



Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381-2000

February 28, 2012

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Watts Bar Nuclear Plant, Unit 2
NRC Docket No. 50-391

10 CFR 50.36

Subject: Watts Bar Nuclear Plant (WBN) Unit 2 – Submittal of Developmental Revision G of the Unit 2 Technical Specification (TS) and Technical Specification Bases

Reference: E-mail from Justin Poole (NRC) to Gordon Arent (TVA) dated January 24, 2012, "Draft Questions on Technical Specifications"

This letter provides Developmental Revision G of the WBN Unit 2 TS and TS Bases.

Enclosure 1 provides a listing of the changes made to Revision F that are included in Revision G. Enclosure 2 responds to the questions and comments provided by the reference on the WBN Unit 2 Steam Generator Program TS. Enclosure 3 provides the annotated pages associated with the changes to the WBN Unit 2 TS and TS Bases. Enclosure 4 provides a clean copy of Revision G on optical media.

There are no new regulatory commitments associated with this letter.

I declare under penalty of perjury that the foregoing is true and correct to the best of my knowledge. Executed on the 28th day of February, 2012.

Respectfully,

Raymond A. Hruby, Jr.
General Manager, Technical Services
Watts Bar Unit 2

Enclosures (see page 2)

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Enclosures:

1. List of Changes to the WBN Unit 2 TS and TS Bases
2. Response to NRC Draft Questions on Technical Specifications
3. WBN Unit 2 TS and TS Bases Developmental Revision G Mark-ups
4. WBN U2 TS and TS Bases Developmental Revision G Optical Media

cc (Enclosures):

U. S. Nuclear Regulatory Commission
Region II
Marquis One Tower
245 Peachtree Center Ave., NE Suite 1200
Atlanta, Georgia 30303-1257

NRC Resident Inspector Unit 2
Watts Bar Nuclear Plant
1260 Nuclear Plant Road
Spring City, Tennessee 37381

Enclosure 1

List of Changes to the WBN U2 TS and TS Bases

- Change to SR 3.3.1.3 on NIS power and associated TS Bases SR 3.3.1.3.
- Change to TS Table 3.3.1-1, Source Range Neutron Flux Allowable Value.
- Change to TS Table 3.3.1-1, Reactor Coolant Pump (RCP) Undervoltage Allowable Value and Nominal Trip Setpoint, associated TS Bases B 3.3.1, and TS Bases B 3.3.1 References.
- Change to TS Table 3.3.1-1, Turbine Trip Low Fluid Oil Pressure Allowable Value and the applicable SR.
- Change to TS Table 3.3.2-1, Auxiliary Feedwater Trip of all Turbine Driven Main Feedwater Pumps Allowable Value.
- Change to SR 3.4.12.1 and 3.4.12.2, cold overpressure arming temperature.
- Change to TS Bases B 3.7.7 providing power and train alignments for the Component Cooling Water System.
- Change to TS Bases B 3.3.6 describing Containment Vent Isolation Instrumentation and associated changes to TS Basis B 3.3.8 Auxiliary Gas Treatment System Actuation Instrumentation, TS Bases B 3.7.12 on Auxiliary Building Gas Treatment System performance, and TS Bases B 3.9.8 Reactor Building Purge Air Cleanup Unit performance during refueling operations or fuel movement.
- Change to TS Bases B 3.8.4 on DC Sources to update the number of battery chargers needed for two-unit operation and to match current Unit 1 wording.
- Change to TS 4.3.1, Criticality describing acceptable spent fuel pool loading patterns.
- Changes to Limiting Condition for Operation (LCO) 3.4.17, TS Section 5.7.2.12 and associated Bases B 3.4-13 and B 3.4.17 to incorporate appropriate information from Technical Specification Task Force (TSTF) Traveler 510 R2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection," and to match the description to the Chapter 15.5 dose analysis for Main Steam Line breaks.

Enclosure 2

Response to NRC Draft Questions on Technical Specifications

The draft questions are provided below by number with the associated response below the individual question or related group of questions.

1. *Section 5.7.2.12.d.1 indicates that 100% of the tubing will be inspected in the first refueling outage. Since Watts Bar Unit 2 may replace steam generators (i.e., have a first refueling outage without replacing steam generators and may have a first refueling outage following steam generator replacement), please discuss your plans to modify this requirement to ensure both conditions are addressed (e.g., "...during the first refueling outage following steam generator installation.").*
2. *Since your proposed technical specifications were prepared, the NRC staff has approved Technical Specification Task Force Traveler 510 (TSTF-510), Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection" (ML110610350). The staff's review is summarized in a Notice of Availability dated October 19, 2011 (ML112101604). Since this TSTF Traveler addresses implementation issues associated with the inspections periods and addresses other administrative changes and clarifications which improve the technical specifications, please discuss your plans to incorporate the TSTF-510, Revision 2, changes into your proposed technical specifications.*

Response to questions 1 and 2:

The WBN U2 developmental TS has been revised to incorporate TSTF-510 on steam generator (SG) program inspection frequencies and tube sampling. This revision addresses the information requested in questions 1 and 2.

3. *On page B 3.4-73, the applicable safety analyses for operational leakage are provided. In this discussion, you indicate that the pre-accident primary-to-secondary leakage for a main steam line break is 1 gallon per minute from one steam generator. The standard technical specification wording indicates that the leakage may increase to this value during the accident. Please confirm that your accident analysis assumes that the leakage prior to a main steam line break is 1 gallon per minute (in which case the actual rate of leakage during the accident would be greater than 1 gallon per minute due to the higher differential pressure during the accident).*

Response to question 3:

The discussion on page B 3.4-73 has been revised to state that for the main steam line break (MSLB) the pre-accident primary-to-secondary leakage is 150 gallons per day (gpd) per SG. At the time of the rupture of the steam line, it is assumed that the total primary-to-secondary leak rate in the faulted SG is 1 gallon per minute (gpm). The primary-to-secondary leak rate in the non-faulted SGs is maintained at 150 gpd per SG.

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Response to NRC Draft Questions on Technical Specifications

4. *On page B 3.4-73, the applicable safety analyses for operational leakage is provided. There are two paragraphs from the standard technical specifications that are not included in this section for Watts Bar Unit 2. These paragraphs address other accidents that involve secondary steam release to the atmosphere and the assumptions for a steam generator tube rupture [SGTR] accident. Please discuss why these paragraphs were not included in your Bases.*

Response to question 4:

One reason that these paragraphs are not included in our Bases is because WBN does not conform to revision 3 of the Standard TS (STS). They match the Unit 1 TS to the extent possible. Unit 1's TS generally match a mix of revision 0 and revision 1 of the STS but do not fully conform to those standards.

More importantly, the paragraphs do not reflect the WBN U2 safety analyses. The STS paragraph on SGTR states "The FSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via safety valves and the majority is steamed to the condenser. The [1 gpm] primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential."

While the write-up in the paragraph above represents the most probable and desirable plant response to a SGTR event, the WBN accident analyses use more conservative assumptions. The WBN dose analysis for the SGTR assumes a relatively large secondary side release from the faulted SG directly to the atmosphere for the period of 0 to 2 hours and a smaller atmospheric release for the 2 to 8 hour period. In addition, the SGTR analysis is the controlling analysis for control room operator dose due to the assumptions mandated for the Iodine spike cases.

The second paragraph in the STS but not in the WBN U2 specifications states, "The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes the entire [1 gpm] primary to secondary LEAKAGE is through the affected generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., a small fraction of these limits)." While the MSLB is more limiting than the SGTR for offsite dose consequences at WBN, as noted in the preceding paragraph that is not the case for control room doses.

TVA concludes that the current discussion on page B 3.4-73, as amended by response 3 above, correctly reflects the WBN U2 accident dose analyses and that inclusion of the two paragraphs from the STS discussed above would not be appropriate.

5. *On page B 3.4-97, the applicable safety analyses for steam generator tube integrity is provided. Please clarify the following sentence: "In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from*

Enclosure 2

Response to NRC Draft Questions on Technical Specifications

150 gallons per day (gpd) per steam generator and 1 gallon per minute (gpm) in the faulted steam generator.” In addition, the standard technical specifications make it clear that the leakage may increase as a result of accident induced conditions to a specific value. This thought is not captured in your proposal. Please clarify why this was not included in your Bases.

Response to question 5:

The change in primary-to-secondary leakage in a faulted SG at the start of a MSLB is an assumption made in the dose analysis to add conservatism. The assumed higher leak rate is not the result of a specific physical process or analysis. There is no regulatory requirement to assume a total SG leak rate above the TS allowable in the dose analysis. This accepted regulatory position is shown clearly in Regulatory Guide 1.195, “Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors,” which states “For facilities with traditional SG specifications (both per generator and total of all generators), the leakage should be apportioned between affected and unaffected SGs in such a manner that the calculated dose is maximized. For example, for a four loop facility with a limiting condition for operation of 500 gpd for any one generator not to exceed 1 gpm from all generators, it would be appropriate to assign 500 gpd to the faulted generator and 313 gpd to each of the unaffected generators.” In this example the tube leak rate does not increase at all after an accident. [Note that the sum of 500 gpd + (3x313 gpd)= 1 gpm.] The approach to post-accident SG tube leakage prescribed in this Regulatory Guide is found in other NRC regulatory guidance as well. Thus, the WBN approach is conservative but not phenomenological. The wording provided in the Unit 2 TSB is the same as is in the Unit 1 TSB for this section. Thus for both technical accuracy and licensing basis consistency, the current wording is appropriate.

Enclosure 3

WBN Unit 2 TS and TS Bases Developmental Revision G Mark-ups

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.3	<p>-----NOTES-----</p> <ol style="list-style-type: none"> 1. Adjust NIS channel if absolute difference is $\geq 3\%$. 2. Required to be performed within 96 hours after THERMAL POWER is $\geq 4525\%$ RTP. <p>-----</p> <p>Compare results of the PDMS measurements to NIS AFD.</p>	31 effective full power days (EFPD)
SR 3.3.1.4	<p>-----NOTE-----</p> <p>This Surveillance must be performed on the reactor trip bypass breaker prior to placing the bypass breaker in service.</p> <p>-----</p> <p>Perform TADOT.</p>	62 days on a STAGGERED TEST BASIS
SR 3.3.1.5	Perform ACTUATION LOGIC TEST.	92 days on a STAGGERED TEST BASIS
SR 3.3.1.6	<p>-----NOTE-----</p> <p>Required to be performed within 6 days after THERMAL POWER is $\geq 50\%$ RTP.</p> <p>-----</p> <p>Calibrate excore channels to agree with the PDMS measurements.</p>	92 EFPD

(continued)

Table 3.3.1-1 (page 2 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
5. Source Range Neutron Flux	2 ^(f)	2	I, J	SR 3.3.1.1 SR 3.3.1.8 ^{(b)(c)} SR 3.3.1.11 ^{(b)(c)}	≤ 1.335 E5 cps ps	1.0 E5 cps
	3 ^(a) , 4 ^(a) , 5 ^(a)	2	J, K	SR 3.3.1.1 SR 3.3.1.8 ^{(b)(c)} SR 3.3.1.11 ^{(b)(c)} SR 3.3.1.15	≤ 1.335 E5 cps	1.0 E5 cps
	3 ^(g) , 4 ^(g) , 5 ^(g)	1	L	SR 3.3.1.1 SR 3.3.1.11 ^{(b)(c)}	N/A	N/A
6. Overtemperature ΔT	1, 2	4	W	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 ^{(b)(c)} SR 3.3.1.10 ^{(b)(c)} SR 3.3.1.15	Refer to Note 1 (Page 3.3-21)	Refer to Note 1 (Page 3.3-21)
7. Overpower ΔT	1, 2	4	W	SR 3.3.1.1 SR 3.3.1.7 ^{(b)(c)} SR 3.3.1.10 ^{(b)(c)} SR 3.3.1.15	Refer to Note 2 (Page 3.3-22)	Refer to Note 2 (Page 3.3-22)
8. Pressurizer Pressure						
a. Low	1 ^(h)	4	X	SR 3.3.1.1 SR 3.3.1.7 ^{(b)(c)} SR 3.3.1.10 ^{(b)(c)} SR 3.3.1.15	≥ 1964.8 psig	1970 psig
b. High	1, 2	4	W	SR 3.3.1.1 SR 3.3.1.7 ^{(b)(c)} SR 3.3.1.10 ^{(b)(c)} SR 3.3.1.15	≤ 2390.2 psig	2385 psig

(continued)

- (a) With Reactor Trip Breakers (RTBs) closed and Rod Control System capable of rod withdrawal.
- (b) If the as found channel setpoint is outside its predefined as found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.
- (c) The instrument channel setpoint shall be reset to a value that is within the as left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. The methodologies used to determine the as found and as left tolerances for the NTSP are specified in FSAR Section 7.1.2.
- (f) Below the P-6 (Intermediate Range Neutron Flux) interlocks.
- (g) With the RTBs open. In this condition, source range Function does not provide reactor trip but does provide indication.
- (h) Above the P-7 (Low Power Reactor Trips Block) interlock.

Table 3.3.1-1 (page 3 of 9)
Reactor Trip System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
9.	Pressurizer Water Level-High	1 ^(h)	3	X	SR 3.3.1.1 SR 3.3.1.7 ^{(b)(c)} SR 3.3.1.10 ^{(b)(c)}	≤ 92.7% span	92% span
10.	Reactor Coolant Flow - Low	1 ^(h)	3 per loop	N	SR 3.3.1.1 SR 3.3.1.7 ^{(b)(c)} SR 3.3.1.10 ^{(b)(c)} SR 3.3.1.15	≥ 89.7% flow	90% flow
11.	Undervoltage RCPs	1 ^(h)	1 per bus	M	SR 3.3.1.9 SR 3.3.1.10 ^{(b)(c)} SR 3.3.1.15	≥ 47345112 V	48305400 V
12.	Underfrequency RCPs	1 ^(h)	1 per bus	M	SR 3.3.1.9 SR 3.3.1.10 ^{(b)(c)} SR 3.3.1.15	≥ 56.9 Hz	57.5 Hz

(continued)

- (b) If the as found channel setpoint is outside its predefined as found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.
- (c) The instrument channel setpoint shall be reset to a value that is within the as left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. The methodologies used to determine the as found and as left tolerances for the NTSP are specified in FSAR Section 7.1.2.
- (h) Above the P-7 (Low Power Reactor Trips Block) interlock.

Table 3.3.1-1 (page 4 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
13. SG Water Level – Low-Low	1, 2	3/SG	U	SR 3.3.1.1 SR 3.3.1.7 ^{(b)(c)} SR 3.3.1.10 ^{(b)(c)} SR 3.3.1.15	≥ 16.4% of narrow range span	17% of narrow range span
Coincident with:						
a) Vessel ΔT Equivalent to power ≤ 50% RTP	1, 2	3	V	SR 3.3.1.7 ^{(b)(c)} SR 3.3.1.10 ^{(b)(c)}	Vessel ΔT variable input ≤ 52.6% RTP	Vessel ΔT variable input 50% RTP
With a time delay (T _s) if one steam generator is affected					≤ 1.01 T _s (Refer to Note 3, Page 3.3-23)	T _s (Refer to Note 3, Page 3.3-23)
or						
A time delay (T _m) if two or more steam generators are affected					≤ 1.01 T _m (Refer to Note 3, Page 3.3-23)	T _m (Refer to Note 3, Page 3.3-23)
b) Vessel ΔT Equivalent to power > 50% RTP with no time delay (T _s and T _m = 0)	1, 2	3	V	SR 3.3.1.7 ^{(b)(c)} SR 3.3.1.10 ^{(b)(c)}	Vessel ΔT variable input ≤ 52.6% RTP	Vessel ΔT variable input 50% RTP
14. Turbine Trip						
a. Low Fluid Oil Pressure	1 ⁽ⁱ⁾	3	O	SR 3.3.1.14 10 ^{(b)(c)} SR 3.3.1.14	≥ 43 38.3 psi g	45 psig
b. Turbine Stop Valve Closure	1 ⁽ⁱ⁾	4	Y	SR 3.3.1.10 SR 3.3.1.14	≥ 1% open	1% open

(continued)

(b) If the as found channel setpoint is outside its predefined as found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. The methodologies used to determine the as found and as left tolerances for the NTSP are specified in FSAR Section 7.1.2.

(i) Above the P-9 (Power Range Neutron Flux) interlock.

Table 3.3.2-1 (page 6 of 8)
Engineered Safety Feature Actuation System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
6.	Auxiliary Feedwater (continued)						
d.	Loss of Offsite Power	1, 2, 3	4 per bus	F	Refer to Function 4 of Table 3.3.5-1 for SRs and Allowable Values. Notes (b) and (c) are applicable to SR 3.3.5.2 for this function.		
e.	Trip of all Turbine Driven Main Feedwater Pumps	1 ⁽ⁱ⁾ , 2 ^(k)	1 per pump	J	SR 3.3.2.8 ^{(b)(c)} SR 3.3.2.9 ^{(b)(c)} SR 3.3.2.10	≥ 48 43.3 psig	50 psig
f.	Auxiliary Feedwater Pumps Train A and B Suction Transfer on Suction Pressure - Low	1, 2, 3	3	F	SR 3.3.2.6 SR 3.3.2.9 ^{(b)(c)} SR 3.3.2.10	A) ≥ 0.5 psig B) ≥ 1.33 psig	A) 1.2 psig B) 2.0 psig
7.	Automatic Switchover to Containment Sump						
a.	Automatic Actuation Logic and Actuation Relays	1, 2, 3, 4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
b.	Refueling Water Storage Tank (RWST) Level - Low	1, 2, 3, 4	4	K	SR 3.3.2.1 SR 3.3.2.4 ^{(b)(c)} SR 3.3.2.9 ^{(b)(c)} SR 3.3.2.10	≥ 155.6 inches from Tank Base	158 inches from Tank Base
	Coincident with Safety Injection and Coincident with Containment Sump Level - High	Refer to Function 1 (Safety Injection) for all initiation functions and requirements. 1, 2, 3, 4	4	K	SR 3.3.2.1 SR 3.3.2.4 ^{(b)(c)} SR 3.3.2.9 ^{(b)(c)} SR 3.3.2.10	≥ 37.2 inches above el. 702.8 ft	38.2 inches above el. 702.8 ft

(continued)

- (b) If the as found channel setpoint is outside its redefined as found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.
- (c) The instrument channel setpoint shall be reset to a value that is within the as left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. The methodologies used to determine the as found and as left tolerances for the NTSP are specified in FSAR Section 7.1.2.
- (j) Entry into Condition J may be suspended for up to 4 hours when placing the second Turbine Driven Main Feedwater (TDMFW) Pump in service or removing one of two TDMFW pumps from service.
- (k) When one or more Turbine Driven Feedwater Pump(s) are supplying feedwater to steam generators.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	Verify no safety injection pumps are capable of injecting into the RCS.	<p>Within 4 hours after entering MODE 4 from MODE 3 prior to the temperature of one or more RCS cold legs decreasing below 325225°F.</p> <p><u>AND</u></p> <p>12 hours thereafter</p>
SR 3.4.12.2	Verify a maximum of one charging pump is capable of injecting into the RCS.	<p>Within 4 hours after entering MODE 4 from MODE 3 prior to the temperature of one or more RCS cold legs decreasing below 325225°F.</p> <p><u>AND</u></p> <p>12 hours thereafter</p>
SR 3.4.12.3	Verify each accumulator is isolated.	12 hours
SR 3.4.12.4	<p>-----NOTE-----</p> <p>Only required to be performed when complying with LCO 3.4.12.b.</p> <p>-----</p> <p>Verify RCS vent open.</p>	<p>12 hours for unlocked open vent paths</p> <p><u>AND</u></p> <p>31 days for locked open vent paths</p>

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained

AND

All SG tubes satisfying the tube **plugging or** repair criteria shall be plugged **or repaired** in accordance with the Steam Generator Program.

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

ACTIONS

-----NOTE-----

Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube plugging or repair criteria and not plugged or repaired in accordance with the Steam Generator Program	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection	7 days
	<u>AND</u> A.2 Plug or repair the affected tube(s) in accordance with the Steam Generator Program	Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met <u>OR</u> SG tube integrity not maintained	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify steam generator tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube plugging or repair criteria is plugged or repaired in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

- 4.3.1.1 The spent fuel storage racks (shown in Figure 4.3-1) are designed and shall be maintained with:
- a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;
 - b. $k_{\text{eff}} \leq 0.95$ if fully flooded with unborated water, which, includes an allowance for uncertainties as described in Sections 4.3.2.7 and 9.1 of the FSAR;
 - c. Distances between fuel assemblies are a nominal 10.375 inch center-to-center spacing in the twenty-four flux trap rack modules.
 - d. ~~Fuel assemblies with enrichments less than or equal to 3.80 weight percent U-235 are allowed unrestricted storage.~~
 - e. Fuel assemblies with initial enrichments greater than 3.80 weight percent and less than a maximum of 5 percent enrichment (nominally 4.95 ± 0.05 percent) may be stored in the spent fuel racks in one of four arrangements with specific limits as identified below:
 1. **New and spent** fuel assemblies may be stored in the racks **in an all cell arrangement without further restrictions** provided the burnup of each assembly is in the acceptable domain identified in Figure 4.3-3, depending upon the specified initial enrichment.
 2. New and spent fuel assemblies may be stored in a checkerboard arrangement of 2 new and 2 spent assemblies, provided that each spent fuel assembly has accumulated a minimum burnup in the acceptable domain identified in Figure 4.3-4.
 3. New fuel assemblies may be stored in 4-cell arrays with 1 of the 4 cells remaining empty of fuel (i.e. containing only water or water with up to 75 percent by volume of non-fuel bearing material).

(continued)

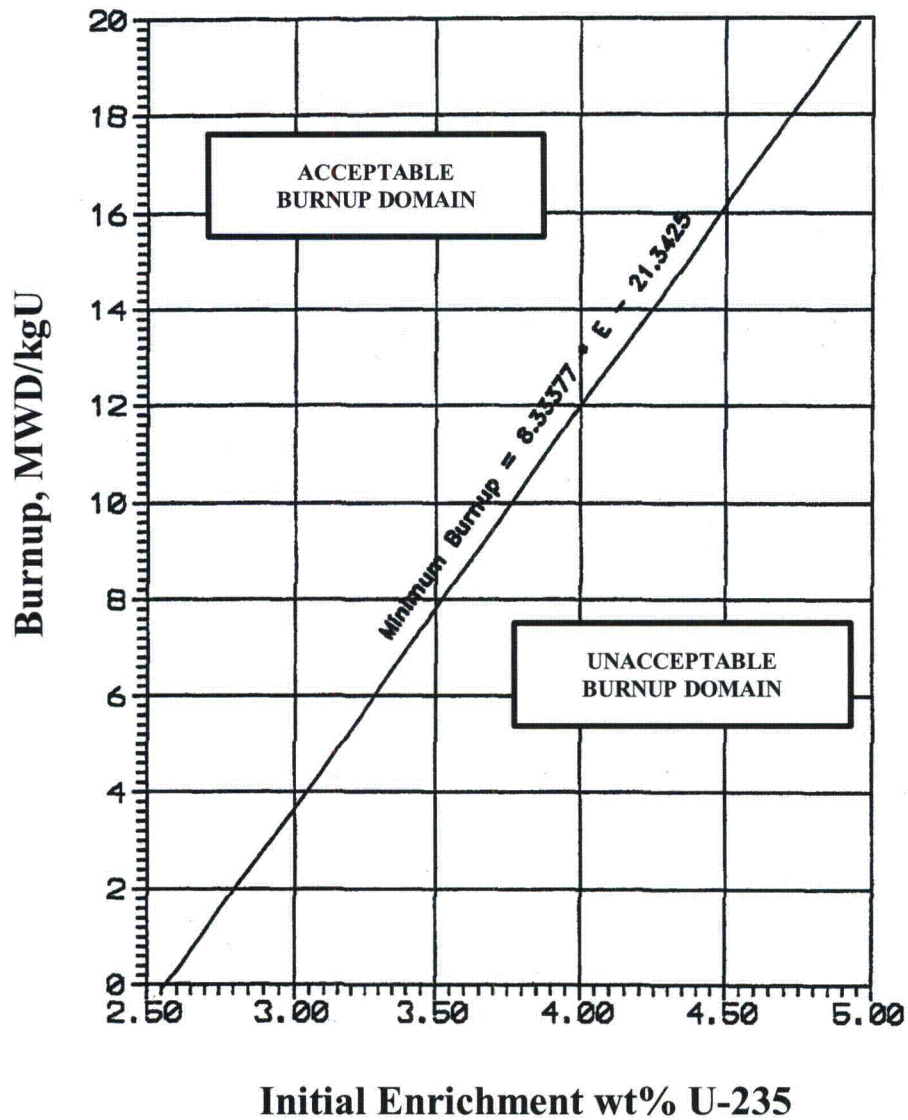


FIGURE 4.3-4
MINIMUM REQUIRED BURNUP FOR ~~A 2X2~~ CHECKERBOARD ARRANGEMENT OF
2 SPENT ~~AND FUEL ASSEMBLIES WITH~~ 2 NEW FUEL ASSEMBLIES OF 5% ENRICHMENT
(MAXIMUM)

5.7 Procedures, Programs, and Manuals (continued)

5.7.2.12 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following ~~provisions~~:

- a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging **or repair** of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged, to confirm that the performance criteria are being met.
- b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.
 1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, cooldown), ~~and~~ all anticipated transients included in the design specification, ~~and~~ design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

(continued)

5.7 Procedures, Programs, and Manuals (continued)

5.7.2.12 Steam Generator (SG) Program (continued)

2. Accident induced leakage performance criterion: The primary-to-secondary accident induced leakage rate for any design basis accident, other than an SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. **Leakage is not to exceed 1 gpm per SG, except for specific types of degradation at specific locations as described in paragraph c. of the Steam Generation Program.** ~~For design basis accidents that have a faulted steam generator, accident induced leakage is not to exceed 1.0 gallon per minute (gpm) for the faulted steam generator and 150 gallons per day (gpd) for the non-faulted steam generators. For design basis accidents that do not have a faulted steam generator, accident induced leakage is not to exceed 150 gpd per steam generator.~~
3. The operational leakage performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."
- c. Provisions for SG tube **plugging or** repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged **or repaired**.
- d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube **plugging or** repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. ~~A n-assessment-of~~ **assessment** shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

(continued)

5.7 Procedures, Programs, and Manuals (continued)

1. Inspect 100% of the tubes in each SG during the first refueling outage **following SG installation.**

5.7.2.12 Steam Generator (SG) Program (continued)

2. ~~Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SGs shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.~~ **After the first refueling outage following SG installation, inspect each SG at least every 24 effective full power months or at least every refueling outage (whichever results in more frequent inspections). In addition, inspect 100% of the tubes at sequential periods of 60 effective full power months beginning after the first refueling outage inspection following SG installation. Each 60 effective full power month inspection period may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube repair criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period.**
3. If crack indications are found in any SG tube, then the next inspection for each **affected and potentially affected** SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever ~~is less~~ **results in more frequent**

(continued)

5.7 Procedures, Programs, and Manuals (continued)

inspections). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

- e. Provisions for monitoring operational primary-to-secondary LEAKAGE.

5.7.2.13 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

- a. Identification of a sampling schedule for the critical variables and control points for these variables;
- b. Identification of the procedures used to measure the values of the critical variables;
- c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;
- d. Procedures for the recording and management of data;

Secondary Water Chemistry Program (continued)

- 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.

(continued)

5.9 Reporting Requirements (continued)

5.9.7 EDG Failures 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.4, or existing Regulatory Guide 1.108 reporting requirement.

5.9.8 PAMS Report

When a Report is required by Condition B or F of LCO 3.3.3, "Post Accident Monitoring (PAM) 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.

5.9.9 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.7.2.12, Steam Generator (SG) Program. The report shall include:

- a. The scope of inspections performed on each SG,
 - b. Active degradation mechanisms found,
 - c. Nondestructive examination techniques utilized for each degradation mechanism,
 - d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,
 - e. Number of tubes plugged **or repaired** during the inspection outage for each active degradation mechanism,
 - f. ~~Total number and percentage of tubes plugged to date~~ **The number and percentage of tubes plugged or repaired to date, and the effective plugging percentage in each SG,**
 - g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and
 - h. ~~The effective plugging percentage for all plugging in each SG~~ **Repair method utilized and the number of tubes repaired by each repair method.**
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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

10. Reactor Coolant Flow - Low (continued)

In MODE 1 above the P-8 setpoint, a loss of flow in one RCS loop could result in DNB conditions in the core because of the higher power level. In MODE 1 below the P-8 setpoint and above the P-7 setpoint, a loss of flow in two or more loops is required to actuate a reactor trip because of the lower power level and the greater margin to the design limit DNBR. Below the P-7 setpoint, all reactor trips on low flow are automatically blocked since there is insufficient heat production to generate DNB conditions.

The Reactor Coolant Flow-Low Trip Setpoint and Allowable Value are specified in % indicated loop flow; however, the Eagle-21™ values entered through the MMI are specified in an equivalent % differential pressure.

11. Undervoltage Reactor Coolant Pumps

The Undervoltage RCPs trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops. The voltage to each RCP is monitored. Above the P-7 setpoint, a loss of voltage detected on two or more RCP buses will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow - Low Trip Setpoint is reached in two or more RCS loops. The loss of voltage in two loops must be sustained for a length of time equal to or greater than that set in the time delay. Time delays are incorporated into the Undervoltage RCPs channels to prevent reactor trips due to momentary electrical power transients.

The "Allowable Value" of greater than or equal to 5112 V includes the 3.5 cycle relay response time. This is required to ensure a reactor trip occurs before reaching a safety limit.

The undervoltage relay "Nominal Trip Setpoint" value of 5400 V includes the 3.5 cycle relay response time. This is required to meet the reactor trip nominal setpoint of 4830 V (Reference 17).

The LCO requires one Undervoltage RCP channel per bus to be OPERABLE.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.2 (continued)

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

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

SR 3.3.1.3

SR 3.3.1.3 compares the power distribution measurement to the NIS channel AFD output every 31 EFPD. If the absolute difference is $\geq 3\%$, the NIS channel is still OPERABLE, but must be readjusted. If the NIS channel cannot be properly readjusted, the channel is declared inoperable. This Surveillance is performed to verify the $f(\Delta I)$ input to the Overtemperature ΔT Function. The incore power distribution measurement is obtained using the OPERABLE Power Distribution Monitoring System (PDMS) (Ref. 16).

Two Notes modify SR 3.3.1.3. Note 1 indicates that the excore NIS channel shall be adjusted if the absolute difference between the incore and excore AFD is $\geq 3\%$. Note 2 clarifies that the Surveillance is required only if reactor power is $\geq 4525\%$ RTP and that 96 hours is allowed for performing the first Surveillance after reaching 4525% RTP. This surveillance is typically performed at greater than or equal to 50% RTP to ensure the results of the evaluation are more accurate and the adjustments more reliable. Ninety-six (96) hours are allowed to ensure Xenon stability and allow for instrumentation alignments.

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

(continued)

BASES

REFERENCES
(continued)

10. Deleted
 11. WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," January 1996
 12. WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," October 1998.
 13. Deleted
 14. WCAP-14333 P-A, Revision 1, "Probablistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times," October 1998.
 15. WCAP-15376-P-A, Revision 1, "Risk Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," March 2003
 16. WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994 (Addendum 2, April 2002).
 17. **TVA Calculation WBPE0689009007, "Demonstrated Accuracy Calculation for Reactor Coolant Pump Undervoltage"**
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B 3.3 INSTRUMENTATION

B 3.3.6 Containment Vent Isolation Instrumentation

BASES

BACKGROUND

Containment Vent Isolation Instrumentation closes the containment isolation valves in the Containment Purge System. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. The Reactor Building Purge System may be in use during reactor operation and with the reactor shutdown.

Containment vent isolation is initiated by a safety injection (SI) signal or by manual actuation. The Bases for LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," discuss initiation of SI signals.

Redundant and independent gaseous radioactivity monitors measure the radioactivity levels of the containment purge exhaust, each of which will initiate its associated train of automatic Containment Vent Isolation upon detection of high gaseous radioactivity.

The Reactor Building Purge System has inner and outer containment isolation valves in its supply and exhaust ducts. This system is described in the Bases for LCO 3.6.3, "Containment Isolation Valves."

The plant design basis requires that when moving irradiated fuel in the Auxiliary Building and/or Containment with the Containment open to the Auxiliary Building ABSCE spaces, a signal from the spent fuel pool radiation monitors 0-RE-90-102 and -103 will initiate a Containment Ventilation Isolation (CVI) in addition to their normal function. In addition, a signal from the containment purge radiation monitors 2-RE-90-130, and -131 or other CVI signal will initiate that portion of the Auxiliary Building Isolation (ABI) normally initiated by the spent fuel pool radiation monitors.

Additionally, a Containment Isolation Phase A (SI signal) from the operating unit, high temperature in the Auxiliary Building air intakes, or manual ABI will cause a CVI signal in the refueling unit. In the case where the containment of both units is open to the Auxiliary Building spaces, a CVI in one unit will initiate a CVI in the other unit in order to maintain those spaces open to the ABSCE. Therefore, the containment ventilation instrumentation must remain

(continued)

BASES (continued)

BACKGROUND (continued)	operable when moving irradiated fuel in the Auxiliary Building if the containment air locks, penetrations, equipment hatch, etc. are open to the Auxiliary Building ABSCE spaces.
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APPLICABLE SAFETY ANALYSES	<p>The containment isolation valves for the Reactor Building Purge System close within six seconds following the DBA. The containment vent isolation radiation monitors act as backup to the SI signal to ensure closing of the purge air system supply and exhaust valves. They are also the primary means for automatically isolating containment in the event of a fuel handling accident during shutdown. Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accidental offsite radiological doses are below 10 CFR 100 (Ref. 1) limits.</p>
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The Containment Vent Isolation instrumentation satisfies Criterion 3 of the NRC Policy Statement.

When moving irradiated fuel inside containment or in the Auxiliary Building with containment air locks or penetrations open to the Auxiliary Building ABSCE spaces, or when moving fuel in the Auxiliary Building with the containment equipment hatch open, the provisions to initiate a CVI from the spent fuel pool radiation monitors and to initiate an ABI (i.e., the portion of an ABI normally initiated by the spent fuel pool radiation monitors) from a CVI, including a CVI generated by the containment purge monitors, in the event of a fuel handling accident (FHA) must be in place and functioning. **Additionally, a Containment Isolation Phase A (SI signal) from the operating unit, high temperature in the Auxiliary Building air intakes, or manual ABI will cause a CVI signal in the refueling unit.** The containment equipment hatch cannot be open when moving irradiated fuel inside containment in accordance with Technical Specification 3.9.4.

The ABGTS is required to be operable during movement of irradiated fuel in the Auxiliary Building during any mode and during movement of irradiated fuel in the Reactor Building when the Reactor Building is established as part of the ABSCE boundary (see TS 3.3.8, 3.7.12, & 3.9.4). When moving irradiated fuel inside containment, at least one train of the containment purge system must be operating or the containment must be isolated. When moving irradiated fuel in the Auxiliary Building during times when the containment is open to the Auxiliary Building ABSCE spaces, containment purge can be operated, but operation of the system is not required. However, whether the containment purge system is operated or not in this configuration, all containment ventilation isolation valves and associated instrumentation must remain operable.

(continued)

BASES (continued)

APPLICABLE
SAFETY
ANALYSES
(continued)

This requirement is necessary to ensure a CVI can be accomplished from the spent fuel pool radiation monitors in the event of an FHA in the Auxiliary Building. **Additionally, a Containment Isolation Phase A (SI signal) from the operating unit, high temperature in the Auxiliary Building air intakes, or manual ABI will cause a CVI signal in the refueling unit.**

LCO

The LCO requirements ensure that the instrumentation necessary to initiate Containment Vent Isolation, listed in Table 3.3.6-1, is OPERABLE.

1. Manual Initiation

The LCO requires two channels OPERABLE. The operator can initiate Containment Vent Isolation at any time by using either of two switches in the control room or from local panel(s). Either switch actuates both trains. This action will cause actuation of all components in the same manner as any of the automatic actuation signals. These manual switches also initiate a Phase A isolation signal.

The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

Each channel consists of one selector switch and the interconnecting wiring to the actuation logic cabinet.

2. Automatic Actuation Logic and Actuation Relays

The LCO requires two trains of Automatic Actuation Logic and Actuation Relays OPERABLE to ensure that no single random failure can prevent automatic actuation.

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b, SI. The applicable MODES and specified conditions for the containment vent isolation portion of the SI Function is different and less restrictive than those for the SI role. If one or more of the SI Functions becomes inoperable in such a manner that only the Containment Vent Isolation Function is affected, the Conditions applicable to the SI Functions need not be entered. The less restrictive Actions specified for inoperability of the Containment Vent Isolation Functions specify sufficient compensatory measures for this case.

(continued)

B 3.3 INSTRUMENTATION

B 3.3.8 Auxiliary Building Gas Treatment (ABGTS) Actuation Instrumentation

BASES

BACKGROUND

The ABGTS ensures that radioactive materials in the fuel building atmosphere following a fuel handling accident or a loss of coolant accident (LOCA) are filtered and adsorbed prior to exhausting to the environment. The system is described in the Bases for LCO 3.7.12, "Auxiliary Building Gas Treatment System (ABGTS)." The system initiates filtered exhaust of air from the fuel handling area, ECCS pump rooms, and penetration rooms automatically following receipt of a fuel pool area high radiation signal or a Containment Phase A Isolation signal. Initiation may also be performed manually as needed from the main control room.

High area radiation, monitored by either of two monitors, provides ABGTS initiation. Each ABGTS train is initiated by high radiation detected by a channel dedicated to that train. There are a total of two channels, one for each train. High radiation detected by any monitor or a Phase A isolation signal from the Engineered Safety Features Actuation System (ESFAS) initiates auxiliary building isolation and starts the ABGTS. These actions function to prevent exfiltration of contaminated air by initiating filtered ventilation, which imposes a negative pressure on the Auxiliary Building Secondary Containment Enclosure (ABSCE).

The plant design basis requires that when moving irradiated fuel in the Auxiliary Building and/or Containment with the Containment and/or annulus open to the Auxiliary Building ABSCE spaces, a signal from the spent fuel pool radiation monitors 0-RE-90-102 and -103 will initiate a Containment Ventilation Isolation (CVI) in addition to their normal function. In addition, a signal from the containment purge radiation monitors 2-RE-90-130, and -131 or other CVI signal will initiate that portion of the Auxiliary Building Isolation (ABI) normally initiated by the spent fuel pool radiation monitors. **Additionally, a Containment Isolation Phase A (SI signal) from the operating unit, high temperature in the Auxiliary Building air intakes, or manual ABI will cause a CVI signal in the refueling unit. In the case where the containment of both units is open to the Auxiliary Building spaces, a CVI in one unit will initiate a CVI in the other unit in order to maintain those spaces open to the ABSCE.** Therefore, the containment ventilation instrumentation must remain operable when

(continued)

BASES (continued)

BACKGROUND
(continued)

moving irradiated fuel in the Auxiliary Building if the containment and/or annulus air locks, penetrations, equipment hatch, etc. are open to the Auxiliary Building ABSCE spaces.

APPLICABLE
SAFETY
ANALYSES

The ABGTS ensures that radioactive materials in the ABSCE atmosphere following a fuel handling accident or a LOCA are filtered and adsorbed prior to being exhausted to the environment. This action reduces the radioactive content in the auxiliary building exhaust following a LOCA or fuel handling accident so that offsite doses remain within the limits specified in 10 CFR 100 (Ref. 1).

The ABGTS Actuation Instrumentation satisfies Criterion 3 of the NRC Policy Statement.

When moving irradiated fuel inside containment or in the Auxiliary Building with containment air locks or penetrations open to the Auxiliary Building ABSCE spaces, or when moving fuel in the Auxiliary Building with the containment equipment hatch open, the provisions to initiate a CVI from the spent fuel pool radiation monitors and to initiate an ABI (i.e., the portion of an ABI normally initiated by the spent fuel pool radiation monitors) from a CVI, including a CVI generated by the containment purge monitors, in the event of a fuel handling accident (FHA) must be in place and functioning. **Additionally, a Containment Isolation Phase A (SI signal) from the operating unit, high temperature in the Auxiliary Building air intakes, or manual ABI will cause a CVI signal in the refueling unit.** The containment equipment hatch cannot be open when moving irradiated fuel inside containment in accordance with Technical Specification 3.9.4.

The ABGTS is required to be operable during movement of irradiated fuel in the Auxiliary Building during any mode and during movement of irradiated fuel in the Reactor Building when the Reactor Building is established as part of the ABSCE boundary (see TS 3.3.8, 3.7.12, & 3.9.4). When moving irradiated fuel inside containment, at least one train of the containment purge system must be operating or the containment must be isolated. When moving irradiated fuel in the Auxiliary Building during times when the containment is open to the Auxiliary Building ABSCE spaces, containment purge can be operated, but operation of the system is not required. However, whether the containment purge system is operated or not in this configuration, all containment ventilation isolation valves and associated instrumentation must remain operable.

(continued)

BASES (continued)

APPLICABLE
SAFETY
ANALYSES
(continued)

This requirement is necessary to ensure a CVI can be accomplished from the spent fuel pool radiation monitors in the event of a FHA in the Auxiliary Building. **Additionally, a Containment Isolation Phase A (SI signal) from the operating unit, high temperature in the Auxiliary Building air intakes, or manual ABI will cause a CVI signal in the refueling unit. In the case where the containment of both units is open to the Auxiliary Building spaces, a CVI in one unit will initiate a CVI in the other unit in order to maintain those spaces open to the ABSCE.**

LCO

The LCO requirements ensure that instrumentation necessary to initiate the ABGTS is OPERABLE.

1. Manual Initiation

The LCO requires two channels OPERABLE. The operator can initiate the ABGTS at any time by using either of two switches in the control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

Each channel consists of one hand switch and the interconnecting wiring to the actuation logic relays.

2. Fuel Pool Area Radiation

The LCO specifies two required Fuel Pool Area Radiation Monitors to ensure that the radiation monitoring instrumentation necessary to initiate the ABGTS remains OPERABLE. One radiation monitor is dedicated to each train of ABGTS.

For sampling systems, channel OPERABILITY involves more than OPERABILITY of channel electronics. OPERABILITY may also require correct valve lineups, sample pump operation, and filter motor operation, as well as detector OPERABILITY, if these supporting features are necessary for trip to occur under the conditions assumed by the safety analyses.

(continued)

BASES (continued)

APPLICABLE
SAFETY
ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for a main steam line break (MSLB) assumes that the ~~pre~~-accident primary-to-secondary LEAKAGE from three steam generators is 150 gallons per day (gpd) per steam generator and 1 gallon per minute (gpm) from one steam generator. ~~This leakage assumption remains the same after the accident.~~ For an SGTR accident, the accident analysis assumes a primary-to-secondary leakage of 150 gpd per steam generator prior to the accident. Subsequent to the SGTR a leakage of 150 gpd is assumed in each of three intact steam generators and RCS blowdown flow through the ruptured tube in the faulted steam generator. Consequently, the LCO requirement to limit primary-to-secondary LEAKAGE through any one steam generator to less than or equal to 150 gpd is acceptable.

The safety analysis for the SLB accident assumes the entire 1 gpm primary-to-secondary LEAKAGE is through the affected steam generator as an initial condition. The dose consequences resulting from the SLB accident are within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., a small fraction of these limits).

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of an off-normal condition. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

BASES

BACKGROUND (continued)	The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).
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APPLICABLE SAFETY ANALYSES	<p>The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of an SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.13, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via safety valves and the majority is discharged to the main condenser.</p>
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The analysis for design basis accidents and transients other than an SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from 150 gallons per day (gpd) per steam generator and 1 gallon per minute (gpm) in the faulted steam generator. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), and 10 CFR 100 (Ref. 3) or the NRC approved licensing basis.

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO	The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the plugging or repair criteria be plugged or repaired in accordance with the Steam Generator Program.
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During an SG inspection, any inspected tube that satisfies the Steam Generator Program **plugging or** repair criteria is **repaired or** removed from service by plugging. If a tube was determined to satisfy the **plugging or** repair criteria but was not plugged **or repaired**, the tube may still have tube integrity.

(continued)

BASES

LCO
(continued)

In the context of this Specification, an SG tube is defined as the entire length of the tube, including the tube wall **and any repairs made to it**, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

An SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.7.2.12, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

(continued)

BASES

ACTIONS
(continued)

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube **plugging or** repair criteria but were not plugged **or repaired** in accordance with the Steam Generator Program as required by SR 3.4.17.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG **plugging or** repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if an SG tube that should have been plugged **or repaired**, has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged **or repaired** prior to entering MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.4.17.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube **plugging or** repair criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, nondestructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.17.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.7.2.12 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections. **If crack indications are found in any SG tube, the maximum inspection interval for all affected and potentially affected SGs is restricted by Specification 5.7.2.12 until subsequent inspections support extending the inspection interval.**

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.4.17.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program **plugging or** repair criteria is removed from service by plugging. The tube **plugging or** repair criteria delineated in Specification 5.7.2.12 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube **plugging or** repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

Steam Generator tube repairs are only performed using approved repair methods as described in the Steam Generator Program.

The Frequency of prior to entering MODE 4 following an SG inspection ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged prior to subjecting the SG tubes to significant primary-to-secondary pressure differential.

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."
 2. 10 CFR 50 Appendix A, GDC 19, Control Room.
 3. 10 CFR 100, Reactor Site Criteria.
 4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
 5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.
 6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."
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B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling System (CCS)

BASES

BACKGROUND

The CCS provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CCS also provides this function for various non-essential components, as well as the spent fuel storage pool. The CCS serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Essential Raw Cooling Water (ERCW) System, and thus to the environment.

The CCS is arranged as two independent, full-capacity cooling trains, Train A and Train B. Train A in Unit 2 is served by CCS Hx AB and CCS pump 1A-2A-A. Pump 1B-2B-B, which is actually Train B equipment, is also normally aligned to the Train A header in Unit 2. However, pump 1B-2B-B can be realigned to Train B on loss of Train A.

Train B is served by CCS Hx C. Normally, only CCS pump C-S is aligned to the Train B header since few non-essential, normally-operating loads are assigned to Train B. However, pump 1B-2B-B can be realigned to the Train B header on a loss of the C-S pump.

Each safety related train is powered from a separate bus. An open surge tank in the system provides pump trip protective functions to ensure that sufficient net positive suction head is available. The pump in each train is automatically started on receipt of a safety injection signal, and all non-essential components will be manually isolated.

Additional information on the design and operation of the system, along with a list of the components served, is presented in the FSAR, Section 9.2.2 (Ref. 1). The principal safety related function of the CCS is the removal of decay heat from the reactor via the Residual Heat Removal (RHR) System. This may be during a normal or post accident cooldown and shutdown.

B 3.7 PLANT SYSTEMS

B 3.7.12 Auxiliary Building Gas Treatment System (ABGTS)

BASES

BACKGROUND

The ABGTS filters airborne radioactive particulates from the area of the fuel pool following a fuel handling accident and from the area of active Unit 2 ECCS components and Unit 2 penetration rooms following a loss of coolant accident (LOCA).

The ABGTS consists of two independent and redundant trains. Each train consists of a heater, a prefilter, moisture separator, a high efficiency particulate air (HEPA) filter, two activated charcoal adsorber sections for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case the main HEPA filter bank fails. The downstream HEPA filter is not credited in the analysis. The system initiates filtered ventilation of the Auxiliary Building Secondary Containment Enclosure (ABSCE) exhaust air following receipt of a Phase A containment isolation signal or a high radiation signal from the spent fuel pool area.

The ABGTS is a standby system, not used during normal plant operations. During emergency operations, the ABSCE dampers are realigned and ABGTS fans are started to begin filtration. Air is exhausted from the Unit 2 ECCS pump rooms, Unit 2 penetration rooms, and fuel handling area through the filter trains. The prefilters or moisture separators remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The plant design basis requires that when moving irradiated fuel in the Auxiliary Building and/or Containment with the Containment open to the Auxiliary Building ABSCE spaces, a signal from the spent fuel pool radiation monitors 0-RE-90-102 and -103 will initiate a Containment Ventilation Isolation (CVI) in addition to their normal function. In addition, a signal from the containment purge radiation monitors 1-RE-90-130 and -131 or other CVI signal will initiate that portion of the ABI normally initiated by the spent fuel pool radiation monitors. **Additionally, a Containment Isolation Phase A (SI signal) from the operating unit, high temperature in the Auxiliary Building air intakes, or manual ABI**

(continued)

BASES

BACKGROUND (continued)

will cause a CVI signal in the refueling unit. In the case where the containment of both units is open to the Auxiliary Building spaces, a CVI in one unit will initiate a CVI in the other unit in order to maintain those spaces open to the ABSCE. Therefore, the containment ventilation instrumentation must remain operable when moving irradiated fuel in the Auxiliary Building if the containment air locks, penetrations, equipment hatch, etc. are open to the Auxiliary Building ABSCE spaces. In addition, the ABGTS must remain operable if these containment penetrations are open to the Auxiliary Building during movement of irradiated fuel inside containment.

The ABGTS is discussed in the FSAR, Sections 6.5.1, 9.4.2, 15.0, and 6.2.3 (Refs. 1, 2, 3, and 4, respectively).

APPLICABLE SAFETY ANALYSES

The ABGTS design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident. The analysis of the fuel handling accident, given in Reference 3, assumes that all fuel rods in an assembly are damaged. The analysis of the LOCA assumes that radioactive materials leaked from the Emergency Core Cooling System (ECCS) are filtered and adsorbed by the ABGTS. The DBA analysis of the fuel handling accident assumes that only one train of the ABGTS is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the one remaining train of this filtration system. The amount of fission products available for release from the ABSCE is determined for a fuel handling accident and for a LOCA. The assumptions and the analysis for a fuel handling accident follow the guidance provided in Regulatory Guide 1.25 (Ref. 5) and NUREG/CR-5009 (Ref. 10). The assumptions and analysis for a LOCA follow the guidance provided in Regulatory Guide 1.4 (Ref. 6).

The ABGTS satisfies Criterion 3 of the NRC Policy Statement.

When moving irradiated fuel inside containment or in the Auxiliary Building with containment air locks or penetrations open to the Auxiliary Building ABSCE spaces, or when moving fuel in the Auxiliary Building with the containment equipment hatch open, the provisions to initiate a CVI from the spent fuel pool radiation monitors and to initiate an ABI (i.e., the portion of an ABI normally initiated by the spent fuel pool radiation monitors) from a CVI, including a CVI initiated by the containment purge monitors, in the event of a fuel handling accident (FHA) must be in place and functioning. **Additionally, a Containment Isolation Phase A (SI signal) from the operating unit, high**

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

temperature in the Auxiliary Building air intakes, or manual ABI will cause a CVI signal in the refueling unit. The containment equipment hatch cannot be open when moving irradiated fuel inside containment in accordance with Technical Specification 3.9.4.

The ABGTS is required to be operable during movement of irradiated fuel in the Auxiliary Building during any mode and during movement of irradiated fuel in the Reactor Building when the Reactor Building is established as part of the ABSCE boundary (see TS 3.3.8, 3.7.12, & 3.9.4). When moving irradiated fuel inside containment, at least one train of the containment purge system must be operating or the containment must be isolated. When moving irradiated fuel in the Auxiliary Building during times when the containment is open to the Auxiliary Building ABSCE spaces, containment purge can be operated, but operation of the system is not required. However, whether the containment purge system is operated or not in this configuration, all containment ventilation isolation valves and associated instrumentation must remain operable. This requirement is necessary to ensure a CVI can be accomplished from the spent fuel pool radiation monitors in the event of a FHA in the Auxiliary Building. **Additionally, a Containment Isolation Phase A (SI signal) from the operating unit, high temperature in the Auxiliary Building air intakes, or manual ABI will cause a CVI signal in the refueling unit. In the case where the containment of both units is open to the Auxiliary Building spaces, a CVI in one unit will initiate a CVI in the other unit in order to maintain those spaces open to the ABSCE.**

LCO

Two independent and redundant trains of the ABGTS are required to be OPERABLE to ensure that at least one train is available, assuming a single failure that disables the other train, coincident with a loss of offsite power. Total system failure could result in the atmospheric release from the ABSCE exceeding the 10 CFR 100 (Ref. 7) limits in the event of a fuel handling accident or LOCA.

The ABGTS is considered OPERABLE when the individual components necessary to control exposure in the fuel handling building are OPERABLE in both trains. An ABGTS train is considered OPERABLE when its associated:

- a. Fan is OPERABLE;
- b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration function; and

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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 DC Sources - Operating

BASES

BACKGROUND

The station DC electrical power system provides the AC emergency power system with control power. It also provides both motive and control power to selected safety related equipment and preferred AC vital bus power (via inverters). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. The DC electrical power system also conforms to the recommendations of Regulatory Guide 1.6 (Ref. 2) and IEEE-308 (Ref. 3).

125 V Vital DC Electrical Power Subsystem

The vital 125 VDC electrical power system is a Class 1E system whose safety function is to provide control power for engineered safety features equipment, emergency lighting, vital inverters, and other safety related DC powered equipment for the entire unit. The system capacity is sufficient to supply these loads and any connected non-safety loads during normal operation and to permit safe shutdown and isolation of the reactor for the "loss of all AC power" condition. The system is designed to perform its safety function subject to a single failure.

The 125V DC vital power system is composed of the four redundant channels (Channels I and III are associated with Train A and Channels II and IV are associated with Train B) and consists of four lead-acid-calcium batteries, ~~six~~**eight** battery chargers (including two **pairs of** spare chargers), four distribution boards, battery racks, and the required cabling, instrumentation and protective features. Each channel is electrically and physically independent from the equipment of all other channels so that a single failure in one channel will not cause a failure in another channel. Each channel consists of a battery charger which supplies normal DC power, a battery for emergency DC power, and a battery board which facilitates load grouping and provides circuit protection. These four channels are used to provide emergency power to the 120V AC vital power system which furnishes control power to the reactor protection system. No automatic connections are used between the four redundant channels.

(continued)

BASES

BACKGROUND
(continued)

The Primary containment exhaust is monitored by a radiation detector which provides automatic containment purge ventilation system isolation upon detecting the setpoint radioactivity in the exhaust air stream. The containment purge ventilation isolation valves will be automatically closed upon the actuation of a Containment Vent Isolation (CVI) signal whenever the primary containment is being purged during normal operation or upon manual actuation from the Main Control Room (Ref. 2). Requirements for Containment Vent Isolation Instrumentation are covered by LCO 3.3.6.

APPLICABLE
SAFETY
ANALYSES

The Reactor Building Purge Ventilation System air cleanup units ensure that the release of radioactivity to the environment is limited by cleaning up containment exhaust during a fuel handling accident before the containment purge exhaust valves are isolated. Reactor Building Purge Ventilation System filter efficiency is one of the inputs for the analysis of the environmental consequences of a fuel handling accident. Containment isolation can only result in smaller releases of radioactivity to the environment (Ref. 1). The Containment Vent Isolation System ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the containment (Ref. 2). Containment Vent Isolation Instrumentation is addressed by LCO 3.3.6.

The Reactor Building Purge Air Cleanup Units satisfy Criterion 3 of the NRC Policy Statement.

In addition, during movement of irradiated fuel in the Auxiliary Building when containment is open to the Auxiliary Building spaces, a high radiation signal from the spent fuel pool accident radiation monitors , **a Containment Isolation Phase A (SI signal) from the operating unit, high temperature in the Auxiliary Building air intakes, or manual ABI** will initiate a CVI. **In the case where the containment of both units is open to the Auxiliary Building spaces, a CVI in one unit will initiate a CVI in the other unit in order to maintain those spaces open to the ABSCE.**

LCO

The safety function of the Reactor Building Purge Air Cleanup Unit is related to the initial control of offsite radiation exposures resulting from a fuel handling accident inside containment. During a fuel handling accident inside containment, the Reactor Building Purge Air Cleanup Unit provides a filtered path for cleaning up any air leaving the containment until the containment ventilation is isolated.

(continued)

BASES (continued)

LCO
(continued)

The plant design basis requires that when moving irradiated fuel in the Auxiliary Building and/or Containment with the Containment open to the Auxiliary Building ABSCE spaces, a signal from the spent fuel radiation monitors 0-RE-90-102 and -103 will initiate a CVI in addition to their normal function. In addition, a signal from the containment purge radiation monitors 2-RE-90-130, and -131 or other CVI signal will initiate that portion of the ABI normally initiated by the spent fuel pool radiation monitors. **Additionally, a Containment Isolation Phase A (SI signal) from the operating unit, high temperature in the Auxiliary Building air intakes, or manual ABI will cause a CVI signal in the refueling unit.** Therefore, the containment ventilation instrumentation must remain operable when moving irradiated fuel in the Auxiliary Building if the containment air locks, penetrations, equipment hatch, etc. are open to the Auxiliary Building ABSCE spaces. In addition, the ABGTS must remain operable if these containment penetrations are open to the Auxiliary Building during movement of irradiated fuel in side containment. **In the case where the containment of both units is open to the Auxiliary Building spaces, a CVI in one unit will initiate a CVI in the other unit in order to maintain those spaces open to the ABSCE.**

APPLICABILITY

An initial assumption in the analysis of a fuel handling accident inside containment is that the accident occurs while irradiated fuel is being handled. Therefore, LCO 3.9.8 is applicable only at this time. See additional discussion in the Applicable Safety Analysis and LCO sections.

ACTIONS

A.1 and A.2

If one Reactor Building Purge Air Cleanup Unit is inoperable, that air cleanup unit must be isolated. This places the system in the required accident configuration, thus allowing refueling to continue after verifying the remaining air cleanup unit is aligned and OPERABLE.

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