 0.7 microns); ~~when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below [± 10%].~~

~~ESF Ventilation System~~	~~Flowrate~~

5.5.9~~11~~ Ventilation Filter Testing Program (VFTP) (continued)

b. ~~Demonstrate for each of the ESF systems thatA~~an ~~inplace~~ halogenated hydrocarbon test of the inplace charcoal adsorber shows a penetration and system bypass < 1~~[0.05]~~% (for DOP, particles having a mean diameter of 0.7 microns); ~~when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below [± 10%].~~ CL5.0-66

~~ESF Ventilation System~~	~~Flowrate~~

CL5.0-66

(continued)

5.5 Programs and Manuals (continued)

c. ~~Demonstrate for each of the ESF systems that a~~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 15% penetration (less than 5% penetration for the Control Room Special Ventilation System)~~the value specified below~~ when tested in accordance with ~~[~~ASTM D3803-1989~~]~~ at a temperature of ~~≤ [~~30°C ~~]~~and 95%~~ greater than or equal to the~~ relative humidity (RH) (or 70% RH with humidity controls if the humidity controls are capable of maintaining the humidity of the air entering the charcoal less than or equal to 70% RH under worst-case design-basis conditions); and~~specified below.~~

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~~ESF Ventilation System~~ ~~Penetration~~ ~~RH~~

~~Reviewer's Note: Allowable penetration = [100% - methyl iodide efficiency for charcoal credited in staff safety evaluation]/ (safety factor).~~

~~Safety factor = [5] for systems with heaters.~~
~~= [7] for systems without heaters.~~

d. ~~Demonstrate for each of the ESF systems that~~Tthe pressure drop across the combined HEPA filters~~,~~ ~~the prefilters,~~ and the charcoal adsorbers is less than 6 inches of water at ~~the value specified below when tested in accordance with [Regulatory Guide 1.52,~~ CL5.0-66

~~5.5.11 Ventilation Filter Testing Program (VFTP) (continued)~~

~~Revision 2, and ASME N510-1989] at~~ the system flowrate ~~specified below [~~± 10%~~]~~.

(continued)

5.5 Programs and Manuals (continued)

~~ESF Ventilation System~~	~~Delta P~~	~~Flowrate~~

~~e. Demonstrate that the heaters for each of the ESF systems dissipate the value specified below [± 10%] when tested in accordance with [ASME N510-1989].~~

~~ESF Ventilation System~~	~~Wattage~~

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.10~~2~~ <u>Explosive Gas and Storage Tank Radioactivity Monitoring Program</u>

This program provides controls for potentially explosive gas mixtures contained in the waste gas holdup system ~~[Waste Gas Holdup System]~~, ~~[~~the quantity of radioactivity contained in gas storage tanks ~~or fed into the offgas treatment system~~, 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"].~~ CL5.0-56

(continued)

5.5 Programs and Manuals (continued)

The program shall include:

a. The limits for concentrations of ~~hydrogen and~~ oxygen in the waste gas holdup system ~~[Waste Gas Holdup System]~~ and a surveillance program to ensure the limits are maintained. Such limits shall be

5.5.10~~2~~ Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

appropriate to the system's design criteria ~~(i.e., whether or not the system is designed to withstand a hydrogen explosion)~~; CL5.0-56

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 or equal to 78,800 curies of noble gas (considered as dose equivalent Xe-133) ~~the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of [an uncontrolled release of the tanks' contents]~~; and

c. A surveillance program to ensure that the quantity of radioactivity contained in each of the following tanks shall be limited to 10 curies, excluding tritium and dissolved or entrained noble gases:

Condensate storage tanks
Outside temporary tanks
~~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 the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply~~

(continued)

5.5 Programs and Manuals (continued)

~~and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.~~

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.

5.5.11~~3~~ Diesel Fuel Oil Testing Program

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 the limits specified in Table 1 of ASTM D975-77 when checked for viscosity, water, and sediment ~~applicable ASTM Standards~~. ~~The purpose of the program is to establish the following:~~ CL5.0-56

~~a.~~ Acceptability of new fuel oil shall be determined ~~for use~~ prior to addition to the safeguards storage tanks. Testing of diesel fuel oil stored in the safeguards storage tanks shall be performed at least every 31 days. ~~by determining that the fuel oil has:~~

~~1. an API gravity or an absolute specific gravity within limits,~~

~~5.5.13 Diesel Fuel Oil Testing Program (continued)~~

~~2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and~~

~~3. a clear and bright appearance with proper color;~~

CL5.0-56

(continued)

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~~b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and~~

~~c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days in accordance with ASTM D-2276, Method A-2 or A-3.~~

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies. TA5.0-67

5.5.12~~4~~ <u>Technical Specifications (TS) Bases Control Program</u>

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require~~involve~~ either of the following: R-2

1. a change in the TS incorporated in the license; or

2. a change ~~to the updated FSAR or Bases~~ that requires NRC approval pursuant to~~involves an unreviewed safety question as defined in~~ 10 CFR 50.59. TA5.0-58 R-2

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the U~~F~~SAR.

d. Proposed changes that meet the criteria of Specification 5.5.12~~4~~ 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 on a frequency consistent with USAR updates ~~10 CFR 50.71(e)~~. CL5.0-56 CL5.0-56

5.5 Programs and Manuals (continued)

5.5.13~~5~~ <u>Safety Function Determination Program (SFDP)</u>

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power or no concurrent loss of onsite diesel generator(s), 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: TA5.0-71 R-2

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

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c. A required system redundant to the inoperable support system(s) for the supported systems (a) and (b) above is also inoperable. PA5.0-72

The SFDP 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. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system. TA5.0-71

5.5.14 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. CL5.0-73

b. The peak calculated containment internal pressure for the design basis loss of coolant accident is less than the containment internal design pressure, P_a, of 46 psig.

c. The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.25% of primary containment air weight per day. For pipes connected to systems that are in the auxiliary building special ventilation zone, the total leakage shall be less than 0.1% of primary containment air weight per day at pressure P_a. For pipes connected to systems that are exterior to CL5.0-73

both the shield building and the auxiliary building special ventilation zone, the total leakage past isolation valves shall be less than 0.01% of primary containment air weight per day at

5.5 Programs and Manuals (continued)

pressure P_a.

CL5.0-73

d. Leakage Rate acceptance criteria are:

1. Primary containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. Prior to unit startup, following testing in accordance with the program, the combined leakage rate acceptance criteria are $\leq 0.60\ L_a$ for all components subject to Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests.
2. Air lock testing acceptance criteria are:
   a)Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at ≥ 46 psig.
   b)For each door intergasket test, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 10 psig.

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

TA5.0-86

5.5.15 <u>Battery Monitoring and Maintenance Program</u>

This Program provides for restoration and maintenance of the 125V plant safeguards batteries and service building batteries, which may be used instead of the safeguards batteries during shutdown conditions in accordance with manufacturer's recommendations, as follows:

a. Actions to restore battery cells with float voltage < 2.13 V will be in accordance with manufacturer's recommendations, and

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the minimum established design limit.

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5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational ~~Radiation~~ Exposure Report

---NOTE---

A single submittal may be made for the plant ~~a multiple unit station~~. The submittal should combine sections common to both ~~all~~ units ~~at the station~~. PA5.0-68

A tabulation on an annual basis of the number of plant~~station~~, utility, and other personnel (including contractors) for whom monitoring was performed, receiving an annual deep dose equivalent~~exposures~~ > 100 mrem~~/yr~~ and the~~ir~~ associated collective deep dose equivalent (reported in person~~man~~-rem)~~exposure~~ according to work and job functions, ~~(~~e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (~~[~~describe maintenance~~]~~), waste processing, and refueling~~)~~. This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber~~dosimeter~~, thermoluminescent dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totalling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total deep ~~whole body~~ dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year. ~~[The initial report shall be submitted by April 30 of the year following the initial criticality.]~~ CL5.0-56 TA5.0-74

(continued)

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Monitoring ~~Operating~~ Report

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

A single submittal may be made for the plant~~ a multiple unit station~~. The submittal should combine sections common to both~~ all~~ units~~ at the station~~. PA5.0-68

The Annual Radiological Environmental Monitoring ~~Operating~~ Report covering the operation of the plant ~~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 CL5.0-56

5.6.2 Annual Radiological Environmental Monitoring~~Operating~~ Report (continued)

CL5.0-56

(ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Monitoring ~~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 Regulatory Guide 4.8, December 1975, of all radiological environmental samples taken during the report period~~ the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979~~]. ~~[The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with~~

(continued)

5.6 Reporting Requirements

~~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.

The report shall also include the following: a summary description of the radiological environmental monitoring program; a map of sampling locations keyed to a table giving distances and directions from the reactor site; and the results of licensees participation in the Interlaboratory Comparison Program defined in the ODCM. CL5.0-56

5.6.3 Radioactive Effluent ~~Release~~ Report

-----NOTE-----

A single submittal may be made for the plant ~~a multiple unit station~~. The submittal shall ~~should~~ combine sections common to both ~~all~~ units ~~at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit~~. PA5.0-68

The Radioactive Effluent ~~Release~~ Report covering the operation of the plant during the previous calendar year ~~unit~~ shall be submitted ~~in accordance~~ by May 15 of each year ~~with 10 CFR 50.36a~~. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the plant ~~unit~~. The material provided shall be consistent with the objectives outlined in the ODCM ~~and Process Control Program~~ and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1. CL5.0-56

(continued)

5.6 Reporting Requirements (continued)

5.6.4 Monthly Operating Reports

TA5.0-54

Routine reports of operating statistics and shutdown experience~~[, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves,]~~ shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

LCO 3.1.1, "SHUTDOWN MARGIN (SDM)";
LCO 3.1.3, "Isothermal Temperature Coefficient (ITC)"; PA5.0-76
LCO 3.1.5, "Shutdown Bank Insertion Limits";
LCO 3.1.6, "Control Bank Insertion Limits";
LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 2";
LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)";
LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)";
LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)";
LCO 3.4.1, "RCS Pressure, Temperature, and Flow - Departure from Nucleate Boiling (DNB) Limits"; and
LCO 3.9.1, "Boron Concentration".

~~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,

(continued)

5.6 Reporting Requirements (continued)

specifically those described in the following documents: PA5.0-76

1. NSPNAD-8101-PA, "Qualification of Reactor Physics Methods for Application to PI Units" (latest approved version);

2. NSPNAD-8102-PA, "Prairie Island Nuclear Power Plant Reload Safety Evaluation Methods for Application to PI Units"(latest approved version);

3. NSPNAD-97002-PA, "Northern States Power Company's Steam Line Break Methodology" (latest approved version);

4. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology", July, 1985;

5. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code", August, 1985;

6. WCAP-10924-P-A, "Westinghouse Large Break LOCA Best-Estimate Methodology", December, 1988;

7. WCAP-10924-P-A, Volume 1, Addendum 4, "Westinghouse Large Break LOCA Best Estimate Methodology", August, 1990;

8. XN-NF-77-57 (A), XN-NF-77-57, Supplement 1 (A), "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors Phase II", May, 1981;

9. WCAP-13677, "10 CFR 50.46 Evaluation Model Report: W-COBRA/TRAC 2-Loop Upper Plenum Injection Model Update to Support ZIRLO$_{TM}$ Cladding Options", April 1993 (approved by NRC SE dated November 26, 1993);

(continued)

5.6 Reporting Requirements

10. NSPNAD-93003-A, "Transient Power Distribution Methodology", (latest approved version).

~~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 Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

TA5.0-77

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat–up, cooldown, low temperature operation, criticality, and hydrostatic testing, LTOP arming, PORV lift settings and Safety Injection Pump Disable Temperature as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

PA5.0-76

LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.10, "Pressurizer Safety Valves";
LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) - Reactor Coolant System Cold Leg Temperature (RCSCLT)

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(continued)

5.6 Reporting Requirements

> Safety Injection (SI) Pump Disable Temperature";
LCO 3.4.13, "Low Temperature Overpressure Protection (LTOP) - Reactor Coolant System Cold Leg Temperature (RCSCLT) ≤ Safety Injection (SI) Pump Disable Temperature"; and
LCO 3.5.3, "ECCS - Shutdown". R-12

~~[The individual specifications that address RCS pressure and temperature limits must be referenced here.]~~

b. The analytical methods used to determine the RCS pressure and temperature limits and Cold Overpressure Mitigation System setpoints shall be those previously reviewed and approved by the NRC, specifically those described in the following documents: CL5.0-56

WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" (includes any exemption granted by NRC to ASME Code Case N-514).

~~[Identify the NRC staff approval document by date.]~~ CL5.0-56

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto. Changes to the curves, setpoints, or parameters in the PTLR resulting from new or additional analysis of beltline material properties shall be submitted to the NRC prior to issuance of an updated PTLR.

~~Reviewers' Notes: The methodology for the calculation of the P-T limits for NRC approval should include the following provisions:~~ PA5.0-76

~~1. The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).~~

~~2. The Reactor Vessel Material Surveillance Program shall comply with~~

(continued)

5.6 Reporting Requirements

~~Appendix H to 10 CFR 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.~~ PA5.0-76

3. ~~Low Temperature Overpressure Protection (LTOP) System lift setting limits for the Power Operated Relief Valves (PORVs), developed using NRC-approved methodologies may be included in the PTLR.~~

4. ~~The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for radiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.~~

5. ~~The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits.~~

~~5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)~~

6. ~~The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves.~~

7. ~~Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value ($2\sigma_{\Delta}$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase RT_{NDT} + $2\sigma_{\Delta}$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.~~

(continued)

5.6 Reporting Requirements

~~5.6.7 EDG Failure Report~~

~~If an individual emergency diesel generator (EDG) experiences four or more valid failures in the last 25 demands, these failures and any nonvalid 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.~~ CL5.0-81

5.6.8 ~~P~~AEM Report

When a report is required by Condition ~~B~~C or ~~G~~J of LCO 3.3.~~[3]~~, "~~Post~~Event ~~Accident~~ Monitoring (~~P~~AEM) Instrumentation," a report shall be submitted within the following 14 days. 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. R-7

~~5.6.9 Tendon Surveillance Report~~

~~Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported~~ PA5.0-61

~~to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.~~

(continued)

5.6 Reporting Requirements (continued)

5.6.7~~10~~ Steam Generator Tube Inspection ~~Inspector~~ Report

CL5.0-83

1. Following each in-service inspection of steam generator tubes, if there are any tubes requiring plugging or sleeving, the number of tubes plugged or sleeved in each steam generator shall be reported to the Commission within 15 days.

2. The results of steam generator tube in-service inspections shall be included with the summary reports of ASME Code Section XI inspections submitted within 90 days of the end of each refueling outage. Results of steam generator tube in-service inspections not associated with a refueling outage shall be submitted within 90 days of the completion of the inspection. These reports shall include: (1) number and extent of tubes inspected, (2) location and percent of wall-thickness penetration for each indication of an imperfection, and (3) identification of tubes plugged or sleeved.

3. Results of steam generator tube inspections which fall into Category C-3 require notification to the Commission prior to resumption of plant operation, and reporting as a special report to the Commission within 30 days. This special report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

4. The results of inspections performed under Specification 5.5.8.b for all tubes that have defects below the F* or EF* distance, and were not plugged, shall be reported to the Commission within 15 days following the inspection. The report shall

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5.6 Reporting Requirements (continued)

include:

a. Identification of F* and EF* tubes, and

b. Location and extent of degradation.

5. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the NRC staff prior to returning the steam generators to service should any of the following conditions arise:

a. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steamline break) for the next operating cycle.

b. If circumferential crack-like indications are detected at the tube support plate intersections.

c. If indications are identified that extend beyond the confines of the tube support plate.

d. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

e. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1×10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.

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5.6 Reporting Requirements (continued)

~~Reviewer's Note: Reports required by the Licensee's current licensing basis regarding steam generator tube surveillance requirements shall be included here. An appropriate administrative controls format should be used.~~

~~Reviewer's Note: These reports may be required covering inspection, test, and maintenance activities. These reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.~~

TA5.0-54

5.0 ADMINISTRATIVE CONTROLS

[5.7 High Radiation Area]

TA5.0-54

~~5.7.1~~ As provided in ~~Pursuant to 10 CFR 20,~~ paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied in place of the controls required by paragraph ~~in lieu of the requirements of~~ 10 CFR 20.1601(a) and (b) of ~~, each high radiation area, as defined in~~ 10 CFR 20:

5.7.1 <u>High Radiation Areas accessible to personnel in which radiation levels could result in an individual receiving a deep dose equivalent</u> **less than** <u>1.0 rem in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates</u>~~, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr,~~

PA5.0-84

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in each such area ~~and entrance thereto~~ shall be controlled by means ~~requiring issuance~~ of a Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and ~~(e.g., [Health Physics Technicians]) or~~ personnel continuously escorted by such individuals may be exempted from the ~~RWP issuance~~ requirement for an RWP or equivalent while performing ~~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 to, exit from, and work in ~~into~~ such ~~high radiation~~ areas.

d. Each ~~Any~~ individual or group entering ~~of individuals permitted to enter~~ such an area~~s~~ shall possess ~~be provided with or accompanied~~

(continued)

~~by one or more of the following~~:

1~~a~~. A radiation monitoring device that continuously displays ~~indicates the~~ radiation dose rates in the area; or~~.~~

2~~b~~. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's ~~a preset integrated~~ dose alarm setpoint is reached, with an appropriate alarm setpoint; or ~~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.~~

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area; or

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area, who is responsible for controlling personnel exposure within the area; or

   (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

(continued)

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry. TA5.0-54

~~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 [Radiation Protection Manager] in the RWP.~~

5.7.2 High Radiation Areas accessible to personnel in which radiation levels could result in an individual receiving a deep dose equivalent **in excess of** 1.0 rem in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates, but less than 500 rad in one hour at one meter from the source ~~In addition to the requirements of Specification 5.7.1, areas with radiation levels ≥ 1000 mrem/hr~~

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door~~s~~ or gate that ~~to~~ prevents unauthorized entry, and, in addition: TA5.0-54

1. All such door and gate ~~the~~ keys shall be maintained under the administrative control of the s~~S~~hift supervisor, radiation protection manager, or their designee ~~Foreman on duty or health physics supervision~~. PA5.0-84
2. Doors and gates shall remain locked except during periods of ~~access by~~ personnel or equipment entry or exit. ~~under an approved RWP that shall specify the dose rate levels in~~

5.7.2 (continued)

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas. TA5.0-54

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint; or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area; or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area, who is responsible for controlling personnel exposure within the area; or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

4. In those cases where options (2) and (3), above are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation

monitoring device shall be used that continuously displays radiation dose rates in the area.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.
~~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.~~ TA5.0-54

~~5.7.3~~ f. Such ~~For~~ individual ~~high radiation~~ areas ~~with radiation levels of > 1000 mrem/hr, accessible to personnel,~~ that are located within a larger area~~s~~ ~~such as reactor containment,~~ where no enclosure exists for the purpose~~s~~ of locking~~, or that cannot be continuously guarded,~~ and where no enclosure can be reasonably constructed around the individual area, that individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, ~~and~~ conspicuously posted, and a flashing light shall be activated at the area as a warning device.

TABLE 5.5.8-1

TABLE 5.5.8-1

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Repair defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Repair defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Repair defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., Repair defective tubes and inspect 2S tubes in each other S.G. Prompt notification to NRC.	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and repair defective tubes. Prompt notification to NRC.	N/A	N/A

S=3%; When two steam generators are inspected during that outage.
S=6%; When one steam generator is inspected during that outage.

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TABLE 5.5.8-2
Steam Generator Tube Sleeve Inspection

1st Sample Inspection			2nd Sample Inspection	
Sample Size	Result	Action Required	Result	Action Required
A minimum of 20% of Tube Sleeves (1)	C-1	None	N/A	N/A
	C-2	Inspect all remaining tube sleeves in this S.G. and plug or repair defective sleeved tubes.	C-1	None
			C-2	Plug or repair defective sleeved tubes
			C-3	Perform action for C-3 result of first sample
	C-3	Inspect all tube sleeves in this S.G., inspect 20% of the tube sleeves in the other S.G., and plug or repair defective sleeved tubes	The other S.G. is C-1	None
			The other S.G. is C-2	Perform action for C-2 results of first sample
			The other S.G. is C-3	Inspect all tube sleeves in each S.G. and plug or repair defective sleeved tubes

(1) Each type of sleeve is considered a separate population for determination of scope expansion

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Difference Category	Difference Number 5.0-	Justification for Differences
TA	63	This change incorporates TSTF-279.
CL	64	The CTS SG program requirements are provided as required by the Reviewer's Note in NUREG-1431. The CTS requirements from 4.12.A through D have been included in ITS.
	65	Not used.
CL	66	In conformance with the guidance of NUREG-1431, program definition for the VFTP is provided. The format and contents of the Program requirements have been changed to incorporate CTS requirements for these systems and incorporate the requirements of NRC Generic Letter 99-02.
TA	67	This change incorporates TSTF-118.
PA	68	The Note in brackets has been modified to correctly apply to PI.
PA	69	A new test interval of "Semiquarterly" has been included to allow accelerated testing of equipment that fails a quarterly test as required by the ASME test program.
	70	Not used.

Difference Category	Difference Number 5.0-	Justification for Differences
	80	Not used.
CL	81	CTS do not require this report; therefore it is not included in the ITS. This change is also consistent with approved TSTF-37, Revision 2.
	82	Not used.
CL	83	The CTS report requirements are provided in the ITS as required by the Reviewer's Notes in NUREG-1431. The CTS requirements from 4.12.E have been included in ITS.
PA	84	The titles of ITS 5.7.1 and 5.7.2 have been revised to be consistent with the guidance of Regulatory Guide 8.38. This change is beneficial in that overall it may reduce plant radiation exposure.

Part G

PACKAGE 5.0

ADMINISTRATIVE CONTROLS

NO SIGNIFICANT HAZARDS DETERMINATION AND ENVIRONMENTAL ASSESSMENT

NO SIGNIFICANT HAZARDS DETERMINATION

The proposed changes to the Operating License have been evaluated to determine whether they constitute a significant hazards consideration as required by 10CFR Part 50, Section 50.91 using the standards provided in Section 50.92.

For ease of review, the changes are evaluated in groupings according to the type of change involved. A single generic evaluation may suffice for some of the changes while others may require specific evaluation in which case the appropriate reference change numbers are provided.

A - Administrative (GENERIC NSHD)
(A5.0-00, A5.0-04, A5.0-06, A5.0-07, A5.0-11, A5.0-12, A5.0-13, A5.0-14, A5.0-16, A5.0-24, A5.0-26, A5.0-27, A5.0-28, A5.0-31, A5.0-32, A5.0-33, A5.0-34, A5.0-36, A5.0-38)

Most administrative changes have not been marked-up in the Current Technical Specifications, and may not be specifically referenced to a discussion of change. This No Significant Hazards Determination (NSHD) may be referenced in a discussion of change by the prefix "A" if the change is not obviously an administrative change and requires an explanation.

These proposed changes are editorial in nature. They involve reformatting, renaming, renumbering, or rewording of existing Technical Specifications to provide consistency with NUREG-1431 or conformance with the Writer's Guide, or change of current plant terminology to conform to NUREG-1431. Some administrative changes involve relocation of requirements within the Technical Specifications without affecting their technical content. Clarifications within the new Prairie Island Improved Technical Specifications which do not impose new requirements on plant operation are also considered administrative.

LR - Less restrictive, Relocated details (GENERIC NSHD)
(LR5.0-01, LR5.0-02, LR5.0-03, LR5.0-05, LR5.0-22)

Some information in the Prairie Island Current Technical Specifications that is descriptive in nature regarding the equipment, system(s), actions or surveillances identified by the specification has been removed from the proposed specification and relocated to the proposed Bases, Updated Safety Analysis Report or licensee controlled procedures. The relocation of this descriptive information to the Bases of the Improved Technical Specifications, Updated Safety Analysis Report or licensee controlled procedures is acceptable because these documents will be controlled by the Improved Technical Specifications required programs, procedures or 10CFR50.59. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed changes relocate detailed, descriptive requirements from the Technical Specifications to the Bases, Updated Safety Analysis Report or licensee controlled procedures. These documents containing the relocated requirements will be maintained under the provisions of 10CFR50.59, a program or procedure based on 10CFR50.59 evaluation of changes, or NRC approved methodologies. Since these documents to which the Technical Specifications requirements have been relocated are evaluated under 10CFR50.59 or its guidance, or in accordance with NRC approved methodologies, no increase in the probability or consequences of an accident previously evaluate will be allowed without prior NRC approval. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   These proposed changes do not necessitate physical alteration of the plant, that is, no new or different type of equipment will be installed, or change parameters governing normal plant operation. The proposed changes will not impose any different requirements and adequate control of the information will be maintained. Thus, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-1C	11	SR	3.3.4.1	
Table 4.1-1C	11	SR	3.3.4.2	
Table 4.1-1C	12		Deleted - Boric Acid LAR	
Table 4.1-1C	13		Relocated - TRM	
Table 4.1-1C	14		CTS Deleted	
Table 4.1-1C	15	TABLE	3.3.1-1	16.b.2
Table 4.1-1C	15		Relocated - TRM	
Table 4.1-1C	16		Relocated - TRM	
Table 4.1-1C	17		Relocated - TRM	
Table 4.1-1C	18	SR	3.3.1.12	
Table 4.1-1C	19		Relocated - TRM	
Table 4.1-1C	20		Relocated - TRM	
Table 4.1-1C	21	SR	3.3.3.1	
Table 4.1-1C	21	SR	3.3.3.2	
Table 4.1-1C	21	SR	3.3.3.3	
Table 4.1-1C	22		CTS Deleted	
Table 4.1-1C	23		CTS Deleted	
Table 4.1-1C	24		Relocated - TRM	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-1C	24	SR	3.3.6.5	
Table 4.1-1C	24	SR	3.3.6.2	
Table 4.1-1C	25	SR	3.4.12.4	
Table 4.1-1C	25	SR	3.4.12.5	
Table 4.1-1C	25	SR	3.4.13.5	
Table 4.1-1C	25	SR	3.4.13.6	
Table 4.1-1C	26		Relocated - TRM	
Table 4.1-1C	27		Relocated - TRM	
Table 4.1-1C	28		Relocated - TRM	
Table 4.1-1C	29	SR	3.3.3.1	
Table 4.1-1C	29	SR	3.3.3.2	
Table 4.1-1C	29	(Partial)	Relocated - TRM	
Table 4.1-1C	30		Relocated - Bases	
Table 4.1-1C	31		Relocated - TRM	
Table 4.1-1C	Note 30	SR	3.1.7.1	
Table 4.1-1C	Note 31		Deleted	
Table 4.1-1C	Note 32		Relocated - TRM	