TVA

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

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

Subject: Watts Bar Nuclear Plant Unit 2 – Fuel Handling Accident Dose Analysis Final Safety Analysis Report and Technical Specification Revision

References:
1. NRC letter to TVA dated June 19, 2013, "Watts Bar Nuclear Plant, Unit 1 - Issuance of Amendment to Allow Selective Implementation of Alternate Source Term to Analyze the Dose Consequences Associated with Fuel-Handling Accidents (TAC NO. ME8877)" (ADAMS Accession No. ML13141A564)
2. TVA letter to NRC dated August 5, 2011, "Watts Bar Nuclear Plant (WBN) Unit 2 – Final Safety Analysis Report (FSAR) – Chapter 15.5 Design Basis Dose Accident Analysis" (ADAMS Accession No. ML11222A022)
3. TVA letter to NRC dated September 23, 2011, "Watts Bar Nuclear Plant (WBN) Unit 2 – Final Safety Analysis Report (FSAR) – Chapter 15.5 Fuel Handling Accident (FHA) Dose Analysis" (ADAMS Accession No. ML11269A064)

This letter provides revised Final Safety Analysis Report (FSAR) discussions and Technical Specification (TS) and Technical Specification Bases (TSB) changes associated with the Design Basis Accident (DBA) discussion for the Fuel Handling Accident (FHA) at Watts Bar Nuclear Plant (WBN) Unit 2. The changes to the WBN Unit 2 documents provide consistency with the recently approved amendment issued for WBN Unit 1 (Reference 1).

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WBN Unit 1 submitted a license amendment request to implement the Alternate Source Term (AST) methodology for the FHA. The amendment included TS and TSB changes to remove the requirements for certain safety-related filtration systems to be operable during refueling because no credit was taken for radionuclide removal by those systems in the FHA. By Reference 3, WBN Unit 2 submitted a FHA based on the AST methodology for a dropped fuel assembly in the Auxiliary Building and in the containment when the containment was not isolated. The Nuclear Regulatory Commission (NRC) determined that the WBN Unit 2 FHA analysis was acceptable in NUREG-0847 Supplemental Safety Evaluation Report (SSER) 25, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Unit 2."

The WBN Unit 1 FHA Amendment did not include a specific evaluation for the case when the primary containment is closed and the purge system is in operation because the results from the containment closed case are clearly bounded by the containment open case. The WBN 2 FSAR currently includes a dose analysis for the FHA with the containment closed based on Regulatory Guide 1.25 guidance. This letter provides a revised WBN Unit 2 FHA FSAR Section 15.5.6 discussion consistent with the approved Unit 1 License Amendment Request (LAR). This change removes the discussion of the Regulatory Guide 1.25 analysis for the closed containment case. Changes to the Unit 2 TS and TSB to be consistent with the approved Unit 1 TS and TSB are provided. Changes to WBN Unit 2 FSAR Chapters 6 and 9 that remove the mitigation of an FHA as a design basis for the safety related filtration systems consistent with the WBN Unit 1 Amendment are also provided.

Enclosure 1 provides a discussion of changes to the FHA analysis currently described in the FSAR.

Enclosures 2 through 5 provide red-line markup and final versions of FSAR Sections in Chapters 6, 9, and 15. Enclosures 6 through 9 provide the red-lined markup and final versions of the WBN Unit 2 TS and TSB consistent with Reference 1. Enclosure 10 shows the deletion of Technical Requirements Manual Section 3.9.1, "Decay Time." This requirement has been moved to a new TS Section 3.9.8, "Decay Time."

The FSAR changes will be incorporated in Amendment 111. This is a new regulatory commitment. If you have any questions, please call me at (423) 365-2004.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 12th day of December, 2013.

Respectfully,

Gordon Arent
Director, Watts Bar Licensing
Nuclear Construction

Enclosures:

1. WBN Unit 2 Revised FSAR Section 15.5 Fuel Handling Accident Dose Analysis Results
2. WBN Unit 2 – Revised FSAR Section 15.5. – Red-Lined
3. WBN Unit 2 – Revised FSAR Sections 6.2, 6.5, 9.4. and 15.5 - Final
4. WBN Unit 2 – Revised FSAR Sections 6.2, 6.5, and 9.4 – Red-Lined
5. WBN Unit 2 – Revised FSAR Sections 6.2, 6.5, and 9.4 - Final
6. WBN Unit 2 – Revised Technical Specification Red-Line Markup
7. WBN Unit 2 – Revised Technical Specification - Final
8. WBN Unit 2 – Revised Technical Specification Bases Red-Line Markup
9. WBN Unit 2 – Revised Technical Specification Bases - Final
10. WBN Unit 2 – Revised Technical Requirements Manual Section 3.9.1

cc (Enclosures):

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Enclosure 1

WBN Unit 2 Revised FSAR Section 15.5
Dose Analysis

The Watts Bar Nuclear Plant (WBN) Unit 2 Fuel Handling Accident (FHA) was updated to use the Alternate Source Term (AST) described in Regulatory Guide (RG) 1.183 for an event in the spent fuel pool located in the Auxiliary Building or in the containment when the equipment hatch, or both doors in a personnel air lock, are open. The analysis for a dropped fuel assembly inside containment when the containment air locks and equipment hatch are closed continued to use the methodology of RG-1.25. This change was approved by the NRC as documented in NUREG-0847 Supplement 25, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Unit 2."

Subsequently, WBN Unit 1 submitted a License Amendment Request (LAR) to selectively implement the AST for the FHA. The NRC approved this request June 19, 2013, as Amendment 92. The WBN Unit 1 LAR presented two cases. Case 1 was the FHA at the Spent Fuel Pool. Case 2 was an FHA in containment with the containment open. The discussion of the FHA with the containment isolated was removed from the Updated Final Safety Analysis Report (FSAR) by Amendment 92. The refueling mode Limiting Condition for Operation and associated Surveillance Requirements for the Purge System and the Auxiliary Building Gas Treatment System were removed from the Technical Specifications (TS) because no credit was taken in the analyses for the filtration units. The approval for these changes was included in Amendment 92.

The WBN Unit 2 FSAR, TS and TS Bases are being revised to match those of WBN Unit 1. For WBN Unit 2, the discussion of the RG 1.25 analysis of the containment closed FHA is being removed from the FSAR. A table will be added in WBN Unit 2 FSAR, Section 15.5, providing the results for the containment open case to be consistent with what was done for WBN Unit 1. The WBN Unit 2 FSAR tables will not include the values for fuel with Tritium Producing Burnable Adsorber Rods (TPBARS), because they are not part of the WBN Unit 2 design basis. The WBN Unit 1 analyses were performed using the same meteorological data (X/Q) and wind speeds that form the basis for the WBN Unit 2 FSAR, Section 15.5 dose analyses. Thus, the results are consistent with the WBN Unit 2 AST approval documented in SSER 25.

The evaluation for the FHA at the spent fuel pool is a bounding analysis for a dropped assembly in containment when the containment is open or closed. The release point for the containment purge system is the WBN Unit 2 shield building stack. The X/Qs are lower for this release point than the normal Auxiliary Building exhaust. In addition, any release from the shield building stack would go through the purge system High Efficiency Particulate Air (HEPA) and charcoal filter assemblies prior to release. Currently, when the purge lines isolate on high radiation, the Auxiliary Building also isolates and the Auxiliary Building Gas Treatment System (ABGTS) is actuated. The release point for ABGTS is the shield building stacks, and the releases are filtered through HEPA and charcoal assemblies. Thus, the AST analysis for the FHA in the Auxiliary Building that considers no filtration and no Auxiliary Building isolation is conservative and acceptable as the basis for the containment open evaluation. When the purge valves close at approximately 12.7 seconds with the containment closed, any further release of radioactivity would be terminated. If the purge valves did not close and the releases continued from the shield building stack, the results would be bounded by the FHA in the Auxiliary Building.

WBN Unit 2 Revised FSAR Section 15.5
Dose Analysis

This change is determined to be acceptable because:

1) If the containment closed case were evaluated using the AST, the results would be bounded by the cases currently presented in the FSAR, and
2) This will bring the WBN Unit 2 FSAR discussion of this event into agreement with the recently approved WBN Unit 1 LAR.

As part of this selective implementation of AST, the following changes are assumed in the analysis:

- The total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11.
- The gap activity is revised to be consistent with RG-1.183.
- The decontamination factors were changed to be consistent with RG-1.183.
- New onsite (control room) and offsite atmospheric dispersion factors (X/Q) are used.
- The time to isolate the control room is increased from 20.6 seconds to 40 seconds.
- No Auxiliary Building isolation is assumed.
- No filtration of the release from the Containment or the spent fuel pool to the environment by the containment purge filters or the ABGTS is assumed.

The WBN design includes a secondary containment that is designed to limit any potential radioactive leakage to the outside environment following a Design Basis Accident (DBA). The secondary containment consists of the concrete shield building that encloses the steel primary containment and the portion of the Auxiliary Building called the Auxiliary Building Secondary Containment Enclosure. The Secondary Containment is described in FSAR Section 6.2.3. The secondary containment structures work in conjunction with safety related ventilation systems and the appropriate isolation of normal ventilation systems to perform its safety functions. In addition to the descriptions in FSAR section 6.2.3, the air clean-up and filtration systems are described in FSAR Section 6.5. The DBA Loss of Coolant Accident (LOCA) is the accident that generally dictates the basis for the design of the Secondary Containment.

In addition to the LOCA, the FHA analyses performed based on RG-1.25 that were part of the original licensing basis for WBN resulted in safety functions being defined for the Secondary Containment. If an FHA occurred either in the Auxiliary Building or in primary containment, the ABGTS was required to start and the Auxiliary Building normal ventilation system isolated. A discussion of the Auxiliary Building Ventilation System is provided in FSAR Sections 9.4.2 and 9.4.3. If the FHA occurred in the primary containment, credit was taken for the Reactor Building Purge Filtration system in the FSAR Chapter 15 dose analysis. A general description of the Reactor Building Purge System is provided in FSAR Section 9.4.6. The FHA based on the AST does not credit containment or Auxiliary Building isolation. No credit is taken for the high efficiency particulate and charcoal filter systems associated with the ABGTS and the purge system. The approved WBN Unit 1 amendment removed the TS requirements associated with refuel mode operation for these systems. The WBN Unit 2 FSAR revisions associated with the approved WBN Unit 1 TS changes are provided in this submittal.

FSAR Section 6.2.4.3 on containment isolation discusses administrative controls to manually close the ice blowing penetrations in the event of an FHA in containment. The updated FHA based on the AST does not require containment isolation to meet dose criteria. This section has been revised accordingly.

Enclosure 1

WBN Unit 2 Revised FSAR Section 15.5
Dose Analysis

The following summarizes the specific changes to the FSAR and TS.

1. Update FSAR Section 15.5.6 to remove the RG-1.25 based analysis of a FHA in containment with the containment isolated except for the purge system.
2. Revise FSAR Sections 6.2.3.1.1 and 6.2.3.1.3 to remove the FHA as a design basis for the Secondary Containment and ABGTS.
3. Revise FSAR Section 6.2.4.3 to remove the administrative requirement to manually isolate the ice blowing penetrations for an FHA.
4. Revise FSAR Sections 6.5.1.1.3 and 6.5.1.2.3 to remove the discussion of the Reactor Building Purge System design basis for the FHA.
5. Revise FSAR Section 9.4.2.3 to remove the requirement for the Auxiliary Building normal ventilation system to isolate for an FHA.
6. Revise FSAR Section 9.4.6 on the Reactor Building Purge System to remove the FHA as a design basis for the system.
7. Add new WBN Unit 2 TS 3.9.10 and associated Bases Section to restrict movement of irradiated fuel assemblies until 100 hours after the reactor core has become sub-critical. TS 3.9.10 ensures that the irradiated fuel meets the minimum decay time established in the radiological analysis of the FHA.
8. Modify WBN Unit 2 TS 3.3.6, "Containment Vent Isolation Instrumentation"; TS 3.3.8, “Auxiliary Building Gas Treatment System (ABGTS) Actuation Instrumentation"; and TS 3.7.12, "Auxiliary Building Gas Treatment System (ABGTS)", to eliminate the requirements associated with movement of irradiated fuel assemblies in the containment or the fuel handling area. Modify associated TS Bases.
9. Eliminate TS 3.9.4, "Containment Penetrations", and TS 3.9.8, "Reactor Building Purge Air Cleanup Units".
10. Modify WBN Unit 2 TS 5.7.2.14 to remove RG-1.52 testing of the Reactor Building Purge HEPA and Charcoal Filter Units.
11. Modify WBN Unit 2 TS 5.7.2.20 to incorporate the Control Room dose limit defined in 10 CFR 50.67(b)(2)(iii).
12. Modify TS Bases 3.6.1, “Containment Penetrations”; 3.6.2, “Containment Air Locks”; and 3.6.3, “Containment Isolation Valves”, to eliminate isolation requirements during fuel movement inside containment. Delete TS Bases 3.9.4.
13. Modify TS Bases 3.7.13, “Spent Fuel Pool Level,” and 3.9.7, “Reactor Cavity Water Level,” to update references associated with AST.
14. Remove the decay time restriction on post shutdown irradiated fuel movement from Section 3.9.1 of the Technical Requirements Manual. This restriction has been added to the TS as described in Item 7 above.

Enclosures 2 and 3 provide a red-lined mark-up and a final version of FSAR Section 15.5.6 on the FHA. Enclosures 4 and 5 provide a red-lined mark-up and a final version of FSAR Sections 6.2.3, 6.2.4, 6.5, and 9.4. Enclosures 6 and 7 provide the red-lined mark-up and final version of the WBN Unit 2 TS sections as enumerated in the numbered list immediately above. Enclosures 8 and 9 provide the red-lined mark-up and final version of the TS Bases sections associated with TS listed in items 7 through 9 of the list provided above. Enclosure 10 deletes Technical Requirements Manual Section, 3.9.1.

Enclosure 2

WBN Unit 2 Red-line Markup of FSAR Section 15.5.6

Enclosure 2

WBN Unit 2 Red-line Markup of FSAR Section 15.5.6

15.5.6 Environmental Consequences of a Postulated Fuel Handling Accident

The analysis of the fuel handling accident considers two ~~three~~ cases. T~~he first case is for a Fuel Handling Accident inside containment with the containment closed and the Reactor Building Purge System operating. This analysis is discussed in Section 15.5.6.1 and is based on Regulatory Guide 1.25[11] and NUREG 5009[24].~~ The first ~~second~~ case is for an accident in the spent fuel pool area located in the Auxiliary Building. This case is ~~discussed in Section 15.5.6.2 and is~~ evaluated using the Alternate Source Term (AST) based on Regulatory Guide 1.183[18], "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." The second ~~third~~ case considered is an open containment case for an accident inside containment where there is open communication between the containment and the Auxiliary Building. This evaluation is also based on the AST ~~discussed in Section 15.5.6.2~~ and ~~is based on~~ Regulatory Guide 1.183. The parameters used for this analysis are listed in Table 15.5-20.a.

~~15.5.6.1 Fuel Handling Accident Based on Regulatory Guide 1.25~~

~~The parameters used for this analysis are listed in Table 15.5-20.~~

~~The bases for the Regulatory Guide 1.25 evaluation are:~~

~~(1) In the Regulatory Guide 1.25 analysis, the accident occurs 100 hours after plant shutdown. Radioactive decay of the fission product inventory during the interval between shutdown and placement of the first spent fuel assembly into the spent fuel pit is taken into account.~~

~~(2) In the Regulatory Guide 1.25 analysis damage is assumed for all rods in one assembly.~~

~~(3) The assembly damaged is the highest powered assembly in the core region to be discharged. The values for individual fission product inventories in the damaged assembly are calculated assuming full-power operation at the end of core life immediately preceding shutdown. Nuclear core characteristics used in the analysis are given in Table 15.5-21. A radial peaking factor of 1.65 is used.~~

~~(4) For the Regulatory Guide 1.25 analysis all of the gap activity in the damaged rods is released to the spent fuel pool and consists of 10% of the total noble gases and radioactive iodine inventory in the rods at the time of the accident with the following gap percentage exceptions which are based on NUREG/CR 5009 [24] as appropriate: 14% of the Kr-85, 5% of the Xe-133, 2% of the Xe-135, and 12% of the I-131.~~

~~(5) Noble gases released in the containment are released through the Shield Building vent to the environment.~~

~~(6) In the Regulatory Guide 1.25 analysis the iodine gap inventory is composed of inorganic species (99.75%) and organic species (0.25%).~~

~~(7) A filter efficiency of 90% for inorganic iodine and 30% for organic iodine for the purge air exhaust filters is used since no relative humidity control is provided.~~

~~(8) No credit is taken for natural decay after the activity has been released to the atmosphere.~~

~~(9) The short-term (i.e., 0-2 hour) atmospheric dilution factors at the exclusion area boundary and low population zone given in Table 15A-2 are used. The thyroid dose utilizes ICRP-30 [25] iodine dose conversion factors. Doses are based on the dose models presented in Appendix 15A.~~

~~15.5.6.2 Fuel Handling Accident Based on Regulatory Guide 1.183~~

~~The analysis of a postulated fuel handling accident in the Auxiliary Building refueling Area or inside the primary containment is based on Regulatory Guide 1.183. i.e., Alternate Source Terms (AST).~~

The bases for evaluation are:

(1) ~~In the Regulatory Guide 1.183 analysis, t~~ The accident occurs 100 hours after plant shutdown. Radioactive decay of the fission product inventory during the interval between shutdown and placement of the first spent fuel assembly into the spent fuel pit is taken into account.

(2) ~~In the Regulatory Guide 1.183 analysis, d~~ Damage was assumed for all rods in one assembly.

(3) The assembly damaged is the highest powered assembly in the core region to be discharged. The values for individual fission product inventories in the damaged assembly are calculated assuming full-power operation at the end of core life immediately preceding shutdown. Nuclear core characteristics used in the analysis are given in Table 15.5-21. A radial peaking factor of 1.65 is used.

(4) ~~The Regulatory Guide 1.183 analysis assumes a~~ All of the gap activity in the damaged rods is released to the spent fuel pool and consists of 8% I-131, 10% Kr-85, and 5% of other noble gases and other halogens.

(5) Noble gases released to the Auxiliary Building spent fuel pool are released through the Auxiliary Building vent to the environment.

(6) ~~In the Regulatory Guide 1.183 analysis, t~~The iodine gap inventory is composed of inorganic species (99.85%) and organic species (0.15%).

(7) ~~In the Regulatory Guide 1.183 analysis, t~~ The overall inorganic and organic iodine spent fuel pool decontamination factor is 200.

(8) ~~In the Regulatory Guide 1.183 analysis, a~~ All iodine escaping from the Auxiliary Building spent fuel pool is exhausted unfiltered through the Auxiliary Building vent.

(9) The release path for the containment scenario is changed to include 12.7 seconds of unfiltered release through the Shield Building vent, with the remainder of the unfiltered release through the Auxiliary Building vent.

(10) No credit is taken for the ABGTS or Containment Purge System Filters in the analysis.

(11) No credit is taken for natural decay either due to holdup in the Auxiliary Building or after the activity has been released to the atmosphere.

(12) The short-term (i.e., 0-2 hour) atmospheric dilution factors at the exclusion area boundary and low population zone given in Table 15A-2 are used. The thyroid dose utilizes ICRP-30 [25] iodine dose conversion factors. Doses are based on the dose models presented in Appendix 15A.

~~15.5.6.3~~ Fuel Handling Accident Results

~~The radiation dose results of the Regulatory Guide 1.25 with the containment closed fuel handling accident (FHA) are given in Table 15.5-23. For a FHA inside containment, no allowance has been made for possible holdup or mixing in the primary containment or isolation of the primary containment as a result of a high radiation signal from the monitors in the ventilation systems for the case where containment penetrations are closed to the Auxiliary Building. However, the containment purge filters are credited. Dose equations in TID-14844 [23] were used to determine the dose. Dose conversion factors in ICRP-30 [25] were used to determine thyroid doses in place of those found in TID-14844.~~

~~The ventilation function of the reactor building purge ventilating system (RBPVS) is not a safety-related function. However, the filtration units and associated exhaust ductwork do provide a safety-related filtration path following a fuel-handling accident prior to automatic closure of the associated isolation valves. The RBPVS contains air cleanup units with prefilters, HEPA filters, and 2-inch-thick charcoal adsorbers. This system is similar to the auxiliary building gas treatment system except that the latter is equipped with 4-inch-thick charcoal adsorbers. Anytime fuel handling operations are being carried on inside the primary containment, either the containment is isolated or the reactor building purge filtration system is operational. The assumptions listed above are, therefore, applicable to a fuel handling accident inside primary containment.~~

~~The thyroid, gamma, and beta doses for FHAs for the closed containment are given in Table 15.5-23 for the exclusion area boundary and low population zone. These doses are less than 25% of the 10 CFR 100.11 limits of 300 rem to the thyroid, and 25 rem gamma to the whole body. These doses are calculated using the computer code FENCDOSE [16].~~

~~The whole body, beta, and thyroid doses to control room personnel from the radiation sources discussed above are presented in Table 15.5-23. The doses are calculated by the COROD computer code [17]. Parameters for the control room analysis are found in Table 15.5-14. The dose to whole body is below the 10 CFR 50 Appendix A, GDC 19 limit of 5 rem for control room personnel and the thyroid dose is below the limit of 30 rem.~~

Enclosure 2

WBN Unit 2 Red-line Markup of FSAR Section 15.5.6

The radiation dose results of the Regulatory Guide 1.183 fuel handling accident (FHA) are given in Table 15.5-23. ~~Alternate source term (AST)~~ The AST described in RG 1.183 was selectively used to evaluate the FHA due to an event in the spent fuel pool located in the Auxiliary Building or in the containment when the equipment hatch or both doors in a personnel air lock are open. As part of this selective implementation of AST, the following assumptions are used in the analysis:

- The total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11.
- The gap activity is revised to be consistent with that required by RG 1.183.
- The decontamination factors were changed to be consistent with those required by RG. 1.183.
- No Auxiliary Building isolation is assumed.
- No filtration of the release from the Containment or the spent fuel pool to the environment by the Containment Purge filters or the ABGTS is assumed.

The evaluation for the FHA at the spent fuel pool is a bounding analysis for a dropped assembly in containment when the containment is open or closed. The release point for the containment purge system is the Unit 2 shield building stack. The X/Qs are lower for this release point than for the normal auxiliary building exhaust. ~~In addition, any release from the shield building stack would go through the purge system HEPA and Charcoal filter assemblies prior to release. Currently, when the purge lines isolate on high radiation, the auxiliary building also isolates and ABGTS is actuated. The release point for ABGTS is the shield building stacks and the releases are filtered through HEPA and Charcoal assemblies. Thus~~ The AST analysis for the FHA in the Auxiliary Building that considers no filtration is conservative and acceptable as the basis for the containment evaluation.

The ~~thyroid, gamma, and beta doses~~ TEDE for FHAs in the Auxiliary and the open containment are given in Table 15.5-23 for the exclusion area boundary and low population zone. These doses are ~~less than 25% of the 10 CFR 100.11 limits of 300 rem to the thyroid, and 25 rem gamma to the whole body and~~ less than the 10 CFR 50.67 limit of 6.3 ~~25~~ rem TEDE. These doses are calculated using the computer code FENCDOSE [16].

The TEDE ~~whole body, beta, and thyroid doses~~ to control room personnel from the radiation sources discussed above are presented in Table 15.5-23. The doses are calculated by the COROD computer code [17]. Parameters for the control room analysis are found in Table 15.5-14. The dose to ~~whole body is below the 10 CFR 50 Appendix A, GDC 19 limit of 5 rem fo~~r control room personnel~~, and the thyroid dose~~ is below ~~the limit of 30 rem and~~ the 10CFR 50.67 limit of 5 rem TEDE.

~~Table 15.5-20~~
~~Parameters Used In Fuel Handling Accident Analysis~~
~~Regulatory Guide 1.25 Analysis~~

~~Time between plant shutdown and accident~~	~~100 hours~~
~~Damage to fuel assembly~~	~~All rods ruptured~~
~~Fuel assembly activity~~	~~Highest powered fuel assembly in core region discharged~~
~~Activity release to spent fuel pool~~	~~Gap activity in ruptured rods(1)~~
~~Radial peaking factor~~	~~1.65~~
~~Form of iodine activity released~~	
~~elemental iodine~~	~~99.75%~~
~~methyl iodine~~	~~0.25%~~
~~Filter efficiencies~~	~~RBPVS (2)~~
~~elemental iodine~~	~~90%~~
~~methyl iodine~~	~~30%~~
~~Amount of mixing of activity in Auxiliary Building~~	~~None~~

~~Meteorology See Table 15.5-14 and Table 15A-2~~

~~(1) 10% of the total radioactive iodine except for 12% of I-131 and 10% of total noble gases, except for 14% for Kr-85, 5% for Xe-133 and 2% for Xe-135 in the damaged rods at the time of the accident.~~

~~(2) Reactor Building Purge Ventilation System~~

Table 15.5-20.a
Parameters Used In Fuel Handling Accident Analysis
Regulatory Guide 1.183 Analysis

Time between plant shutdown and accident	100 hours
Damage to fuel assembly	All rods ruptured
Fuel assembly activity	Highest powered fuel assembly in core region discharged
Activity release to spent fuel pool	Gap activity in ruptured rods(1)
Radial peaking factor	1.65
Form of iodine activity released to spent fuel pool	
elemental iodine methyl iodine	99.85%(AST) 0.15%(AST)
Decontamination factor in spent fuel pool	AST Overall=200
Filter efficiencies	No credit taken
Amount of mixing of activity in Auxiliary Building	None
Meteorology See Table 15.5-14 and Table15A-2	

(1) 8% I-131, 10% Kr-85, and 5% other gasses and other halogens.

Enclosure 2

WBN Unit 2 Red-line Markup of FSAR Section 15.5.6

Table 15.5-23

Doses From A Fuel Handling Accident (FHA) (rem)

Doses from Fuel Handling Accident Regulatory Guide 1.183 Analyses

FHA in Auxiliary Building (rem) ~~or In Containment – Containment Open (rem)~~

	2 HR EAB	30 DAY LPZ	CONTROL ROOM
~~Gamma~~	~~3.994E-01 4.29E-01~~	~~9.278E-02 1.20E-01~~	~~4.935E-01 5.86E-01~~
~~Beta~~	~~1.177E+00 1.19E+00~~	~~2.734E-01 3.33E-01~~	~~4.068E+00 4.68E+00~~
~~Thyroid - ICRP-30~~	~~1.577E+00 5.51E+01~~	~~3.663E-01 1.54E+01~~	~~1.540E+00 1.32E+01~~
TEDE	2.38E+00	6.66E-01	1.02E-00

Doses from Fuel Handling Accident Regulatory Guide 1.183 Analyses

FHA in Containment – Containment Open (rem)

	2 HR EAB	30 DAY LPZ	CONTROL ROOM
~~Gamma~~	~~3.994E-01 4.29E-01~~	~~9.278E-02 1.20E-01~~	~~4.935E-01 5.86E-01~~
~~Beta~~	~~1.177E+00 1.19E+00~~	~~2.734E-01 3.33E-01~~	~~4.068E+00 4.68E+00~~
~~Thyroid - ICRP-30~~	~~1.577E+00 5.51E+01~~	~~3.663E-01 1.54E+01~~	~~1.540E+00 1.32E+01~~
TEDE	2.38E+00	6.66E-01	1.01E-00

~~**Doses from Fuel Handling Accident Regulatory Guide 1.25 Analyses**~~

~~**FHA in Reactor Building, Containment Closed (rem)**~~

	~~2 HR EAB~~	~~30 DAY LPZ~~	~~CONTROL ROOM~~
~~Gamma~~	~~4.102E-01 4.31E-01~~	~~9.529E-02 1.21E-01~~	~~2.677E-01 2.72E-01~~
~~Beta~~	~~1.182E+00 1.24E+00~~	~~2.746E-01 3.48E-01~~	~~2.207E+00 2.25E+00~~
~~Thyroid - ICRP-30~~	~~39.42E+00 4.15E+01~~	~~9.158E+00 1.16E+01~~	~~5.209E+00 6.81E+00~~

Enclosure 3

WBN Unit 2 – Revised FSAR Section 15.5
Final

15.5.6 Environmental Consequences of a Postulated Fuel Handling Accident

The analysis of the fuel handling accident considers two cases. The first case is for an accident in the spent fuel pool area located in the Auxiliary Building. This case is evaluated using the Alternate Source Term based on Regulatory Guide 1.183[18], "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." The second case considered is an open containment case for an accident inside containment where there is open communication between the containment and the Auxiliary Building. This evaluation is also based on the AST and Regulatory Guide 1.183. The parameters used for this analysis are listed in Table 15.5-20.a.

The bases for evaluation are:

(1) The accident occurs 100 hours after plant shutdown. Radioactive decay of the fission product inventory during the interval between shutdown and placement of the first spent fuel assembly into the spent fuel pit is taken into account.

(2) Damage was assumed for all rods in one assembly.

(3) The assembly damaged is the highest powered assembly in the core region to be discharged. The values for individual fission product inventories in the damaged assembly are calculated assuming full-power operation at the end of core life immediately preceding shutdown. Nuclear core characteristics used in the analysis are given in Table 15.5-21. A radial peaking factor of 1.65 is used.

(4) All of the gap activity in the damaged rods is released to the spent fuel pool and consists of 8% I-131, 10% Kr-85, and 5% of other noble gases and other halogens.

(5) Noble gases released to the Auxiliary Building spent fuel pool are released through the Auxiliary Building vent to the environment.

(6) The iodine gap inventory is composed of inorganic species (99.85%) and organic species (0.15%).

(7) The overall inorganic and organic iodine spent fuel pool decontamination factor is 200.

(8) All iodine escaping from the Auxiliary Building spent fuel pool is exhausted unfiltered through the Auxiliary Building vent.

(9) The release path for the containment scenario is changed to include 12.7 seconds of unfiltered release through the Shield Building vent, with the remainder of the unfiltered release through the Auxiliary Building vent.

(10) No credit is taken for the ABGTS or Containment Purge System Filters in the analysis.

(11) No credit is taken for natural decay either due to holdup in the Auxiliary Building or after the activity has been released to the atmosphere.

(12) The short-term (i.e., 0-2 hour) atmospheric dilution factors at the exclusion area boundary and low population zone given in Table 15A-2 are used. The thyroid dose utilizes ICRP-30 [25] iodine dose conversion factors. Doses are based on the dose models presented in Appendix 15A.

15.5.6.3 Fuel Handling Accident Results

The radiation dose results of the Regulatory Guide 1.183 fuel handling accident (FHA) are given in Table 15.5-23. The AST described in RG 1.183 was selectively used to evaluate the FHA due to an event in the spent fuel pool located in the Auxiliary Building or in the containment when the equipment hatch or both doors in a personnel air lock are open. As part of this selective implementation of AST, the following assumptions are used in the analysis:

- The total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11.
- The gap activity is revised to be consistent with that required by RG 1.183.
- The decontamination factors were changed to be consistent with those required by RG. 1.183.
- No Auxiliary Building isolation is assumed.
- No filtration of the release from the Containment or the spent fuel pool to the environment by the Containment Purge filters or the ABGTS is assumed.

The evaluation for the FHA at the spent fuel pool is a bounding analysis for a dropped assembly in containment when the containment is open or closed. The release point for the containment purge system is the Unit 2 shield building stack. The X/Qs are lower for this release point than for the normal auxiliary building exhaust. The AST analysis for the FHA in the Auxiliary Building is conservative and acceptable as the basis for the containment evaluation.

The thyroid, gamma, and beta doses for FHAs in the Auxiliary and the open containment are given in Table 15.5-23 for the exclusion area boundary and low population zone. These doses are less than the 10 CFR 50.67 limit of 6.3 rem TEDE. These doses are calculated using the computer code FENCDOSE [16].

The whole body, beta, and thyroid doses to control room personnel from the radiation sources discussed above are presented in Table 15.5-23. The doses are calculated by the COROD computer code [17]. Parameters for the control room analysis are found in Table 15.5-14. The dose to control room personnel is below the 10CFR 50.67 limit of 5 rem TEDE.

Table 15.5-20.a
Parameters Used In Fuel Handling Accident Analysis
Regulatory Guide 1.183 Analysis

Time between plant shutdown and accident	100 hours
Damage to fuel assembly	All rods ruptured
Fuel assembly activity	Highest powered fuel assembly in core region discharged
Activity release to spent fuel pool	Gap activity in ruptured rods(1)
Radial peaking factor	1.65
Form of iodine activity released to spent fuel pool	
elemental iodine methyl iodine	99.85%(AST) 0.15%(AST)
Decontamination factor in spent fuel pool	AST Overall=200
Filter efficiencies	No credit taken
Amount of mixing of activity in Auxiliary Building	None
Meteorology See Table 15.5-14 and Table15A-2	

(2) 8% I-131, 10% Kr-85, and 5% other gasses and other halogens.

Table 15.5-23

Doses From A Fuel Handling Accident (FHA) (rem)

FHA in Auxiliary Building (rem)

	2 HR EAB	30 DAY LPZ	CONTROL ROOM
TEDE	2.38E+00	6.66E-01	1.02E-00

FHA in Containment – Containment Open (rem)

	2 HR EAB	30 DAY LPZ	CONTROL ROOM
TEDE	2.38E+00	6.66E-01	1.01E-00

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6.2.3 Secondary Containment Functional Design

Structures included as part of the secondary containment system are the Shield Building of each reactor unit, the Auxiliary Building, the Condensate Demineralizer Waste Evaporator (CDWE) Building and the essential raw cooling water (ERCW) pipe tunnels adjacent to the Auxiliary Building. Depending on the configuration of the plant, the Primary Containment Building(s) may also be included as a structure which is part of the secondary containment system. This condition exists when the primary containment is open to the Auxiliary Building. The emergency gas treatment system (EGTS) is provided for ventilation control and cleanup of the atmosphere inside the annulus between the Shield Building and the Primary Containment Building. The Reactor Building purge air system is also available for cleaning up the atmosphere inside the Shield Building Annulus. Refer to Section 9.4.6 for further details relating to the purge air system. The Auxiliary Building Gas Treatment System (ABGTS) provides a similar contamination control capability in the Auxiliary Building Secondary Containment Enclosure (ABSCE),which includes all of the areas listed above.

6.2.3.1 Design Bases

6.2.3.1.1 Secondary Containment Enclosures

Design bases for the secondary containment structures were devised to assure that an effective barrier exists for airborne fission products that may leak from the primary containment, or the Auxiliary Building fuel handling area, during a loss-of-coolant accident (LOCA)~~, or a fuel handling accident (FHA)~~. Within the scope of these design bases are requirements that influence the size, structural integrity, and leak tightness of the secondary containment enclosure. Specifically, these include a capability to: (a) maintain an effective barrier for gases and vapors that may leak from the primary containment during all normal and abnormal events; (b) delay the release of any gases and vapors that may leak from the primary containment during accidents; (c) allow gases and vapors that may leak through the primary containment during accidents to flow into the contained air volume within the secondary containment where they are diluted, held up, and purified prior to being released to the environs; (d) bleed to the secondary containment each air-filled containment penetration enclosure which extends beyond the Shield Building and which is formed by automatically actuated isolation valves; (e) maintain an effective barrier for airborne radioactive contaminants, gases, and vapors originating in the ABSCE during normal and abnormal events. Refer to Sections 3.8.1 and 3.8.4 for further details relating to the design of the Shield Building and the Auxiliary Building.

6.2.3.1.3 Auxiliary Building Gas Treatment System (ABGTS)

The design bases for the ABGTS are:

1. To establish and keep an air pressure that is below atmospheric within the portion of the buildings serving as a secondary containment enclosure during accidents.
2. To reduce the concentration of radioactive nuclides in air releases from the secondary containment enclosures to the environs during accidents to levels sufficiently low to keep the site boundary and LPZ dose rates below the 10 CFR 100 guideline values.
3. To minimize the spreading of airborne radioactivity within the Auxiliary Building following an accidental release in the fuel handling and waste packaging areas. ABGTS is not required to mitigate the consequences of a fuel handling accident.
4. To withstand the safe shutdown earthquake.

5. To provide for initial and periodic testing of the system capability to function as designed (See Chapter 14.0 for information on initial testing of systems).

6.2.3.2 System Design

6.2.3.2.1 Secondary Containment Enclosures

(1) Shield Building

[Text not provided as no changes needed]

(2) Auxiliary Building

[Text not provided as no changes needed]

(3) Auxiliary Building Secondary Containment Enclosure (ABSCE)

The Auxiliary Building secondary containment enclosure (ABSCE) is that portion of the Auxiliary Building and CDWE Building (and for certain configurations, the annulus and primary containment, as discussed below) which serves to maintain an effective barrier for airborne radioactive contaminants released in the auxiliary building during abnormal events. Mechanical and electrical penetrations of this enclosure are provided with seals to minimize infiltration.

[Text for next 3 paragraphs no provided as no changes were needed]

During periods when the primary containment and/or annulus of both units are open to the Auxiliary Building, the ABSCE also includes these areas. ~~During fuel handling operations in this configuration, a high radiation signal from spent fuel pool radiation monitors will result in a Containment Ventilation Isolation (CVI) in addition to an Auxiliary Building isolation and ABGTS start. Similarly, a CVI signal, including a CVI signal generated by a high radiation signal from the containment purge air exhaust radiation monitors, will initiate an Auxiliary Building isolation and start of ABGTS. Likewise, a~~ A Containment Isolation Phase A (SI Signal) from the operating unit or high temperature in the Unit 1 or Unit 2 Auxiliary Building air intake, or manual ABI will cause a CVI signal in the refueling unit. These actions will ensure proper operation of the ABSCE. Both doors of the containment vessel personnel airlocks may be open at the same time during refueling activities while the purge air ventilation system is operating. During fuel handling operations in this configuration, a high radiation signal from spent fuel pool radiation monitors will result in a Containment Ventilation Isolation (CVI) in addition to an Auxiliary Building isolation and ABGTS start. Similarly, a CVI signal, including a CVI signal generated by a high radiation signal from the containment purge air exhaust radiation monitors, will initiate an Auxiliary Building isolation and start of ABGTS. These are not required functions for the ABGTS and the Reactor Building Purge System filters or for Purge System isolation as no credit for these features is in mitigating a fuel handling accident. ~~Under special administrative controls, one of the airlock doors at each location will be closed and the purge air ventilation systems will be shutdown and isolated in a timely manner subsequent to a fuel handling accident to ensure ABSCE boundary integrity.~~ In the case where 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.

6.2.3.2.3 Auxiliary Building Gas Treatment System (ABGTS)

The ABGTS is a fully redundant air cleanup network provided to reduce radioactive nuclide releases from the secondary containment enclosure during accidents. It does this by drawing air from the fuel handling and waste packaging areas through ducting normally used for ventilation purposes to air cleanup equipment, and then directing this air to the reactor unit vent. In doing so, this system draws air from all parts of the ABSCE to establish a negative pressure region in which virtually no unprocessed air passes from this secondary containment enclosure to the atmosphere.

During periods when the primary containment and/or annulus of both units are open to the Auxiliary Building, the ABSCE also includes these areas. The ABGTS has been designed to establish a negative pressure in these additional areas for this configuration. During fuel handling operations in this configuration, a high radiation signal from the spent fuel pool radiation monitors will result in a Containment Ventilation Isolation (CVI) in addition to an Auxiliary Building isolation and ABGTS start. Similarly, a CVI signal, including a CVI signal generated by a high radiation signal from the containment purge air exhaust radiation monitors, will initiate an Auxiliary Building isolation and start of ABGTS. Likewise, a Containment Isolation Phase A (SI Signal) from the operating unit or high temperature in the Unit 1 or Unit 2 Auxiliary Building air intake, or manual ABI will cause a CVI signal in the refueling unit. These actions will ensure proper operation of the ABSCE. However, as an added precaution to protect the ABGTS operational boundary, operational action is needed to ensure the closure of the containment purge exhaust isolation valves (system valves not containment isolation valves) which are controlled by hand switches. In the case where 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.

6.2.3.3.3 Auxiliary Building Gas Treatment System (ABGTS)

The ABGTS has the capabilities needed to preserve safety in accidents as severe as a LOCA. This was determined by conducting functional analyses of the system to verify that the system has the proper features for accident mitigation which consist of a failure modes and effects analysis, a review of Regulatory Guide 1.52 sections to assure licensing requirement conformance, and a performance analysis to verify that the system has the desired accident mitigation capabilities. A detailed failure modes and effects analysis is presented in Table 6.2.3-3.

The functional analyses conducted on the ABGTS have shown that:

1. The air intakes for the system are properly located to minimize accident effects. The use of the air intakes provided in the fuel handling and waste disposal areas minimizes the spread of airborne contamination that may be accidentally released at these positions in which the probability of an accidental release, e.g., a fuel handling accident, is more likely. This localization effect is provided without reducing the effectiveness of the system to cope with multiple activity released throughout the ABSCE that may occur during a LOCA. Such coverage is accomplished by utilizing the normal ventilation ducting to draw outside air inleakage from any point along the secondary containment enclosure to the fuel handling and waste disposal areas.

2. Accident indication signals are utilized to bring the ABGTS into operation to assure that the system functions when needed to mitigate accident effects. Accidents in which this system is needed to preserve safety are automatically detected by at least one of the three instrumentation sets used to generate accident signals that result in system startup.

3. System startup reliability is very high. The practice of allowing the automatic startup of both full capacity trains in the system gives greater assurance that one train of equipment functions upon receipt of an accident signal.

4. The method adopted to establish and keep the negative pressure level within this secondary containment enclosure minimizes the time needed to reach the desired pressure level. Initially, the full capacity of the ABGTS fans is utilized for this purpose. After reaching the desired operating level, the system control module allows outside air to enter the air flow network just upstream of the fan at a rate to keep the fans operating at full capacity with the enclosed volume at the desired negative pressure level. In this situation, the amount of air withdrawn from the enclosed volume is equal to the amount of outside air inleakage through the ABSCE. In addition, two vacuum breaker dampers in series are provided to admit outside air in case the modulating dampers fail.

5. The ABSCE is maintained at a slightly negative pressure to reduce the amount of unprocessed air escaping from this secondary containment enclosure to the atmosphere to insignificant quantities. In addition, this negative pressure level is less than that which is maintained within the annulus; such that, any air leakage between the Auxiliary Building and the Shield Building is from the Auxiliary Building into the Shield Building.

6. The Train A and Train B air cleanup units are sufficiently separated from each other to eliminate the possibility of a single failure destroying the capability to process Auxiliary Building air prior to its release to the atmosphere. Two concrete walls and a distance of more than 80 feet separate the two trains. The use of separate trains of the emergency power system to drive the air cleanup trains gives further assurance of proper equipment separation.

The review of the ABGTS conducted to determine its conformance with Regulatory Guide 1.52 has shown that this system, designed prior to issuance of the guide, is in general agreement with its requirements. Details on compliance with Regulatory Guide 1.52 are given in Table 6.5-2.

The performance analysis conducted to verify that the ABGTS has the required accident mitigation capabilities has shown that the system flow rate is sized properly to handle all expected outside air inleakage at a 1/4-inch water gauge negative pressure differential. This indicates that the nominal flow rate of 9000 cfm is sufficient to assure an adequate margin above the expected ABSCE inleakage (ACU filters are replaced as needed to maintain a minimum flow capability of 9300 cfm under surveillance instructions).

The performance analysis evaluated the capability of the ABGTS to reach and maintain a negative pressure of 1/4-inch water gauge with respect to the outside within the boundaries of the ABSCE. The following was utilized in the analysis:

1. Leakage into the ABSCE is proportional to the square root of the pressure differential.

2. Only one air cleanup unit in the ABGTS operates at the rated capacity.

3. The air cleanup unit fan begins to operate 30 seconds after the initiation of an ABI signal~~, or a high radiation signal (See Section 6.2.3.2.3)~~.

4. The initial static pressure inside the ABSCE is conservatively considered to be atmospheric pressure, although the ABSCE is under a negative pressure during normal operation.

5. The effective pressure head due to wind equals 1/8-inch water gauge.

6. Initial average air temperature inside the ABSCE equals 140°F.

7. Atmospheric temperature and pressure are 70°F and 14.4 psia, respectively.

8. ABSCE isolation dampers/valves close within 30 seconds after receiving an ABI ~~or a high radiation~~ signal~~, except for the fuel handling area exhaust dampers which must close within 9.3 seconds~~.

9. The non-safety-related general ventilation and fuel handling area exhaust fans are designed to shut down automatically following a LOCA. Each fan is provided with a safety related Class 1E primary circuit breaker and a safety related Class

1E shunt trip isolation switch which is tripped by a signal of the opposite train from that for the primary circuit breaker to ensure that power is isolated from the fan.

6.2.4.3 Design Evaluation

The containment isolation systems are designed to present a double barrier to any flow path from the inside to the outside of the containment using the double-barrier approach to meet the single-failure criterion.

(e) The design configuration for penetrations X-79A (ice blowing), and X–79B (negative return) is temporarily modified in operating Modes 5 and 6 and when the reactor is defueled (Mode 7) to support ice blowing activities. The normally closed blind flange on each penetration will be opened and temporary piping will be installed in the penetrations. A 12–inch silicone seal will be installed between the piping segment and the penetration. Manual isolation valves will be connected to the piping on the inboard and outboard side of the penetrations. This configuration is being installed to permit ice blowing operations to occur concurrently with fuel handling activities inside containment. ~~Administrative controls will ensure timely closure of the valves subsequent to a fuel handling accident.~~ The penetrations will be returned to their normal design configuration prior to entry into Mode 4 operations.

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6.5 FISSION PRODUCT REMOVAL AND CONTROL SYSTEMS

6.5.1 Engineered Safety Feature (ESF) Filter Systems

Four Engineered Safety Feature (ESF) air cleanup systems' units are provided for fission product removal in post-accident environments. These are:
(1) The emergency gas treatment system (EGTS) air cleanup units.
(2) The Auxiliary Building gas treatment system (ABGTS) air cleanup units.
(3) The Reactor Building purge system air cleanup units.
(4) The Main Control Room emergency air cleanup units.

6.5.1.1 Design Bases

6.5.1.1.1 Emergency Gas Treatment System Air Cleanup Units

The design bases are:
(1) To provide fission product removal capabilities sufficient to keep radioactivity levels in the Shield Building annulus air released to the environs during a DBA LOCA sufficiently low to assure compliance with 10 CFR 100 guidelines.
(2) These air cleanup units are a part of the EGTS. See Section 6.2.3.1.2 for the design bases for other portions of this system.

6.5.1.1.2 Auxiliary Building Gas Treatment System Air Cleanup Units

The design bases are:
(1) To provide fission product removal capabilities sufficient to keep radioactivity levels in the Auxiliary Building secondary containment enclosure (ABSCE) air released to the environs during a postulated accident sufficiently low to assure compliance with 10 CFR 100 guidelines.
(2) These air cleanup units are a part of the ABGTS. See Section 6.2.3.1.3 for the design basis for other portions of this system.

~~6.5.1.1.3 Reactor Building Purge Air System Air Cleanup Units~~

~~The design bases are:~~
~~*(1)* To provide fission product removal capabilities sufficient to keep radioactivity levels in the primary containment air released to the environs following a fuel handling accident within the containment sufficiently low to assure compliance with 10 CFR 100 guidelines.~~
~~*(2)* These air cleanup units are a part of the Reactor Building purge air system. See Section 9.4.6.1 for the design basis for other portions of this system.~~

~~6.5.1.2.3 Reactor Building Purge System Air Cleanup Units~~

~~See Section 9.4.6.2 for description of the system design of the Reactor Building purge system and the function, operation, and control of the air cleanup units within that system.~~

~~Two 50% capacity air cleanup units, designed to supply a total of 22,949 cfm (two fans together), are provided for each Reactor Building. Both units are located in the same room on Elevation 713 adjacent to the Reactor Building they serve.~~

~~Each air cleanup unit has a stainless steel housing equipped with air treatment components, samples, test fittings, and access facilities for maintenance. The air treatment components within the housing include a prefilter section, a HEPA filter bank, and a carbon filter bank. This equipment is installed in the order listed. Integral to the housing are test fittings properly sized and proportioned to permit orderly and efficient testing of the HEPA filter and carbon adsorber banks.~~

~~The HEPA filters installed in the air cleanup units are 1000 cfm units designed to remove at least 99.97% of the particulates greater than 0.3 microns in diameter, and meet the requirements of military specification MIL-F-51068. The carbon adsorbers installed in the air cleanup units are Type II unit trays, fabricated in accordance with AACC Standard CS-8T requirements. AACC CS-8T has been superseded; and, ANSI/ASME-N509-989 specifies ASME AG-1-1988 to be used. Therefore, all new charcoal Type II cells shall meet AG-1, Section FD, with the exception that the 1991 version of the code be used. Existing Type II cells do not have to be replaced to meet the AG-1 code if being refilled. New replacement charcoal adsorbent (for use in new and refilled Type II cells) shall be procured to meet the ASME AG-1-1991 requirements in lieu of the 1988 version (or later version, provided proper evaluation justifies adequacy), with the exception that laboratory testing of adsorbent be in accordance with ASTM D3803-1989. The total numbers of filters and adsorber unit trays provided in each air cleanup unit are listed in Table 6.5-5.~~

~~Compliance of the design, testing, and maintenance features of the Reactor Building purge system air cleanup units with Regulatory Guide 1.52 is tabulated in Table 6.5-3.~~

~~Table 6.5-3 Regulatory Guide 1.52, Rev.2, Section Applicability For The Reactor Building Purge Ventilation System (Page 1 of 2)~~

~~Reg. Guide Section~~	~~Applicability To This System~~	~~Comment Index~~	~~Reg. Guide Section~~	~~Applicability To This System~~	~~Comment Index~~
~~C.1.a~~	~~yes~~	~~Note 1~~	~~C.3.e~~	~~yes~~	~~Note 14~~
~~C.1.b~~	~~yes~~	~~--~~	~~C.3.f~~	~~yes~~	~~----~~
~~C.1.c~~	~~yes~~	~~--~~	~~C.3.g~~	~~yes~~	~~Note 14~~
~~C.1.d~~	~~yes~~	~~--~~	~~C.3.h~~	~~yes~~	~~----~~
~~C.1.e~~	~~yes~~	~~--~~	~~C.3.i~~	~~yes~~	~~Note 14~~
			~~C.3.j~~	~~yes~~	~~Note 14~~
~~C.2.a~~	~~no~~	~~Notes 3 & 13~~	~~C.3.k~~	~~yes~~	~~Note 11~~
~~C.2.b~~	~~no~~	~~Note 4~~	~~C.3.l~~	~~no~~	~~Note 14~~
~~C.2.c~~	~~yes~~	~~--~~	~~C.3.m~~	~~yes~~	~~----~~
~~C.2.d~~	~~no~~	~~Note 5~~	~~C.3.n~~	~~no~~	~~Notes 9 & 16~~
~~C.2.e~~	~~yes~~	~~--~~	~~C.3.o~~	~~yes~~	~~----~~
~~C.2.f~~	~~yes~~	~~--~~	~~C.3.p~~	~~no~~	~~Notes 12 & 14~~
~~C.2.g~~	~~no~~	~~Note 6~~			
~~C.2.h~~	~~no~~	~~Note 1~~	~~C.4.a~~	~~no~~	~~Note 12~~
~~C.2.i~~	~~yes~~	~~--~~	~~C.4.b~~	~~no~~	~~Note 17~~
~~C.2.j~~	~~no~~	~~Note 8~~	~~C.4.c~~	~~no~~	~~Note 14~~
~~C.2.k~~	~~yes~~	~~--~~	~~C.4.d~~	~~yes~~	~~----~~
~~C.2.l~~	~~no~~	~~Note 9~~	~~C.4.e~~	~~yes~~	~~----~~
~~C.3.a~~	~~no~~	~~Notes 3 & 10~~	~~C.5.a~~	~~yes~~	~~Note 15~~
~~C.3.b~~	~~no~~	~~Notes 3 & 10~~	~~C.5.b~~	~~yes~~	~~Note 15~~
~~C.3.c~~	~~yes~~	~~Note 14~~	~~C.5.c~~	~~yes~~	~~Note 15~~
~~C.3.d~~	~~yes~~	~~Note 14~~	~~C.5.d~~	~~yes~~	~~Note 15~~
			~~C.6.a~~	~~yes~~	~~Notes 14 & 15~~
			~~C.6.b~~	~~yes~~	~~Notes 14 & 18~~

~~Notes~~

1. ~~The postulated design basis accident (DBA) for the reactor building purge ventilation system is a fuel handling accident within the Primary Containment.~~
2. ~~Deleted~~
3. ~~Each air cleanup unit contains a prefilter bank, HEPA filter bank, and carbon adsorber bank in the order listed.~~
4. ~~The short duration of the air cleanup unit operation needed following the postulated DBA identified in Note 1 makes this requirement unnecessary because the probability of such destructive events to equipment already in operation during a short period of time is extremely small.~~
5. ~~No pressure surges of any significance to this air cleanup equipment are envisioned during the postulated DBA identified in Note 1.~~

~~**Notes Continued**~~

6. ~~The system design provides for temperature and pressure differential indication to allow for periodic surveillance of the filter trains. Also, indication of fan operation is provided in the main control room.~~
7. ~~Deleted~~
8. ~~The amount of radioactive material collected by the filter and adsorber banks during the postulated DBA identified in Note 1 is not sufficient to create a radiation hazard when the time comes to replace the filters and adsorbers.~~
9. ~~No safety enhancement is foreseen by the use of low leakage ductwork in this system. In the event of a postulated DBA, all system ductwork carrying radioactive material is at a pressure below atmospheric. Consequently, duct leakage in this part is from the outside into the contaminated air stream.~~
10. ~~No equipment of this kind is utilized in this system because moisture entrainment is considered highly unlikely in the postulated DBA.~~
11. ~~The amount of radioactive material collected during the postulated DBA is too small to raise the adsorber bank temperature near the carbon ignition temperature. However, water sprays are provided in the event of a charcoal fire.~~
12. ~~Compliance with this section is not a licensing requirement.~~
13. ~~The system is sized to maintain acceptable air purity in the containment during normal fuel handling operations. Two system requirements affect the sizing of the reactor building purge ventilation system. One of these is the fuel handling accident in the containment. The other is the ventilation required to maintain acceptable air purity in the containment during normal fuel handling operations. In evaluating these needs, it was found that the ventilation capacity needed to maintain a safe working environment in the containment is greater than that needed to mitigate the effects of a fuel handling accident. Therefore, the system was sized for the normal ventilation needs. Since fuel handling operations only take place when the purge ventilation system is in operation, at least 200% of the purging capacity needed to clean up the containment atmosphere in the postaccident period is operating should an accident occur. Availability is therefore assured to perform the only engineered safety feature function assigned to this system.~~
14. ~~Compliance with ANSI/ASME N509 is not required since the system was designed and fabricated before publication of the ANSI documents. The system conformed to this section of Regulatory Guide 1.52 Rev. 0 at the time of design and fabrication, and leakage testing is performed in accordance with ANSI N509-1976. Whenever possible, parts or components used as replacements comply with the latest issue of ANSI/ASME N509. For welding requirements for ductwork, see Note 16.~~
15. ~~Compliance with ANSI/ASME N510 is not required since the system was designed and fabricated before publications of the ANSI documents. However, the system is tested, when possible, using the procedures outlined in ASME N510-1989.~~
16. ~~Those portions of TVA Classes Q and S Category I duct which are of welded construction and are fabricated or repaired after January 12, 1987, meet the welding requirements of ANSI/ASME N509-1976. The workmanship samples are not required to have penetrant testing (PT) or magnetic testing (MT).~~
17. ~~Space constraints do not permit compliance with this section.~~
18. ~~Laboratory testing frequency of the adsorber shall be in accordance with the requirements of RG1.52 (i.e., 720 hours of system operation) for Mode 6, and RG 1.140 (i.e., at approximately 18–month intervals) for Modes 1-5.~~

9.4.2.3 Safety Evaluation

A fuel handling accident in the Auxiliary Building is detected by the two gamma radiation detectors mounted above the fuel pool, as shown in Figure 9.4-12. The high radiation signals via redundant trains will then shut off the fuel handling and Auxiliary Building general supply and exhaust fans and start the ABGTS, as shown in Figures 9.4-9 and 9.4-10. No credit is taken in the dose or accident analyses for these functions. ~~To accomplish its safety function following a fuel handling accident, t~~ The fuel handling area ventilation system ~~must~~ will accomplish the following functions:

(1) Isolate the normal ventilation pathways between the spent fuel pool and the environment.

(2) Filter the contaminants out of the air by the ABGTS before exhausting it to the environment.

The two redundant radiation monitors (non-safety-related) located above the spent fuel pit assure that the accident is promptly detected and that a high radiation signal is provided to each ventilation train, even if one monitor fails. Also, during refueling operations when containment and/or the annulus is open to the Auxiliary Building ABSCE spaces, a Containment Vent Isolation (CVI) signal from either the operating or refueling unit is procedurally configured to assure that a fuel handling accident in containment is promptly detected and the CVI signal is provided to each ventilation train.

In addition, the Auxiliary Building radiation monitor (non-safety related) which monitors the Auxiliary Building exhaust vent is also capable of providing a high radiation signal to the MCR. A high radiation signal from either of the monitors located above the spent fuel pit or a operating or refueling unit CVI signal whenever containment and/or the annulus is open to the Auxiliary Building ABSCE spaces during refueling operations causes the fuel handling area (FHA) and Auxiliary Building general supply and exhaust fans to shut down and their associated dampers to close, as shown in Figures 9.4-9 and 9.4-10. Each of the two FHA exhaust fans has both train A and train B dampers, to ensure building isolation in the event of one damper's failure to close. As an added safety feature, all ABSCE boundary isolation dampers are designed to fail-closed on loss of instrument air or electrical power.

9.4.6 Reactor Building Purge Ventilating System (RBPVS)

9.4.6.1 Design Bases

The RBPVS is designed to maintain the environment in the primary containment and Shield Building annulus within acceptable limits for equipment operation and for personnel access during inspection, testing, maintenance, and refueling operations, and to provide a filtration path for any through-duct outleakage from the primary containment to limit the release of radioactivity to the environment.

The RBPVS performs three distinct functions, the forced air purge function, the continuous pressure relief function, and the alternate containment pressure relief function.

The forced air purge function is performed by a purge supply and purge exhaust system consisting of two trains, each of which is designed to provide 50% of the capacity needed for normal purging. Each train consists of a supply fan, an exhaust fan, a HEPA filter-charcoal adsorber assembly, containment isolation valves and associated dampers and ductwork. This function provides a means by which containment air may be forcibly exchanged and filtered. The purge function provides a means by which containment air may be forcibly exchanged and filtered. The purge function of the RBPVS is not a safety-related function. ~~However, the filtration units are required to provide a safety-related filtration path following a fuel handling accident until all containment isolation valves are closed~~. The safety functions are to assure isolation of primary containment during an accident and to isolate the purge air supply intake upon receipt of an Auxiliary Building Isolation (ABI) signal.

In the case of a fuel handling accident the filtration units provide a filtration path following a fuel handling accident until all containment isolation valves are closed. However, neither the filtration nor the isolation functions are credited in the Fuel Handling Dose Analysis. Thus they are not safety functions for this accident.

During Operating Modes 1 thru 5, continuous pressure relief is provided by a passive ducting system which passes through containment penetration X-80, through two 100% redundant containment vent air cleanup units (CVACU) containing HEPA filters and charcoal adsorbers. Containment air is moved into the annulus by the motive force created by differential pressure between the two spaces. Filtration redundancy allows maintenance on one unit at a time while maintaining an open pathway through the other. This ventilation pathway is isolable using containment isolation valves FCV-30-40 and FCV-30-37 which are closed ~~during Mode 6 or~~ when containment isolation is required. ~~This system is not required for fuel handling accident mitigation and is not available for that purpose since it is essentially isolated by containment isolation valves during fuel loading or handling activities (Mode 6).~~

The alternate pressure relief function is provided by way of a configuration alignment in the forced air purge system. This function is accomplished by opening lower compartment purge lines (one supply and one exhaust) or one of the two pairs of lines (one supply and one exhaust) in the upper compartment. During purging mode, the purge air fans may or may not be used. To prevent inadvertent pressurization of containment due to supply and exhaust side ductwork flow imbalances, the supply ductwork airflow may be temporarily throttled as needed.

The purge function of the RBPVS is not a safety-related function. ~~However, the filtration units are required to provide a safety-related filtration path following a fuel handling accident.~~

The design bases include provisions to:

(1) Supply fresh air for breathing and contamination control when the primary containment or annulus is occupied.

(2) Exhaust primary containment and annulus air to the outdoors whenever the purge air supply system is operated.

(3) Clean up containment exhaust during normal operation by routing the air through HEPA-carbon filter trains before release to the atmosphere to limit potential release of radioactivity to the environment.

(4) Provide a reduced quantity of ventilating air to permit occupancy of the instrument room during reactor operation. The provisions for 1, 2, and 3 above will apply.

(5) Assure closure of primary and secondary containment isolation valves following accidents which result in the initiation of a containment ventilation isolation signal.

(6) Assure closure of the system air intake dampers, which form part of the ABSCE (see Section 6.2.3.2.1), upon receipt of a signal for Auxiliary Building isolation.

(7) Provide continuous containment pressure relief path through HEPA-carbon filter trains before release to the atmosphere during normal operations.

Items 5 and 6 above are safety-related functions, except in the case of the fuel handling accident.

The primary containment penetrations for the ventilation supply and exhaust subsystems are designed to primary containment structural standards. These are discussed in detail in Section 6.2.4.

The RBPVS is sized to maintain an acceptable working environment within the containment during all normal operations. The system has the capabilities to provide a filtration path for outleakage from the primary containment, and clean up containment atmosphere following a design basis accident. It also has provisions to filter air flow exhausted from containment for pressure control, during normal operation.

The controls are designed to have simultaneous starting and stopping of the matching supply and exhaust equipment and to initiate an automatic shutdown and isolation upon receipt of the containment ventilation isolation signal.

In addition, RBPVS supply fans will shut down and the ABSCE isolation dampers in purge air supply ducts will close on an ABI signal.

~~The RBPVS air cleanup equipment assures that activity released inside containment from a refueling accident and prior to containment isolation, is processed through both HEPA filters and carbon adsorbers before release to the atmosphere. Fuel handling operations inside the primary containment are constrained by the operability requirement for the RBPVS air cleanup units contained in the plant technical specifications.~~

The RBPVS components are designed or qualified to meet Seismic Category I requirements, except all purge ductwork within the containment, up to the inboard isolation valves, and the supply air ductwork from the downstream flange of the ABSCE isolation dampers to the upstream flange of the Shield Building isolation valves, which are designed to meet Seismic Category I(L) requirements.

The primary containment exhaust is monitored by redundant radiation detectors which provide automatic RBPVS isolation upon detecting the setpoint radioactivity in the exhaust air stream. The RBPVS isolation valves automatically close upon the actuation of a containment ventilation isolation signal or upon manual actuation from the MCR. In addition, during fuel handling operations in the Auxiliary Building with containment and/or the annulus open to the Auxiliary Building ABSCE spaces, the RBPVS isolation valves will close upon a high radiation signal from the spent fuel pool radiation monitors via a CVI signal from the operating or refueling unit.

The system air supply and exhaust ducts are routed through the Shield Building annulus to several primary containment penetrations. Two air supply locations are provided for each of the upper and lower compartments and one for the instrument room. Air is supplied to areas of low potential radioactivity and is allowed to flow to the air pickup exhaust points in areas of higher potential radioactivity. The air pickup points, located to exhaust air from the lower compartment and instrument room, also provide an air sweep across the surface of the refueling canal...

The purge function of the RBPVS is not a safety-related function~~. However,~~ and the filtration units are not required to provide a safety-related filtration path following a fuel handling accident. The primary containment isolation valves and intermediate piping of the RBPVS are designed in accordance with ANS safety class 2A; other portions are designated ANS safety class 2B except the purge fans, all purge ductwork within the containment, purge supply air ductwork from the ABSCE boundary, fire protection, and drain piping. The instrument room purge subsystem is not an engineered safety feature, and credit for its operability for a LOCA or a fuel-handling accident is not claimed.

A containment ventilation isolation signal automatically shuts down the fans and isolates the RBPVS by closing its respective dampers and butterfly valves. Each RBPVS primary containment isolation valve is designed for fail safe closing within 4 seconds of receipt of a closure signal for containment penetrations (See Tables 6.2.4- 1 through 6.2.4-4 and Figure 6.2.4-21). The RBPVS primary containment isolation valve locations and descriptions are given in Table 6.2.4-1. Each valve is provided with an air cylinder valve operator, control air solenoid valve, and valve position indicating limit switches.

Smoke detectors, located in the Auxiliary Building air intake and the general ventilation supply ducts, shut down the purge air supply and the incore instrument room purge supply fans and their isolation dampers.

9.4.6.3 Safety Evaluation

Functional analyses and failure modes and effects analysis have shown that the RBPVS meets the containment isolation requirements. ~~The purge air filtration units and associated exhaust ductwork provide a safety-related filtration path following a fuel-handling accident.~~ The CVACUs, performing a continuously filtered containment vent function during normal operation, are isolated by the closure of their containment isolation valves; therefore are not operable after accidents. I~~n addition, the containment ventilation system is not allowed to be used during Mode 6.~~

A functional analysis of the system shows that:

(1) During normal operation, adequate fresh air is provided for breathing and for contamination control when the primary or secondary containment (annulus) is occupied.

(2) Primary and secondary containment exhaust air is cleaned up during normal operations and following a fuel handling accident.

(3) Purge supply and exhaust fan operations cease and isolation dampers in the intake and exhaust ducting close when the system is in the accident isolation mode of operation.

(4) Three signals cause the system to change from the normal purge mode to the accident isolation mode. These signals, which include manual, SIS auto initiate, and high purge exhaust radiation (automatic), initiate a containment ventilation isolation signal. Additionally, during refueling operations whenever containment and/or the annulus is open to the Auxiliary Building ABSCE spaces, a high radiation signal from the spent fuel pool accident radiation monitors or CVI signal from the operating unit automatically cause the system to change from the purge mode to the accident isolation mode.

(5) Discharges from the annulus, during normal operation, which are exhausted through the Auxiliary Building exhaust stack, are monitored at the stack. Although these radiation monitors do not initiate an automatic containment isolation signal, radioactive release limits have been established as a basis for controlling plant discharge during operation. Radioactive releases from the plant resulting from equipment faults of moderate frequency are within 10 CFR 50 Appendix I and 40 CFR 190 limits as specified in the ODCM (See Section 11.3 for further details). In addition, analyses have shown that any accident with the potential consequence to exceed the 10 CFR 100 limits, would be detected by other indicators (see item 4 above) and cause an automatic primary and/or secondary containment isolation.

~~Containment vent system is not allowed to be used during Mode 6.~~

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6.2.3 Secondary Containment Functional Design

Structures included as part of the secondary containment system are the Shield Building of each reactor unit, the Auxiliary Building, the Condensate Demineralizer Waste Evaporator (CDWE) Building and the essential raw cooling water (ERCW) pipe tunnels adjacent to the Auxiliary Building. Depending on the configuration of the plant, the Primary Containment Building(s) may also be included as a structure which is part of the secondary containment system. This condition exists when the primary containment is open to the Auxiliary Building. The emergency gas treatment system (EGTS) is provided for ventilation control and cleanup of the atmosphere inside the annulus between the Shield Building and the Primary Containment Building. The Reactor Building purge air system is also available for cleaning up the atmosphere inside the Shield Building Annulus. Refer to Section 9.4.6 for further details relating to the purge air system. The Auxiliary Building Gas Treatment System (ABGTS) provides a similar contamination control capability in the Auxiliary Building Secondary Containment Enclosure (ABSCE),which includes all of the areas listed above.

6.2.3.1 Design Bases

6.2.3.1.1 Secondary Containment Enclosures

Design bases for the secondary containment structures were devised to assure that an effective barrier exists for airborne fission products that may leak from the primary containment, or the Auxiliary Building fuel handling area, during a loss-of-coolant accident (LOCA). Within the scope of these design bases are requirements that influence the size, structural integrity, and leak tightness of the secondary containment enclosure. Specifically, these include a capability to: (a) maintain an effective barrier for gases and vapors that may leak from the primary containment during all normal and abnormal events; (b) delay the release of any gases and vapors that may leak from the primary containment during accidents; (c) allow gases and vapors that may leak through the primary containment during accidents to flow into the contained air volume within the secondary containment where they are diluted, held up, and purified prior to being released to the environs; (d) bleed to the secondary containment each air-filled containment penetration enclosure which extends beyond the Shield Building and which is formed by automatically actuated isolation valves; (e) maintain an effective barrier for airborne radioactive contaminants, gases, and vapors originating in the ABSCE during normal and abnormal events. Refer to Sections 3.8.1 and 3.8.4 for further details relating to the design of the Shield Building and the Auxiliary Building.

6.2.3.1.3 Auxiliary Building Gas Treatment System (ABGTS)

The design bases for the ABGTS are:

1. To establish and keep an air pressure that is below atmospheric within the portion of the buildings serving as a secondary containment enclosure during accidents.
2. To reduce the concentration of radioactive nuclides in air releases from the secondary containment enclosures to the environs during accidents to levels sufficiently low to keep the site boundary and LPZ dose rates below the 10 CFR 100 guideline values.
3. To withstand the safe shutdown earthquake.
4. To provide for initial and periodic testing of the system capability to function as designed (See Chapter 14.0 for information on initial testing of systems).

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6.2.3.3.3 Auxiliary Building Gas Treatment System (ABGTS)

The ABGTS has the capabilities needed to preserve safety in accidents as severe as a LOCA. This was determined by conducting functional analyses of the system to verify that the system has the proper features for accident mitigation which consist of a failure modes and effects analysis, a review of Regulatory Guide 1.52 sections to assure licensing requirement conformance, and a performance analysis to verify that the system has the desired accident mitigation capabilities. A detailed failure modes and effects analysis is presented in Table 6.2.3-3.

The functional analyses conducted on the ABGTS have shown that:

1. The air intakes for the system are properly located to minimize accident effects. The use of the air intakes provided in the fuel handling and waste disposal areas minimizes the spread of airborne contamination that may be accidentally released at these positions in which the probability of an accidental release, e.g., a fuel handling accident, is more likely. This localization effect is provided without reducing the effectiveness of the system to cope with multiple activity released throughout the ABSCE that may occur during a LOCA. Such coverage is accomplished by utilizing the normal ventilation ducting to draw outside air inleakage from any point along the secondary containment enclosure to the fuel handling and waste disposal areas.

2. Accident indication signals are utilized to bring the ABGTS into operation to assure that the system functions when needed to mitigate accident effects. Accidents in which this system is needed to preserve safety are automatically detected by at least one of the three instrumentation sets used to generate accident signals that result in system startup.

3. System startup reliability is very high. The practice of allowing the automatic startup of both full capacity trains in the system gives greater assurance that one train of equipment functions upon receipt of an accident signal.

4. The method adopted to establish and keep the negative pressure level within this secondary containment enclosure minimizes the time needed to reach the desired pressure level. Initially, the full capacity of the ABGTS fans is utilized for this purpose. After reaching the desired operating level, the system control module allows outside air to enter the air flow network just upstream of the fan at a rate to keep the fans operating at full capacity with the enclosed volume at the desired negative pressure level. In this situation, the amount of air withdrawn from the enclosed volume is equal to the amount of outside air inleakage through the ABSCE. In addition, two vacuum breaker dampers in series are provided to admit outside air in case the modulating dampers fail.

5. The ABSCE is maintained at a slightly negative pressure to reduce the amount of unprocessed air escaping from this secondary containment enclosure to the atmosphere to insignificant quantities. In addition, this negative pressure level is less than that which is maintained within the annulus; such that, any air leakage

between the Auxiliary Building and the Shield Building is from the Auxiliary Building into the Shield Building.

6. The Train A and Train B air cleanup units are sufficiently separated from each other to eliminate the possibility of a single failure destroying the capability to process Auxiliary Building air prior to its release to the atmosphere. Two concrete walls and a distance of more than 80 feet separate the two trains. The use of separate trains of the emergency power system to drive the air cleanup trains gives further assurance of proper equipment separation.

The review of the ABGTS conducted to determine its conformance with Regulatory Guide 1.52 has shown that this system, designed prior to issuance of the guide, is in general agreement with its requirements. Details on compliance with Regulatory Guide 1.52 are given in Table 6.5-2.

The performance analysis conducted to verify that the ABGTS has the required accident mitigation capabilities has shown that the system flow rate is sized properly to handle all expected outside air inleakage at a 1/4-inch water gauge negative pressure differential. This indicates that the nominal flow rate of 9000 cfm is sufficient to assure an adequate margin above the expected ABSCE inleakage (ACU filters are replaced as needed to maintain a minimum flow capability of 9300 cfm under surveillance instructions).

The performance analysis evaluated the capability of the ABGTS to reach and maintain a negative pressure of 1/4-inch water gauge with respect to the outside within the boundaries of the ABSCE. The following was utilized in the analysis:

1. Leakage into the ABSCE is proportional to the square root of the pressure differential.

2. Only one air cleanup unit in the ABGTS operates at the rated capacity.

3. The air cleanup unit fan begins to operate 30 seconds after the initiation of an ABI signal.

4. The initial static pressure inside the ABSCE is conservatively considered to be atmospheric pressure, although the ABSCE is under a negative pressure during normal operation.

5. The effective pressure head due to wind equals 1/8-inch water gauge.

6. Initial average air temperature inside the ABSCE equals 140°F.

7. Atmospheric temperature and pressure are 70°F and 14.4 psia, respectively.

8. ABSCE isolation dampers/valves close within 30 seconds after receiving an ABI signal.

9. The non-safety-related general ventilation and fuel handling area exhaust fans are designed to shut down automatically following a LOCA. Each fan is provided with a safety related Class 1E primary circuit breaker and a safety related Class 1E shunt trip isolation switch which is tripped by a signal of the opposite train from that for the primary circuit breaker to ensure that power is isolated from the fan.

6.2.4.3 Design Evaluation

The containment isolation systems are designed to present a double barrier to any flow path from the inside to the outside of the containment using the double-barrier approach to meet the single-failure criterion.

(f) The design configuration for penetrations X-79A (ice blowing), and X–79B (negative return) is temporarily modified in operating Modes 5 and 6 and when the reactor is defueled (Mode 7) to support ice blowing activities. The normally closed blind flange on each penetration will be opened and temporary piping will be installed in the penetrations. A 12–inch silicone seal will be installed between the piping segment and the penetration. Manual isolation valves will be connected to the piping on the inboard and outboard side of the penetrations. This configuration is being installed to permit ice blowing operations to occur concurrently with fuel handling activities inside containment. The penetrations will be returned to their normal design configuration prior to entry into Mode 4 operations.

6.5 FISSION PRODUCT REMOVAL AND CONTROL SYSTEMS

6.5.1 Engineered Safety Feature (ESF) Filter Systems

Four Engineered Safety Feature (ESF) air cleanup systems' units are provided for fission product removal in post-accident environments. These are:
(1) The emergency gas treatment system (EGTS) air cleanup units.
(2) The Auxiliary Building gas treatment system (ABGTS) air cleanup units.
(3) The Reactor Building purge system air cleanup units.
(4) The Main Control Room emergency air cleanup units.

6.5.1.1 Design Bases

6.5.1.1.1 Emergency Gas Treatment System Air Cleanup Units

The design bases are:
(1) To provide fission product removal capabilities sufficient to keep radioactivity levels in the Shield Building annulus air released to the environs during a DBA LOCA sufficiently low to assure compliance with 10 CFR 100 guidelines.
(2) These air cleanup units are a part of the EGTS. See Section 6.2.3.1.2 for the design bases for other portions of this system.

6.5.1.1.2 Auxiliary Building Gas Treatment System Air Cleanup Units

The design bases are:
(1) To provide fission product removal capabilities sufficient to keep radioactivity levels in the Auxiliary Building secondary containment enclosure (ABSCE) air released to the environs during a postulated accident sufficiently low to assure compliance with 10 CFR 100 guidelines.
(2) These air cleanup units are a part of the ABGTS. See Section 6.2.3.1.3 for the design basis for other portions of this system.

9.4.2.3 Safety Evaluation

A fuel handling accident in the Auxiliary Building is detected by the two gamma radiation detectors mounted above the fuel pool, as shown in Figure 9.4-12. The high radiation signals via redundant trains will then shut off the fuel handling and Auxiliary Building general supply and exhaust fans and start the ABGTS, as shown in Figures 9.4-9 and 9.4-10. No credit is taken in the dose or accident analyses for these functions. The fuel handling area ventilation system will accomplish the following functions:

(1) Isolate the normal ventilation pathways between the spent fuel pool and the environment.

(2) Filter the contaminants out of the air by the ABGTS before exhausting it to the environment.

The two redundant radiation monitors (non-safety-related) located above the spent fuel pit assure that the accident is promptly detected and that a high radiation signal is provided to each ventilation train, even if one monitor fails. Also, during refueling operations when containment and/or the annulus is open to the Auxiliary Building ABSCE spaces, a Containment Vent Isolation (CVI) signal from either the operating or refueling unit is procedurally configured to assure that a fuel handling accident in containment is promptly detected and the CVI signal is provided to each ventilation train.

In addition, the Auxiliary Building radiation monitor (non-safety related) which monitors the Auxiliary Building exhaust vent is also capable of providing a high radiation signal to the MCR. A high radiation signal from either of the monitors located above the spent fuel pit or a operating or refueling unit CVI signal whenever containment and/or the annulus is open to the Auxiliary Building ABSCE spaces during refueling operations causes the fuel handling area (FHA) and Auxiliary Building general supply and exhaust fans to shut down and their associated dampers to close, as shown in Figures 9.4-9 and 9.4-10. Each of the two FHA exhaust fans has both train A and train B dampers, to ensure building isolation in the event of one damper's failure to close. As an added safety feature, all ABSCE boundary isolation dampers are designed to fail-closed on loss of instrument air or electrical power.

9.4.6 Reactor Building Purge Ventilating System (RBPVS)

9.4.6.1 Design Bases

The RBPVS is designed to maintain the environment in the primary containment and Shield Building annulus within acceptable limits for equipment operation and for personnel access during inspection, testing, maintenance, and refueling operations, and to provide a filtration path for any through-duct outleakage from the primary containment to limit the release of radioactivity to the environment.

The RBPVS performs three distinct functions, the forced air purge function, the continuous pressure relief function, and the alternate containment pressure relief function.

The forced air purge function is performed by a purge supply and purge exhaust system consisting of two trains, each of which is designed to provide 50% of the capacity needed for normal purging. Each train consists of a supply fan, an exhaust fan, a HEPA filter-charcoal adsorber assembly, containment isolation valves and associated dampers and ductwork. This function provides a means by which containment air may be forcibly exchanged and filtered. The purge function provides a means by which containment air may be forcibly exchanged and filtered. The purge function of the RBPVS is not a safety-related function. The safety functions are to assure isolation of primary containment during an accident and to isolate the purge air supply intake upon receipt of an Auxiliary Building Isolation (ABI) signal.

In the case of a fuel handling accident the filtration units provide a filtration path following a fuel handling accident until all containment isolation valves are closed. However, neither the filtration nor the isolation functions are credited in the Fuel Handling Dose Analysis. Thus they are not safety functions for this accident.

During Operating Modes 1 thru 5, continuous pressure relief is provided by a passive ducting system which passes through containment penetration X-80, through two 100% redundant containment vent air cleanup units (CVACU) containing HEPA filters and charcoal adsorbers. Containment air is moved into the annulus by the motive force created by differential pressure between the two spaces. Filtration redundancy allows maintenance on one unit at a time while maintaining an open pathway through the other. This ventilation pathway is isolable using containment isolation valves FCV-30-40 and FCV-30-37 which are closed when containment isolation is required.

The alternate pressure relief function is provided by way of a configuration alignment in the forced air purge system. This function is accomplished by opening lower compartment purge lines (one supply and one exhaust) or one of the two pairs of lines (one supply and one exhaust) in the upper compartment. During purging mode, the purge air fans may or may not be used. To prevent inadvertent pressurization of containment due to supply and exhaust side ductwork flow imbalances, the supply ductwork airflow may be temporarily throttled as needed.

The purge function of the RBPVS is not a safety-related function.

The design bases include provisions to:

(1) Supply fresh air for breathing and contamination control when the primary containment or annulus is occupied.

(2) Exhaust primary containment and annulus air to the outdoors whenever the purge air supply system is operated.

(3) Clean up containment exhaust during normal operation by routing the air through HEPA-carbon filter trains before release to the atmosphere to limit potential release of radioactivity to the environment.

(4) Provide a reduced quantity of ventilating air to permit occupancy of the instrument room during reactor operation. The provisions for 1, 2, and 3 above will apply.

(5) Assure closure of primary and secondary containment isolation valves following accidents which result in the initiation of a containment ventilation isolation signal.

(6) Assure closure of the system air intake dampers, which form part of the ABSCE (see Section 6.2.3.2.1), upon receipt of a signal for Auxiliary Building isolation.

(7) Provide continuous containment pressure relief path through HEPA-carbon filter trains before release to the atmosphere during normal operations.

Items 5 and 6 above are safety-related functions, except in the case of the fuel handling accident.

The primary containment penetrations for the ventilation supply and exhaust subsystems are designed to primary containment structural standards. These are discussed in detail in Section 6.2.4.

The RBPVS is sized to maintain an acceptable working environment within the containment during all normal operations. The system has the capabilities to provide a filtration path for outleakage from the primary containment, and clean up containment atmosphere following a design basis accident. It also has provisions to filter air flow exhausted from containment for pressure control, during normal operation.

The controls are designed to have simultaneous starting and stopping of the matching supply and exhaust equipment and to initiate an automatic shutdown and isolation upon receipt of the containment ventilation isolation signal.

In addition, RBPVS supply fans will shut down and the ABSCE isolation dampers in purge air supply ducts will close on an ABI signal.

The RBPVS components are designed or qualified to meet Seismic Category I requirements, except all purge ductwork within the containment, up to the inboard isolation valves, and the supply air ductwork from the downstream flange of the ABSCE isolation dampers to the upstream flange of the Shield Building isolation valves, which are designed to meet Seismic Category I(L) requirements.

The primary containment exhaust is monitored by redundant radiation detectors which provide automatic RBPVS isolation upon detecting the setpoint radioactivity in the exhaust air stream. The RBPVS isolation valves automatically close upon the actuation of a containment ventilation isolation signal or upon manual actuation from the MCR. In addition, during fuel handling operations in the Auxiliary Building with containment and/or the annulus open to the Auxiliary Building ABSCE spaces, the RBPVS isolation valves will close upon a high radiation signal from the spent fuel pool radiation monitors via a CVI signal from the operating or refueling unit.

The system air supply and exhaust ducts are routed through the Shield Building annulus to several primary containment penetrations. Two air supply locations are provided for each of the upper and lower compartments and one for the instrument room. Air is supplied to areas of low potential radioactivity and is allowed to flow to the air pickup exhaust points in areas of higher potential radioactivity. The air pickup points, located to exhaust air from the lower compartment and instrument room, also provide an air sweep across the surface of the refueling canal...

The purge function of the RBPVS is not a safety-related function and the filtration units are not required to provide a safety-related filtration path following a fuel handling accident. The primary containment isolation valves and intermediate piping of the RBPVS are designed in accordance with ANS safety class 2A; other portions are designated ANS safety class 2B except the purge fans, all purge ductwork within the containment, purge supply air ductwork from the ABSCE boundary, fire protection, and drain piping. The instrument room purge subsystem is not an engineered safety feature, and credit for its operability for a LOCA or a fuel-handling accident is not claimed.

A containment ventilation isolation signal automatically shuts down the fans and isolates the RBPVS by closing its respective dampers and butterfly valves. Each RBPVS primary containment isolation valve is designed for fail safe closing within 4 seconds of receipt of a closure signal for containment penetrations (See Tables 6.2.4- 1 through 6.2.4-4 and Figure 6.2.4-21). The RBPVS primary containment isolation valve locations and descriptions are given in Table 6.2.4-1. Each valve is provided with an air cylinder valve operator, control air solenoid valve, and valve position indicating limit switches.

Smoke detectors, located in the Auxiliary Building air intake and the general ventilation supply ducts, shut down the purge air supply and the incore instrument room purge supply fans and their isolation dampers.

9.4.6.3 Safety Evaluation

Functional analyses and failure modes and effects analysis have shown that the RBPVS meets the containment isolation requirements. The CVACUs, performing a continuously filtered containment vent function during normal operation, are isolated by the closure of their containment isolation valves; therefore are not operable after accidents.

A functional analysis of the system shows that:

(1) During normal operation, adequate fresh air is provided for breathing and for contamination control when the primary or secondary containment (annulus) is occupied.

(2) Primary and secondary containment exhaust air is cleaned up during normal operations and following a fuel handling accident.

(3) Purge supply and exhaust fan operations cease and isolation dampers in the intake and exhaust ducting close when the system is in the accident isolation mode of operation.

(4) Three signals cause the system to change from the normal purge mode to the accident isolation mode. These signals, which include manual, SIS auto initiate, and high purge exhaust radiation (automatic), initiate a containment ventilation isolation signal. Additionally, during refueling operations whenever containment and/or the annulus is open to the Auxiliary Building ABSCE spaces, a high radiation signal from the spent fuel pool accident radiation monitors or CVI signal from the operating unit automatically cause the system to change from the purge mode to the accident isolation mode.

(5) Discharges from the annulus, during normal operation, which are exhausted through the Auxiliary Building exhaust stack, are monitored at the stack. Although these radiation monitors do not initiate an automatic containment isolation signal, radioactive release limits have been established as a basis for controlling plant discharge during operation. Radioactive releases from the plant resulting from equipment faults of moderate frequency are within 10 CFR 50 Appendix I and 40 CFR 190 limits as specified in the ODCM (See Section 11.3 for further details). In addition, analyses have shown that any accident with the potential consequence to exceed the 10 CFR 100 limits, would be detected by other indicators (see item 4 above) and cause an automatic primary and/or secondary containment isolation.

Enclosure 6

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Red-Line Markup

3.3 INSTRUMENTATION

3.3.6 Containment Vent Isolation Instrumentation

LCO 3.3.6 The Containment Vent Isolation instrumentation for each Function in Table 3.3.6-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4,
~~During movement of irradiated fuel assemblies within containment.~~

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One radiation monitoring channel inoperable.	A.1	Restore the affected channel to OPERABLE status.	4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. ~~-------------NOTE-------------~~ ~~Only applicable in MODE 1, 2, 3, or 4.~~ ------------------------------------ One or more Functions with one or more manual or automatic actuation trains inoperable. OR Two radiation monitoring channels inoperable. OR Required Action and associated Completion Time of Condition A not met.	--------------------NOTE------------------- One train of automatic actuation logic may be bypassed and Required Action B.1 may be delayed for up to 4 hours for Surveillance testing provided the other train is OPERABLE. -- B.1 Enter applicable Conditions and Required Actions of LCO 3.6.3, "Containment Isolation Valves," for containment purge and exhaust isolation valves made inoperable by isolation instrumentation.	 Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
~~C. ------------NOTE------------ Only applicable during movement of irradiated fuel assemblies within containment. ------------------------------~~	~~C.1~~	~~Place and maintain containment purge and exhaust valves in closed position.~~	~~Immediately~~
	~~OR~~		
~~One or more Functions with one or more manual or automatic actuation trains inoperable.~~ ~~OR~~ ~~Two radiation monitoring channels inoperable.~~ ~~OR~~ ~~Required Action and associated Completion Time for Condition A not met.~~	~~C.2~~	~~Enter applicable Conditions and Required Actions of LCO 3.9.4, "Containment Penetrations," for containment purge and exhaust isolation valves made inoperable by isolation instrumentation.~~	~~Immediately~~

SURVEILLANCE REQUIREMENTS

--NOTE--
Refer to Table 3.3.6-1 to determine which SRs apply for each Containment Vent Isolation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.6.2	------------------------------NOTE------------------------------ This surveillance is only applicable to the actuation logic of the ESFAS instrumentation. --- Perform ACTUATION LOGIC TEST.	92 days on a STAGGERED TEST BASIS
SR 3.3.6.3	------------------------------NOTE------------------------------ This surveillance is only applicable to the master relays of the ESFAS instrumentation. --- Perform MASTER RELAY TEST.	92 days on a STAGGERED TEST BASIS
SR 3.3.6.4	Perform COT.	92 days
SR 3.3.6.5	Perform SLAVE RELAY TEST.	92 days <u>OR</u> 18 months for Westinghouse type AR and Potter & Brumfield MDR Series relays

(continued)

SURVEILLANCE REQUIREMENTS (Continued)

SURVEILLANCE		FREQUENCY
SR 3.3.6.6	------------------------------NOTE------------------------------ Verification of setpoint is not required. --- Perform TADOT.	 18 months
SR 3.3.6.7	Perform CHANNEL CALIBRATION.	18 months

Table 3.3.6-1 (page 1 of 1)
Containment Vent Isolation Instrumentation

FUNCTION	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Manual Initiation	2	SR 3.3.6.6	NA
2. Automatic Actuation Logic and Actuation Relays	2 trains	SR 3.3.6.2 SR 3.3.6.3 SR 3.3.6.5	NA
3. Containment Purge Exhaust Radiation Monitors	2	SR 3.3.6.1 SR 3.3.6.4 SR 3.3.6.7	~~≤ 8.41E-02 μCi/cc[(a)] (3.43x10^4 cpm)~~ ≤ 2.8E-02 μCi/cc~~[(b)]~~ (1.14x10^4 cpm)
4. Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.		

~~(a) During movement of irradiated fuel assemblies within containment.~~

~~(b) Modes 1, 2, 3, and 4.~~

3.3 INSTRUMENTATION

3.3.8 Auxiliary Building Gas Treatment System (ABGTS) Actuation Instrumentation

LCO 3.3.8 The ABGTS actuation instrumentation for each Function in Table 3.3.8-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.8-1.

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more Functions with one channel or train inoperable.	A.1	Place one ABGTS train in operation.	7 days
B. One or more Functions with two channels or two trains inoperable.	B.1.1	Place one ABGTS train in operation.	Immediately
	AND		
	B.1.2	Enter applicable Conditions and Required Actions of LCO 3.7.12, "Auxiliary Building Gas Treatment System (ABGTS)," for one train made inoperable by inoperable actuation instrumentation	Immediately
	OR		

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Place both trains in emergency radiation protection mode.	Immediately
~~C. Required Action and associated Completion Time for Condition A or B not met during movement of irradiated fuel assemblies in the fuel handling area.~~	~~C.1~~	~~Suspend movement of irradiated fuel assemblies in the fuel handling area~~.	~~Immediately~~
~~D.~~ C. Required Action and associated Completion Time for Condition A or B not met ~~in MODE 1, 2, 3, or 4.~~	C~~D~~.1	Be in MODE 3.	6 hours
	AND		
	C~~D~~.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

---NOTE---

Refer to Table 3.3.8-1 to determine which SRs apply for each ABGTS Actuation Function.

SURVEILLANCE		FREQUENCY
~~SR 3.3.8.1~~	~~Perform CHANNEL CHECK.~~	~~12 hours~~
~~SR 3.3.8.2~~	~~Perform COT.~~	~~92 days~~

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.8.~~3~~1	------------------------------NOTE----------------------------- Verification of setpoint is not required. --- Perform TADOT.	18 months
~~SR 3.3.8.4~~	~~Perform CHANNEL CALIBRATION.~~	~~18 months~~

Table 3.3.8-1 (page 1 of 1)
ABGTS Actuation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Manual Initiation	1,2,3,4 ~~(a)~~	2 ~~2~~	SR 3.3.8.~~3~~1 ~~SR 3.3.8.3~~	NA ~~NA~~
~~2. Fuel Pool Area Radiation Monitors~~	~~(a)~~	~~2~~	~~SR 3.3.8.1~~ ~~SR 3.3.8.2~~ ~~SR 3.3.8.4~~	~~≤ 1161 mR/hr~~
2. ~~3.~~ Containment Isolation	Refer to LCO 3.3.2, Function 3.a., for all Phase A initiating functions and requirements.			

~~(a) During movement of irradiated fuel assemblies in the fuel handling area.~~

3.7 PLANT SYSTEMS

3.7.12 Auxiliary Building Gas Treatment System (ABGTS)

LCO 3.7.12 Two ABGTS trains shall be OPERABLE

APPLICABILITY: MODES 1, 2, 3, and 4,
~~During movement of irradiated fuel assemblies in the fuel handling area.~~

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One ABGTS train inoperable	A.1	Restore ABGTS train to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met ~~in MODE 1, 2, 3, or 4~~. OR Two ABGTS trains inoperable ~~in MODE 1, 2, 3, or 4~~.	B.1 AND B.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours
~~C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the fuel handling area.~~	~~C.1~~ ~~OR~~ ~~C.2~~	~~Place OPERABLE ABGTS train in operation.~~ ~~Suspend movement of irradiated fuel assemblies in the fuel handling area~~	~~Immediately~~ ~~Immediately~~
~~D. Two ABGTS trains inoperable during movement of irradiated fuel assemblies in the fuel handling area.~~	~~D.1~~	~~Suspend movement of irradiated fuel assemblies in the fuel handling area.~~	~~Immediately~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Operate each ABGTS train for ≥ 10 continuous hours with the heaters operating.	31 days
SR 3.7.12.2	Perform required ABGTS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.12.3	Verify each ABGTS train actuates on an actual or simulated actuation signal.	18 months
SR 3.7.12.4	Verify one ABGTS train can maintain a pressure between -0.25 inches and -0.5 inches water gauge with respect to atmospheric pressure during the post accident mode of operation at a flow rate ≥ 9300 cfm and ≤ 9900 cfm.	18 months on a STAGGERED TEST BASIS

~~3.9 REFUELING OPERATIONS~~

~~3.9.4 Containment Penetrations~~

~~LCO 3.9.4~~ ~~The containment penetrations shall be in the following status:~~

~~a. The equipment hatch closed and held in place by a minimum of four bolts;~~

~~b. One door in each air lock closed; or capable of being closed provided ABGTS is OPERABLE in accordance with TS 3.7.12; and~~

~~c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:~~

~~1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or~~

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

---~~NOTE~~---

~~Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls provided ABGTS is OPERABLE in accordance with TS 3.7.12.~~

~~APPLICABILITY:~~ ~~During movement of irradiated fuel assemblies within containment.~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~		~~COMPLETION TIME~~
~~C. 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~~	~~Verify each required containment vent isolation valve actuates to the isolation position on an actual or simulated actuation signal.~~	~~18 months~~

3.9 REFUELING OPERATIONS

~~3.9.8 Reactor Building Purge Air Cleanup Units~~

~~LCO 3.9.8~~ ~~Two Reactor Building Purge Air Cleanup Units shall be OPERABLE.~~

~~APPLICABILITY:~~ ~~During movement of irradiated fuel assemblies within the containment.~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One Reactor Building Purge Air Cleanup Unit inoperable.~~	~~A.1~~ ~~Isolate the inoperable air cleanup unit.~~	~~Immediately~~
	~~AND~~	
	~~A.2~~ ~~Verify the OPERABLE air cleanup unit is in operation.~~	~~Immediately~~
~~B. Two Reactor Building Purge Air Cleanup Units inoperable.~~	~~B.1~~ ~~Suspend movement of irradiated fuel assemblies within containment.~~	~~Immediately~~

~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~		~~FREQUENCY~~
~~SR 3.9.8.1~~	~~Perform required filter testing in accordance with the Ventilation Filter Testing Program (VFTP).~~	~~In accordance with the VFTP~~

3.9 REFUELING OPERATIONS

3.9.10 Decay Time

LCO 3.9.10 The reactor shall be subcritical for ≥ 100 hours.

APPLICABILITY: During movement of irradiated fuel assemblies within the containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor subcritical for < 100 hours.	A.1 Suspend all operations involving movement of irradiated fuel assemblies within the containment.	Immediately

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.10.1	Verify the reactor has been subcritical for ≥ 100 hours by confirming the date and time of subcriticality.	Prior to movement of irradiated fuel in the reactor vessel

5.7.2.14 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in accordance with Regulatory Guide 1.52, Revision 2; ASME N510-1989, and the exceptions noted for each ESF system in Tables 6.5-1, 6.5-2, 6.5-3, and 6.5-4 of the FSAR.

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass within acceptance criterion when tested in accordance with Regulatory Guide 1.52, Revision 2, the exceptions noted for each ESF system in Tables 6.5-1, 6.5-2, 6.5-3, and 6.5-4 of the FSAR, and ASME N510-1989 at the system flowrate specified below.

ESF VENTILATION SYSTEM	ACCEPTANCE CRITERIA	FLOW RATE
~~Reactor Building Purge~~	~~< 1.00%~~	~~14,000 cfm ± 10%~~
Emergency Gas Treatment	< 0.05%	4,000 cfm ± 10%
Auxiliary Building Gas Treatment	< 0.05%	9,000 cfm ± 10%
Control Room Emergency	< 1.00%	4,000 cfm ± 10%

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.14 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass within acceptance criterion when tested in accordance with Regulatory Guide 1.52, Revision 2, the exceptions noted for each ESF system in Tables 6.5-1, 6.5-2, 6.5-3, and 6.5-4 of the FSAR, and ASME N510-1989 at the system flowrate specified below.

ESF VENTILATION SYSTEM	ACCEPTANCE CRITERIA	FLOW RATE
~~Reactor Building Purge~~	~~< 1.00%~~	~~14,000 cfm ± 10%~~
Emergency Gas Treatment	< 0.05%	4,000 cfm ± 10%
Auxiliary Building Gas Treatment	< 0.05%	9,000 cfm ± 10%
Control Room Emergency	< 1.00%	4,000 cfm ± 10%

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.14 Ventilation Filter Testing Program (VFTP) (continued)

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, and the exceptions noted for each ESF system in Tables 6.5-1, 6.5-2, 6.5-3, and 6.5-4 of the FSAR, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of ≤ 30°C and greater than or equal to the relative humidity specified below.

ESF VENTILATION SYSTEM	METHYL IODIDE PENETRATION	RELATIVE HUMIDITY
~~Reactor Building Purge~~	~~< 10%~~	~~95%~~
Emergency Gas Treatment	< 0.175%	70%
Auxiliary Building Gas Treatment	< 0.175%	70%
Control Room Emergency	< 1.0%	70%

d. Demonstrate for each of the ESF systems that the pressure drop across the entire filtration unit is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, the exceptions noted for each ESF system in Tables 6.5-1, 6.5-2, 6.5-3, and 6.5-4 of the FSAR, and ASME N510-1989 at the system flowrate specified below.

ESF VENTILATION SYSTEM	PRESSURE DROP	FLOW RATE
~~Reactor Building Purge~~	~~< 4.7 inches water~~	~~14,000 cfm ± 10%~~
Emergency Gas Treatment	< 7.6 inches water	4,000 cfm ± 10%
Auxiliary Building Gas Treatment	< 7.6 inches water	9,000 cfm ± 10%
Control Room Emergency	< 3.5 inches water	4,000 cfm ± 10%

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.14 Ventilation Filter Testing Program (VFTP) (continued)

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

ESF VENTILATION SYSTEM	AMOUNT OF HEAT
Emergency Gas Treatment	20 $\pm$ 2.0 kW
Auxiliary Building Gas Treatment	50 $\pm$ 5.0 kW

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

5.7.2.15 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 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."

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Waste Gas Holdup System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., the system is not designed to withstand a hydrogen explosion);

(continued)

5.7.2.19 Containment Leakage Rate Testing Program (continued)

Leakage rate acceptance criteria are:

a. Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests.

b. Air lock testing acceptance criteria are:

 1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

 2. For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 6 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

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

5.7.2.20 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of the applicable regulatory requirement (i.e., 5 rem Total Effective Dose Equivalent (TEDE) for a fuel handling accident or 5 rem whole body or its equivalent to any part of the body) for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

5.7 Procedures, Programs, and Manuals

Enclosure 7

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3.3 INSTRUMENTATION

3.3.6 Containment Vent Isolation Instrumentation

LCO 3.3.6 The Containment Vent Isolation instrumentation for each Function in Table 3.3.6-1 shall be OPERABLE.

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

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One radiation monitoring channel inoperable.	A.1	Restore the affected channel to OPERABLE status.	4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. --------------NOTE-------------- One or more Functions with one or more manual or automatic actuation trains inoperable. <u>OR</u> Two radiation monitoring channels inoperable. <u>OR</u> Required Action and associated Completion Time of Condition A not met.	-------------------NOTE------------------- One train of automatic actuation logic may be bypassed and Required Action B.1 may be delayed for up to 4 hours for Surveillance testing provided the other train is OPERABLE. --- B.1 Enter applicable Conditions and Required Actions of LCO 3.6.3, "Containment Isolation Valves," for containment purge and exhaust isolation valves made inoperable by isolation instrumentation.	 Immediately

(continued)

SURVEILLANCE REQUIREMENTS

--NOTE--
Refer to Table 3.3.6-1 to determine which SRs apply for each Containment Vent Isolation Function.

	SURVEILLANCE	FREQUENCY
SR 3.3.6.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.6.2	------------------------------NOTE---------------------------- This surveillance is only applicable to the actuation logic of the ESFAS instrumentation. --- Perform ACTUATION LOGIC TEST.	92 days on a STAGGERED TEST BASIS
SR 3.3.6.3	------------------------------NOTE---------------------------- This surveillance is only applicable to the master relays of the ESFAS instrumentation. --- Perform MASTER RELAY TEST.	92 days on a STAGGERED TEST BASIS
SR 3.3.6.4	Perform COT.	92 days
SR 3.3.6.5	Perform SLAVE RELAY TEST.	92 days <u>OR</u> 18 months for Westinghouse type AR and Potter & Brumfield MDR Series relays

(continued)

SURVEILLANCE REQUIREMENTS (Continued)

SURVEILLANCE		FREQUENCY
SR 3.3.6.6	------------------------------NOTE------------------------------ Verification of setpoint is not required. -- Perform TADOT.	18 months
SR 3.3.6.7	Perform CHANNEL CALIBRATION.	18 months

Table 3.3.6-1 (page 1 of 1)
Containment Vent Isolation Instrumentation

FUNCTION	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Manual Initiation	2	SR 3.3.6.6	NA
2. Automatic Actuation Logic and Actuation Relays	2 trains	SR 3.3.6.2 SR 3.3.6.3 SR 3.3.6.5	NA
3. Containment Purge Exhaust Radiation Monitors	2	SR 3.3.6.1 SR 3.3.6.4 SR 3.3.6.7	$\leq$ 2.8E-02 μCi/cc (1.14×10^4 cpm)
4. Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.		

3.3 INSTRUMENTATION

3.3.8 Auxiliary Building Gas Treatment System (ABGTS) Actuation Instrumentation

LCO 3.3.8 The ABGTS actuation instrumentation for each Function in Table 3.3.8-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.8-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one channel or train inoperable.	A.1 Place one ABGTS train in operation.	7 days
B. One or more Functions with two channels or two trains inoperable.	B.1.1 Place one ABGTS train in operation. AND B.1.2 Enter applicable Conditions and Required Actions of LCO 3.7.12, "Auxiliary Building Gas Treatment System (ABGTS)," for one train made inoperable by inoperable actuation instrumentation OR	Immediately Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Place both trains in emergency radiation protection mode.	Immediately
C. Required Action and associated Completion Time for Condition A or B not met.	C.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	C.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

---NOTE---

Refer to Table 3.3.8-1 to determine which SRs apply for each ABGTS Actuation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.8.1	------------------------------NOTE------------------------------ Verification of setpoint is not required. --- Perform TADOT.	18 months

Table 3.3.8-1 (page 1 of 1)
ABGTS Actuation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Manual Initiation	1,2,3,4	2	SR 3.3.8.3	NA
2. Containment Isolation	Refer to LCO 3.3.2, Function 3.a., for all Phase A initiating functions and requirements.			

3.7 PLANT SYSTEMS

3.7.12 Auxiliary Building Gas Treatment System (ABGTS)

LCO 3.7.12 Two ABGTS trains shall be OPERABLE

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One ABGTS train inoperable	A.1	Restore ABGTS train to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met	B.1	Be in MODE 3.	6 hours
	AND		
OR	B.2	Be in MODE 5.	36 hours
Two ABGTS trains inoperable			

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Operate each ABGTS train for ≥ 10 continuous hours with the heaters operating.	31 days
SR 3.7.12.2	Perform required ABGTS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.12.3	Verify each ABGTS train actuates on an actual or simulated actuation signal.	18 months
SR 3.7.12.4	Verify one ABGTS train can maintain a pressure between -0.25 inches and -0.5 inches water gauge with respect to atmospheric pressure during the post accident mode of operation at a flow rate ≥ 9300 cfm and ≤ 9900 cfm.	18 months on a STAGGERED TEST BASIS

3.9 REFUELING OPERATIONS

3.9.10 Decay Time

LCO 3.9.10 The reactor shall be subcritical for ≥ 100 hours.

APPLICABILITY: During movement of irradiated fuel assemblies within the containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor subcritical for < 100 hours.	A.1 Suspend all operations involving movement of irradiated fuel assemblies within the containment.	Immediately

TECHNICAL SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.9.10.1	Verify the reactor has been subcritical for ≥ 100 hours by confirming the date and time of subcriticality.	Prior to movement of irradiated fuel in the reactor vessel

5.7 Procedures, Programs, and Manuals

5.7.2.14 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in accordance with Regulatory Guide 1.52, Revision 2; ASME N510-1989, and the exceptions noted for each ESF system in Tables 6.5-1, 6.5-2, 6.5-3, and 6.5-4 of the FSAR.

f. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass within acceptance criterion when tested in accordance with Regulatory Guide 1.52, Revision 2, the exceptions noted for each ESF system in Tables 6.5-1, 6.5-2, 6.5-3, and 6.5-4 of the FSAR, and ASME N510-1989 at the system flowrate specified below.

ESF VENTILATION SYSTEM	ACCEPTANCE CRITERIA	FLOW RATE
Emergency Gas Treatment	< 0.05%	4,000 cfm ± 10%
Auxiliary Building Gas Treatment	< 0.05%	9,000 cfm ± 10%
Control Room Emergency	< 1.00%	4,000 cfm ± 10%

(continued)

5.7.2.14 Ventilation Filter Testing Program (VFTP) (continued)

g. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass within acceptance criterion when tested in accordance with Regulatory Guide 1.52, Revision 2, the exceptions noted for each ESF system in Tables 6.5-1, 6.5-2, 6.5-3, and 6.5-4 of the FSAR, and ASME N510-1989 at the system flowrate specified below.

ESF VENTILATION SYSTEM	ACCEPTANCE CRITERIA	FLOW RATE
Emergency Gas Treatment	< 0.05%	4,000 cfm $\pm$ 10%
Auxiliary Building Gas Treatment	< 0.05%	9,000 cfm $\pm$ 10%
Control Room Emergency	< 1.00%	4,000 cfm $\pm$ 10%

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.14 Ventilation Filter Testing Program (VFTP) (continued)

h. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, and the exceptions noted for each ESF system in Tables 6.5-1, 6.5-2, 6.5-3, and 6.5-4 of the FSAR, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of ≤ 30°C and greater than or equal to the relative humidity specified below.

ESF VENTILATION SYSTEM	METHYL IODIDE PENETRATION	RELATIVE HUMIDITY
Emergency Gas Treatment	< 0.175%	70%
Auxiliary Building Gas Treatment	< 0.175%	70%
Control Room Emergency	< 1.0%	70%

i. Demonstrate for each of the ESF systems that the pressure drop across the entire filtration unit is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, the exceptions noted for each ESF system in Tables 6.5-1, 6.5-2, 6.5-3, and 6.5-4 of the FSAR, and ASME N510-1989 at the system flowrate specified below.

ESF VENTILATION SYSTEM	PRESSURE DROP	FLOW RATE
Emergency Gas Treatment	< 7.6 inches water	4,000 cfm ± 10%
Auxiliary Building Gas Treatment	< 7.6 inches water	9,000 cfm ± 10%
Control Room Emergency	< 3.5 inches water	4,000 cfm ± 10%

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.14 Ventilation Filter Testing Program (VFTP) (continued)

j. Demonstrate that the heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ASME N510-1989.

ESF VENTILATION SYSTEM	AMOUNT OF HEAT
Emergency Gas Treatment	20 ± 2.0 kW
Auxiliary Building Gas Treatment	50 ± 5.0 kW

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

5.7.2.15 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 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."

The program shall include:

b. The limits for concentrations of hydrogen and oxygen in the Waste Gas Holdup System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., the system is not designed to withstand a hydrogen explosion);

(continued)

5.7.2.19 Containment Leakage Rate Testing Program (continued)

Leakage rate acceptance criteria are:

c. Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests.

d. Air lock testing acceptance criteria are:

   1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   2. For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 6 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

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

5.7.2.20 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of the applicable regulatory requirement (i.e., 5 rem Total Effective Dose Equivalent (TEDE) for a fuel handling accident or 5 rem whole body or its equivalent to any part of the body) for the duration of the accident. The program shall include the following elements:

c. The definition of the CRE and the CRE boundary.

d. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

Enclosure 8

WBN Unit 2 – Revised Technical Specification Bases
Red-Line Markup

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 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.~~

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

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.

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)

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.

LCO
(continued)

3. Containment Radiation

The LCO specifies two required channels of radiation monitors to ensure that the radiation monitoring instrumentation necessary to initiate Containment Vent Isolation remains OPERABLE.

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

Only the Allowable Value is specified for the Containment Purge Exhaust Radiation Monitors in the LCO. The Allowable Value is based on expected concentrations for a small break LOCA, which is more restrictive than 10 CFR 100 limits. The Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis in order to account for instrument uncertainties appropriate to the trip function. The actual nominal Trip Setpoint is normally still more conservative than that required by the Allowable Value. If the setpoint does not exceed the Allowable Value, the radiation monitor is considered OPERABLE.

4. Safety Injection (SI)

Refer to LCO 3.3.2, Function 1, for all initiating Functions and requirements.

APPLICABILITY

The Manual Initiation, Automatic Actuation Logic and Actuation Relays, Safety Injection, and Containment Radiation Functions are required OPERABLE in MODES 1, 2, 3, and 4~~, and during movement of irradiated fuel assemblies within containment~~. Under these conditions, the potential exists for an accident that could release significant fission product radioactivity into containment. Therefore, the Containment Vent Isolation Instrumentation must be OPERABLE in these MODES. See additional discussion in the Background and Applicable Safety Analysis sections.

While in MODES 5 and 6 ~~without fuel handling in progress,~~ the Containment Vent Isolation Instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within the limits of Reference 1.

ACTIONS

The most common cause of channel inoperability is outright failure or drift sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately, and the appropriate Condition entered.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.6-1. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies to the failure of one containment purge isolation radiation monitor channel. Since the two containment radiation monitors are both gaseous detectors, failure of a single channel may result in loss of the redundancy. Consequently, the failed channel must be restored to OPERABLE status. The 4 hours allowed to restore the affected channel is justified by the low likelihood of events occurring during this interval, and recognition that one or more of the remaining channels will respond to most events.

B.1

Condition B applies to all Containment Vent Isolation Functions and addresses the train orientation of the Solid State Protection System (SSPS) and the master and slave relays for these Functions. It also addresses the failure of multiple radiation monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1.

If a train is inoperable, multiple channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action for the applicable Conditions of LCO 3.6.3 is met for each valve made inoperable by failure of isolation instrumentation. A Note has been added above the Required Actions to allow one train of actuation logic to be placed in bypass and to delay entering the Required Actions for up to four hours to perform surveillance testing provided the other train is OPERABLE. The 4-hour allowance is consistent with the Required Actions for actuation logic trains in LCO 3.3.2, "Engineered Safety Features Actuation System

BASES

ACTIONS

B.1 (continued)

Instrumentation" and allows periodic testing to be conducted while at power without causing an actual actuation. The delay for entering the Required Actions relieves the administrative burden of entering the Required Actions for isolation valves inoperable solely due to the performance of surveillance testing on the actuation logic and is acceptable based on the OPERABILITY of the opposite train.

~~A Note is added stating that Condition B is only applicable in MODE 1, 2, 3, or 4.~~

~~C.1 and C.2~~

~~Condition C applies to all Containment Vent Isolation Functions and addresses the train orientation of the SSPS and the master and slave relays for these Functions. It also addresses the failure of multiple radiation monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1. If a train is inoperable, multiple channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action to place and maintain containment purge and exhaust isolation valves in their closed position is met or the applicable Conditions of LCO 3.9.4, "Containment Penetrations," are met for each valve made inoperable by failure of isolation instrumentation. The Completion Time for these Required Actions is Immediately.~~

~~A Note states that Condition C is only applicable during movement of irradiated fuel assemblies within containment.~~

SURVEILLANCE REQUIREMENTS

~~A Note has been added to the SR Table to clarify that Table 3.3.6-1 determines which SRs apply to which Containment Vent Isolation Functions.~~

SR 3.3.6.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value.

(continued)

BASES

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~~ 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 moving irradiated fuel in the Auxiliary Building if the containment and/or annulus air locks, penetrations, equipment hatch, etc.~~

BASES

~~are open to the Auxiliary Building ABSCE spaces.~~

BASES

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.~~

~~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.~~

(continued)

BASES

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.~~

~~Only the Allowable Value is specified for the Fuel Pool Area Radiation Monitors in the LCO. The Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis in order to account for instrument uncertainties appropriate to the trip function. The actual nominal Trip Setpoint is normally still more conservative than that required by the Allowable Value. If the measured setpoint does not exceed the Allowable Value, the radiation monitor is considered OPERABLE.~~

2. Containment Phase A Isolation

Refer to LCO 3.3.2, Function 3.a, for all initiating Functions and requirements.

(continued)

BASES

APPLICABILITY

The manual ABGTS initiation must be OPERABLE in MODES 1, 2, 3, and 4 ~~and when moving irradiated fuel assemblies in the fuel handling area~~ to ensure the ABGTS operates to remove fission products associated with leakage after a LOCA ~~or a fuel handling accident~~. The Phase A ABGTS Actuation is also required in MODES 1, 2, 3, and 4 to remove fission products caused by post LOCA Emergency Core Cooling Systems leakage.

~~High radiation initiation of the ABGTS must be OPERABLE in any MODE during movement of irradiated fuel assemblies in the fuel handling area to ensure automatic initiation of the ABGTS when the potential for a fuel handling accident exists.~~

While in MODES 5 and 6 ~~without fuel handling in progress~~, the ABGTS instrumentation need not be OPERABLE ~~since a fuel handling accident cannot occur.~~ See additional discussion in the Background and Applicable Safety Analysis sections.

ACTIONS

The most common cause of channel inoperability is outright failure or drift sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.8-1 in the accompanying LCO. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

(continued)

BASES

ACTIONS (continued)

A.1

Condition A applies to the actuation logic train function from the Phase A Isolation~~, the radiation monitor functions,~~ and the manual initiation function. Condition A applies to the failure of a single actuation logic train~~, radiation monitor channel,~~ or manual channel. If one channel or train is inoperable, a period of 7 days is allowed to restore it to OPERABLE status. If the train cannot be restored to OPERABLE status, one ABGTS train must be placed in operation. This accomplishes the actuation instrumentation function and places the unit in a conservative mode of operation. The 7-day Completion Time is the same as is allowed if one train of the mechanical portion of the system is inoperable. The basis for this time is the same as that provided in LCO 3.7.12.

B.1.1, B.1.2, B.2

Condition B applies to the failure of two ABGTS actuation logic signals from the Phase A Isolation~~, two radiation monitors,~~ or two manual channels. The Required Action is to place one ABGTS train in operation immediately. This accomplishes the actuation instrumentation function that may have been lost and places the unit in a conservative mode of operation. The applicable Conditions and Required Actions of LCO 3.7.12 must also be entered for the ABGTS train made inoperable by the inoperable actuation instrumentation. This ensures appropriate limits are placed on train inoperability as discussed in the Bases for LCO 3.7.12.

Alternatively, both trains may be placed in the emergency radiation protection mode. This ensures the ABGTS Function is performed even in the presence of a single failure.

~~C.1~~

~~Condition C applies when the Required Action and associated Completion Time for Condition A or B have not been met and irradiated fuel assemblies are being moved in the fuel building. Movement of irradiated fuel assemblies in the fuel building must be suspended immediately to eliminate the potential for events that could require ABGTS actuation. Performance of these actions shall not preclude moving a component to a safe position.~~

(continued)

BASES

ACTIONS (continued)

~~D.1 and D.2~~ C1 and C2

Condition ~~D~~ C applies when the Required Action and associated Completion Time for Condition A or B have not been met and the plant is in MODE 1, 2, 3, or 4. The plant must be brought to a MODE in which the LCO requirements are not applicable. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

~~A Note has been added to the SR Table to clarify that Table 3.3.8-1 determines which SRs apply to which ABGTS Actuation Functions.~~

~~SR 3.3.8.1~~

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

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

~~The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~SR 3.3.8.2~~

~~A COT is performed once every 92 days on each required channel to ensure the entire channel will perform the intended function. This test verifies the capability of the instrumentation to provide the ABGTS actuation. The Frequency of 92 days is based on the known reliability of the monitoring equipment and has been shown to be acceptable through operating experience. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as left and as found setting are consistent with those established by the setpoint methodology.~~

SR 3.3.8.~~3~~1

SR 3.3.8.~~3~~ 1 is the performance of a TADOT. This test is a check of the manual actuation functions and is performed every 18 months. Each manual actuation function is tested up to, and including, the relay coils. In some instances, the test includes actuation of the end device (e.g., pump starts, valve cycles, etc.). The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Functions tested have no setpoints associated with them.

~~SR 3.3.8.4~~

~~A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. The Frequency is based on operating experience and is consistent with the typical industry refueling cycle. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as left and as found setting are consistent with those established by the setpoint methodology.~~

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY. The containment satisfies Criterion 3 of the NRC Policy Statement.
LCO	Containment OPERABILITY is maintained by limiting leakage to ≤ 1.0 L_a, except prior to the first start up after performing a required Containment Leakage Rate Testing Program leakage test. At this time, applicable leakage limits must be met. Compliance with this LCO will ensure a containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis. Individual leakage rates specified for the containment air lock (LCO 3.6.2), purge valves with resilient seals, and Shield Building containment bypass leakage (LCO 3.6.3) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J, Option B. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the acceptance criteria of Appendix J, Option B.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 and 6 to prevent leakage of radioactive material from containment. ~~The requirements for containment during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."~~

APPLICABLE SAFETY ANALYSES

The DBAs that result in a significant release of radioactive material within containment are a loss of coolant accident and a rod ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate (L_a) of 0.25% of containment air weight per day (Ref. 2), at the calculated peak containment pressure of 15.0 psig. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

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

LCO

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

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into and exit from containment.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 and 6 to prevent leakage of radioactive material from containment. ~~The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."~~

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5 and 6. ~~The requirements for containment isolation valves during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."~~

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator (licensed or unlicensed) at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. For valve controls located in the control room, an operator (other than the Shift Operations Supervisor (SOS), ASOS, or the Operator at the Controls) may monitor containment isolation signal status rather than be stationed at the valve controls. Other secondary responsibilities which do not prevent adequate monitoring of containment isolation signal status may be performed by the operator provided his/her primary responsibility is rapid isolation of the penetration when needed for containment isolation. Use of the Unit Control Room Operator (CRO) to perform this function should be limited to those situations where no other operator is available.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are further modified by third Note, which ensures appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve.

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

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.3.7

Verifying that each 24 inch containment lower compartment purge valve is blocked to restrict opening to ≤ 50° is required to ensure that the valves can close under DBA conditions within the times assumed in the analyses of References 1 and 2. If a LOCA occurs, the purge valves must close to maintain containment leakage within the values assumed in the accident analysis. At other times when containment ~~when purge valves are required to be capable of closing (e.g., during movement of irradiated fuel assemblies),~~ pressurization concerns are not present, ~~thus~~ the purge valves can be fully open. The 18-month Frequency is appropriate because the blocking devices are typically removed only during a refueling outage.

SR 3.6.3.8

This SR ensures that the combined leakage rate of all Shield Building bypass leakage paths is less than or equal to the specified leakage rate. This provides assurance that the assumptions in the safety analysis are met. The as left bypass leakage rate prior to the first startup after performing a leakage test, requires calculation using maximum pathway leakage (leakage through the worse of the two isolation valves). If the penetration is isolated by use of one closed and de-activated automatic valve, closed manual valve, or blind flange, then the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves. At all other times, the leakage rate will be calculated using minimum pathway leakage.

The frequency is required by the Containment Leakage Rate Testing Program. This SR simply imposes additional acceptance criteria. Although not a part of L_a, the Shield Building Bypass leakage path combined leakage rate is determined using the 10 CFR 50, Appendix J, Option B, Type B and C leakage rates for the applicable barriers.

(continued)

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)

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 LOCA ~~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~~ 5).

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~~ 6) 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~~ Auxiliary 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

(continued)

BASES

LCO (continued)

c. Heater, moisture separator, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

APPLICABILITY

In MODE 1, 2, 3, or 4, the ABGTS is required to be OPERABLE to provide fission product removal associated with ECCS leaks due to a LOCA and leakage from containment and annulus.

In MODE 5 or 6, the ABGTS is not required to be OPERABLE since the ECCS is not required to be OPERABLE. ~~During movement of irradiated fuel in the fuel handling area, the ABGTS is required to be OPERABLE to alleviate the consequences of a fuel handling accident. See additional discussion in the Background and Applicable Safety Analysis sections.~~

ACTIONS

A.1

With one ABGTS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this period, the remaining OPERABLE train is adequate to perform the ABGTS function. The 7-day Completion Time is based on the risk from an event occurring requiring the inoperable ABGTS train, and the remaining ABGTS train providing the required protection.

B.1 and B.2

~~In MODE 1, 2, 3, or 4, when~~ When Required Action A.1 cannot be completed within the associated Completion Time, or when both ABGTS trains are inoperable, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in MODE 3 within 6 hours, and in MODE 5 within 36 hours. The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

~~C.1 and C.2~~

~~When Required Action A.1 cannot be completed within the required Completion Time, during movement of irradiated fuel assemblies in the fuel handling area, the OPERABLE ABGTS train must be started immediately or fuel movement suspended. This action ensures that the remaining train is OPERABLE, that no undetected failures preventing system operation will occur, and that any active failure will be readily detected.~~

~~If the system is not placed in operation, this action requires suspension of fuel movement, which precludes a fuel accident. This does not preclude the movement of fuel assemblies to a safe position.~~

~~D.1~~

~~When two trains of the ABGTS are inoperable during movement of irradiated fuel assemblies in the fuel handling area, action must be taken to place the unit in a condition in which the LCO does not apply. Action must be taken immediately to suspend movement of irradiated fuel assemblies in the fuel handling area. This does not preclude the movement of fuel to a safe position.~~

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

Standby systems should be checked periodically to ensure that they function properly. As the environmental and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system.

Monthly heater operation dries out any moisture accumulated in the charcoal from humidity in the ambient air. The system must be operated for ≥ 10 continuous hours with the heaters energized. The 31-day Frequency is based on the known reliability of the equipment and the two train redundancy available.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.2

This SR verifies that the required ABGTS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The ABGTS filter tests are in accordance with Regulatory Guide 1.52 (Ref. ~~8~~ 7). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.12.3

This SR verifies that each ABGTS train starts and operates on an actual or simulated actuation signal. The 18-month Frequency is consistent with Reference ~~8~~ 7.

SR 3.7.12.4

This SR verifies the integrity of the ABSCE. The ability of the ABSCE to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the ABGTS. During the post accident mode of operation, the ABGTS is designed to maintain a slight negative pressure in the ABSCE, to prevent unfiltered LEAKAGE. The ABGTS is designed to maintain a negative pressure between -0.25 inches water gauge and -0.5 inches water gauge (value does not account for instrument error) with respect to atmospheric pressure at a nominal flow rate $\geq$ 9300 cfm and $\leq$ 9900 cfm. The Frequency of 18 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. ~~9~~ 8).

An 18-month Frequency (on a STAGGERED TEST BASIS) is consistent with Reference ~~8~~ 7.

REFERENCES

1. Watts Bar FSAR, Section 6.5.1, “Engineered Safety Feature (ESF) Filter Systems.”
2. Watts Bar FSAR, Section 9.4.2, “Fuel Handling Area Ventilation System.”
3. Watts Bar FSAR, Section 15.0, “Accident Analysis.”

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REFERENCES (continued)

4. Watts Bar FSAR, Section 6.2.3, "Secondary Containment Functional Design."

~~5. Regulatory Guide 1.25, March 1972, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."~~

~~6~~ 5. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors."

~~7~~ 6. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

~~8~~ 7. Regulatory Guide 1.52 (Rev. 2), "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Power Plants."

~~9~~ 8. NUREG-0800, Section 6.5.1, "Standard Review Plan," Rev. 2, "ESF Atmosphere Cleanup System," July 1981.

~~10. NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," U. S. Nuclear Regulatory Commission, February 1988.~~

(continued)

BASES (continued)

REFERENCES

1. Watts Bar FSAR, Section 9.1.2, "Spent Fuel Storage."

2. Watts Bar FSAR, Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System."

3. Watts Bar FSAR, Section 15.5.6 ~~15.4.5~~, "Fuel Handling Accident."

4. ~~Regulatory Guide 1.25, March 1972, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."~~

5. ~~Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."~~

6. Regulatory Guide 1.183, "Alternate Source Terms for Evaluation Design Basis Accidents at Nuclear Power Reactors", July 2000.

7. Title 10, Code of Federal Regulations 50.67, "Accident Source Term."

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.13 Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the fuel storage pool design is given in the FSAR, Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the FSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the FSAR, Section ~~15.4.5~~ 15.5.6 (Ref. 3).

APPLICABLE SAFETY ANALYSES

The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide ~~1.25 (Ref. 4)~~ 1.183 Rev. 6. The Total effective Dose equivalent (TEDE) for control room occupants, individuals at the exclusion area boundary, and individuals within the low population zone will remain with 10 CFR 50.67 (Ref. 7) and Regulatory Position C.4.4 of Regulatory Guide 1.183 (Ref 6) for a fuel handling accident. The resultant 2 hour thyroid dose per person at the exclusion area boundary is a small fraction of the 10 CFR 100 (Ref. 5) limits.

According to Reference 3 4, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel handling accident. With 23 ft of water, the assumptions of Reference 6 4 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks; however, there may be < 23 ft of water above the top of the fuel bundle and the surface, indicated by the width of the bundle. To offset this small non-conservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rows fail from a hypothetical maximum drop.

The fuel storage pool water level satisfies Criterion 2 of the NRC Policy Statement.

(continued)

(continued)

BASES (continued)

LCO	The fuel storage pool water level is required to be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for fuel storage and movement within the fuel storage pool.
APPLICABILITY	This LCO applies during movement of irradiated fuel assemblies in the fuel storage pool, since the potential for a release of fission products exists.
ACTIONS	<u>A.1</u> Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the fuel storage pool water level is lower than the required level, the movement of irradiated fuel assemblies in the fuel storage pool is immediately suspended. This action effectively precludes the occurrence of a fuel handling accident. This does not preclude movement of a fuel assembly to a safe position. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.
SURVEILLANCE REQUIREMENTS	<u>SR 3.7.13.1</u> This SR verifies sufficient fuel storage pool water is available in the event of a fuel handling accident. The water level in the fuel storage pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by plant procedures and are acceptable based on operating experience. During refueling operations, the level in the fuel storage pool is in equilibrium with the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.7.1.

(continued)

B 3.9 REFUELING OPERATIONS

B 3.9.7 Refueling Cavity Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 2 and 8 ~~1 and 2~~). Sufficient iodine activity would be retained to limit offsite doses from the accident to ~~< 25% of 10 CFR 100 limits, as provided by the guidance of Reference 3~~ the limits defined in 10 CFR 50.67 (Ref. 7) and Regulatory Position C.4.4 of Regulatory Guide 1.183 (Ref. 8).

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment~~, as postulated by Regulatory Guide 1.25 (Ref. 1)~~. A minimum water level of 23 ft (Regulatory Position 2 of Appendix B to Regulatory Guide 1.183 (Ref. 8)) allows an overall iodine decontamination factor of 200 ~~C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1)~~ to be used in the accident analysis ~~for iodine~~. This relates to the assumption that 99.5% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 8% of the I-131, 10% of the Kr-85, and 5% of the other noble gases and iodines from the total fission product inventory in accordance with Regulatory Position 3.1 of Regulatory Guide 1.183 (Ref. 8). ~~10% of the total fuel rod iodine inventory (Ref. 1) except for I-131 which is assumed to be 12% (Ref. 6)~~.

The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft in conjunction with ~~and~~ a minimum decay time of 100 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Refs. 7 and 8 ~~4 and 5~~).

Refueling cavity water level satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO	A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits, as provided by the guidance of Reference 3.
APPLICABILITY	LCO 3.9.7 is applicable when moving irradiated fuel assemblies within containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.13, “Fuel Storage Pool Water Level.”
ACTIONS	<u>A.1</u> With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving movement of irradiated fuel assemblies within the containment shall be suspended immediately to ensure that a fuel handling accident cannot occur. The suspension of fuel movement shall not preclude completion of movement of a component to a safe position. <u>A.2</u> In addition to immediately suspending movement of irradiated fuel, actions to restore refueling cavity water level must be initiated immediately.
SURVEILLANCE REQUIREMENTS	<u>SR 3.9.7.1</u> Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2). The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.

(continued)

BASES (continued)

REFERENCES

1. ~~Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, March 23, 1972.~~

2. Watts Bar FSAR, Section 15.5.6 ~~15.4.5~~, "Fuel Handling Accident."

3. NUREG-0800, "Standard Review Plan," Section 15.7.4, "Radiological Consequences of Fuel-Handling Accidents," U.S. Nuclear Regulatory Commission.

4. Title 10, Code of Federal Regulations, Part 20.1201(a), (a)(1), and (2)(2), "Occupational Dose Limits for Adults."

~~5.~~ ~~Malinowski, D. D., Bell, M. J., Duhn, E., and Locante, J., WCAP-7828, Radiological Consequences of a Fuel Handling Accident, December 1971.~~

~~6.~~ ~~NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," U. S. Nuclear Regulatory Commission, February 1988.~~

7. Title 10, Code of Federal Regulations 50.67, "Accident Source Term."

8. Regulatory Guide 1.183, "Alternate Source Terms for Evaluation Design Basis Accidents at Nuclear Power Reactors", July 2000.

Enclosure 9

WBN Unit 2 – Revised Technical Specification Bases
Final

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."

APPLICABLE SAFETY ANALYSES

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. 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.

The Containment Vent Isolation instrumentation satisfies Criterion 3 of the NRC Policy Statement.

(continued)

LCO

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

5. 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.

6. 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)

LCO
(continued)

7. Containment Radiation

The LCO specifies two required channels of radiation monitors to ensure that the radiation monitoring instrumentation necessary to initiate Containment Vent Isolation remains OPERABLE.

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

Only the Allowable Value is specified for the Containment Purge Exhaust Radiation Monitors in the LCO. The Allowable Value is based on expected concentrations for a small break LOCA, which is more restrictive than 10 CFR 100 limits. The Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis in order to account for instrument uncertainties appropriate to the trip function. The actual nominal Trip Setpoint is normally still more conservative than that required by the Allowable Value. If the setpoint does not exceed the Allowable Value, the radiation monitor is considered OPERABLE.

8. Safety Injection (SI)

Refer to LCO 3.3.2, Function 1, for all initiating Functions and requirements.

APPLICABILITY

The Manual Initiation, Automatic Actuation Logic and Actuation Relays, Safety Injection, and Containment Radiation Functions are required OPERABLE in MODES 1, 2, 3, and 4. Under these conditions, the potential exists for an accident that could release significant fission product radioactivity into containment. Therefore, the Containment Vent Isolation Instrumentation must be OPERABLE in these MODES. See additional discussion in the Background and Applicable Safety Analysis sections.

While in MODES 5 and 6, the Containment Vent Isolation Instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within the limits of Reference 1.

(continued)

BASES (continued)

ACTIONS

The most common cause of channel inoperability is outright failure or drift sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately, and the appropriate Condition entered.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.6-1. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies to the failure of one containment purge isolation radiation monitor channel. Since the two containment radiation monitors are both gaseous detectors, failure of a single channel may result in loss of the redundancy. Consequently, the failed channel must be restored to OPERABLE status. The 4 hours allowed to restore the affected channel is justified by the low likelihood of events occurring during this interval, and recognition that one or more of the remaining channels will respond to most events.

B.1

Condition B applies to all Containment Vent Isolation Functions and addresses the train orientation of the Solid State Protection System (SSPS) and the master and slave relays for these Functions. It also addresses the failure of multiple radiation monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1.

If a train is inoperable, multiple channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action for the applicable Conditions of LCO 3.6.3 is met for each valve made inoperable by failure of isolation instrumentation. A Note has been added above the Required Actions to allow one train of actuation logic to be placed in bypass and to delay entering the Required Actions for up to four hours to perform surveillance testing provided the other train is OPERABLE. The 4-hour allowance is consistent with the Required Actions for actuation logic trains in LCO 3.3.2, "Engineered Safety Features Actuation System

(continued)

BASES

ACTIONS

B.1 (continued)

Instrumentation" and allows periodic testing to be conducted while at power without causing an actual actuation. The delay for entering the Required Actions relieves the administrative burden of entering the Required Actions for isolation valves inoperable solely due to the performance of surveillance testing on the actuation logic and is acceptable based on the OPERABILITY of the opposite train.

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value.

(continued)

BASES

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 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.

There are a total of two channels, one for each train. 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).

APPLICABLE SAFETY ANALYSES

The ABGTS ensures that radioactive materials in the ABSCE atmosphere following 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.

(continued)

BASES

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. Containment Phase A Isolation

   Refer to LCO 3.3.2, Function 3.a, for all initiating Functions and requirements.

(continued)

BASES

APPLICABILITY

The manual ABGTS initiation must be OPERABLE in MODES 1, 2, 3, and 4 to ensure the ABGTS operates to remove fission products associated with leakage after a LOCA. The Phase A ABGTS Actuation is also required in MODES 1, 2, 3, and 4 to remove fission products caused by post LOCA Emergency Core Cooling Systems leakage.

While in MODES 5 and 6, the ABGTS instrumentation need not be OPERABLE. See additional discussion in the Background and Applicable Safety Analysis sections.

ACTIONS

The most common cause of channel inoperability is outright failure or drift sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.8-1 in the accompanying LCO. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies to the actuation logic train function from the Phase A Isolation and the manual initiation function. Condition A applies to the failure of a single actuation logic train or manual channel. If one channel or train is inoperable, a period of 7 days is allowed to restore it to OPERABLE status. If the train cannot be restored to OPERABLE status, one ABGTS train must be placed in operation. This accomplishes the actuation instrumentation function and places the unit in a conservative mode of operation. The 7-day Completion Time is the same as is allowed if one train of the mechanical portion of the system is inoperable. The basis for this time is the same as that provided in LCO 3.7.12.

(continued)

BASES

ACTIONS (continued)

B.1.1, B.1.2, B.2

Condition B applies to the failure of two ABGTS actuation logic signals from the Phase A Isolation or two manual channels. The Required Action is to place one ABGTS train in operation immediately. This accomplishes the actuation instrumentation function that may have been lost and places the unit in a conservative mode of operation. The applicable Conditions and Required Actions of LCO 3.7.12 must also be entered for the ABGTS train made inoperable by the inoperable actuation instrumentation. This ensures appropriate limits are placed on train inoperability as discussed in the Bases for LCO 3.7.12.

Alternatively, both trains may be placed in the emergency radiation protection mode. This ensures the ABGTS Function is performed even in the presence of a single failure.

C1 and C2

Condition C applies when the Required Action and associated Completion Time for Condition A or B have not been met and the plant is in MODE 1, 2, 3, or 4. The plant must be brought to a MODE in which the LCO requirements are not applicable. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1

SR 3.3.8.1 is the performance of a TADOT. This test is a check of the manual actuation functions and is performed every 18 months. Each manual actuation function is tested up to, and including, the relay coils. In some instances, the test includes actuation of the end device (e.g., pump starts, valve cycles, etc.). The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Functions tested have no setpoints associated with them.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of the NRC Policy Statement.

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq 1.0\ L_a$, except prior to the first start up after performing a required Containment Leakage Rate Testing Program leakage test. At this time, applicable leakage limits must be met.

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

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

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 and 6 to prevent leakage of radioactive material from containment.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The DBAs that result in a significant release of radioactive material within containment are a loss of coolant accident and a rod ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate (L_a) of 0.25% of containment air weight per day (Ref. 2), at the calculated peak containment pressure of 15.0 psig. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

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

LCO

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

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into and exit from containment.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 and 6 to prevent leakage of radioactive material from containment.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5 and 6.

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator (licensed or unlicensed) at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. For valve controls located in the control room, an operator (other than the Shift Operations Supervisor (SOS), ASOS, or the Operator at the Controls) may monitor containment isolation signal status rather than be stationed at the valve controls. Other secondary responsibilities which do not prevent adequate monitoring of containment isolation signal status may be performed by the operator provided his/her primary responsibility is rapid isolation of the penetration when needed for containment isolation. Use of the Unit Control Room Operator (CRO) to perform this function should be limited to those situations where no other operator is available.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are further modified by third Note, which ensures appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve.

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.7

Verifying that each 24 inch containment lower compartment purge valve is blocked to restrict opening to ≤ 50° is required to ensure that the valves can close under DBA conditions within the times assumed in the analyses of References 1 and 2. If a LOCA occurs, the purge valves must close to maintain containment leakage within the values assumed in the accident analysis. At other times when containment pressurization concerns are not present, the purge valves can be fully open. The 18-month Frequency is appropriate because the blocking devices are typically removed only during a refueling outage.

SR 3.6.3.8

This SR ensures that the combined leakage rate of all Shield Building bypass leakage paths is less than or equal to the specified leakage rate. This provides assurance that the assumptions in the safety analysis are met. The as left bypass leakage rate prior to the first startup after performing a leakage test, requires calculation using maximum pathway leakage (leakage through the worse of the two isolation valves). If the penetration is isolated by use of one closed and de-activated automatic valve, closed manual valve, or blind flange, then the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves. At all other times, the leakage rate will be calculated using minimum pathway leakage.

The frequency is required by the Containment Leakage Rate Testing Program. This SR simply imposes additional acceptance criteria. Although not a part of L_a, the Shield Building Bypass leakage path combined leakage rate is determined using the 10 CFR 50, Appendix J, Option B, Type B and C leakage rates for the applicable barriers.

(continued)

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 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.

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 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).

(continued)

APPLICABLE SAFETY ANALYSES

The ABGTS design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a LOCA. 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 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 LOCA. The assumptions and analysis for a LOCA follow the guidance provided in Regulatory Guide 1.4 (Ref. 5).

The ABGTS satisfies Criterion 3 of the NRC Policy Statement.

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. 6) limits in the event of a LOCA.

The ABGTS is considered OPERABLE when the individual components necessary to control exposure in the Auxiliary 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

c. Heater, moisture separator, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

(continued)

BASES

LCO
(continued)

APPLICABILITY

In MODE 1, 2, 3, or 4, the ABGTS is required to be OPERABLE to provide fission product removal associated with ECCS leaks due to a LOCA and leakage from containment and annulus.

In MODE 5 or 6, the ABGTS is not required to be OPERABLE since the ECCS is not required to be OPERABLE.

ACTIONS

A.1

With one ABGTS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this period, the remaining OPERABLE train is adequate to perform the ABGTS function. The 7-day Completion Time is based on the risk from an event occurring requiring the inoperable ABGTS train, and the remaining ABGTS train providing the required protection.

B.1 and B.2

When Required Action A.1 cannot be completed within the associated Completion Time, or when both ABGTS trains are inoperable, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in MODE 3 within 6 hours, and in MODE 5 within 36 hours. The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS
(continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

Standby systems should be checked periodically to ensure that they function properly. As the environmental and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system.

Monthly heater operation dries out any moisture accumulated in the charcoal from humidity in the ambient air. The system must be operated for ≥ 10 continuous hours with the heaters energized. The 31-day Frequency is based on the known reliability of the equipment and the two train redundancy available.

SR 3.7.12.2

This SR verifies that the required ABGTS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The ABGTS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 7). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.12.3

This SR verifies that each ABGTS train starts and operates on an actual or simulated actuation signal. The 18-month Frequency is consistent with Reference 7.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.4

This SR verifies the integrity of the ABSCE. The ability of the ABSCE to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the ABGTS. During the post accident mode of operation, the ABGTS is designed to maintain a slight negative pressure in the ABSCE, to prevent unfiltered LEAKAGE. The ABGTS is designed to maintain a negative pressure between -0.25 inches water gauge and -0.5 inches water gauge (value does not account for instrument error) with respect to atmospheric pressure at a nominal flow rate ≥ 9300 cfm and ≤ 9900 cfm. The Frequency of 18 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 8).

An 18-month Frequency (on a STAGGERED TEST BASIS) is consistent with Reference 7.

REFERENCES

1. Watts Bar FSAR, Section 6.5.1, "Engineered Safety Feature (ESF) Filter Systems."
2. Watts Bar FSAR, Section 9.4.2, "Fuel Handling Area Ventilation System."
3. Watts Bar FSAR, Section 15.0, "Accident Analysis."
4. Watts Bar FSAR, Section 6.2.3, "Secondary Containment Functional Design."
5. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors."
6. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."
7. Regulatory Guide 1.52 (Rev. 2), "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Power Plants."
8. NUREG-0800, Section 6.5.1, "Standard Review Plan," Rev. 2, "ESF Atmosphere Cleanup System," July 1981.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.13 Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the fuel storage pool design is given in the FSAR, Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the FSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the FSAR, Section 15.5.6 (Ref. 3).

APPLICABLE SAFETY ANALYSES

The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.183 Rev. 6. The Total effective Dose equivalent (TEDE) for control room occupants, individuals at the exclusion area boundary, and individuals within the low population zone will remain with 10 CFR 50.67 (Ref. 7) and Regulatory Position C.4.4 of Regulatory Guide 1.183 (Ref 6) for a fuel handling accident.

According to Reference 3, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel handling accident. With 23 ft of water, the assumptions of Reference 6 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks; however, there may be < 23 ft of water above the top of the fuel bundle and the surface, indicated by the width of the bundle. To offset this small non-conservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rows fail from a hypothetical maximum drop.

The fuel storage pool water level satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

The fuel storage pool water level is required to be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for fuel storage and movement within the fuel storage pool.

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the fuel storage pool, since the potential for a release of fission products exists.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the fuel storage pool water level is lower than the required level, the movement of irradiated fuel assemblies in the fuel storage pool is immediately suspended. This action effectively precludes the occurrence of a fuel handling accident. This does not preclude movement of a fuel assembly to a safe position.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

This SR verifies sufficient fuel storage pool water is available in the event of a fuel handling accident. The water level in the fuel storage pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by plant procedures and are acceptable based on operating experience.

During refueling operations, the level in the fuel storage pool is in equilibrium with the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.7.1.

(continued)

BASES (continued)

REFERENCES

1. Watts Bar FSAR, Section 9.1.2, "Spent Fuel Storage."
2. Watts Bar FSAR, Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System."
3. Watts Bar FSAR, Section 15.5.6, "Fuel Handling Accident."
4. Deleted
5. Deleted
6. Regulatory Guide 1.183, "Alternate Source Terms for Evaluation Design Basis Accidents at Nuclear Power Reactors", July 2000.
7. Title 10, Code of Federal Regulations 50.67, "Accident Source Term."

(continued)

B 3.9 REFUELING OPERATIONS

B 3.9.7 Refueling Cavity Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 2 and 8). Sufficient iodine activity would be retained to limit offsite doses from the accident to the limits defined in 10 CFR 50.67 (Ref. 7) and Regulatory Position C.4.4 of Regulatory Guide 1.183 (Ref. 8).

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment. A minimum water level of 23 ft (Regulatory Position 2 of Appendix B to Regulatory Guide 1.183 (Ref. 8)) allows an overall iodine decontamination factor of 200 to be used in the accident analysis. This relates to the assumption that 99.5% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 8% of the I-131, 10% of the Kr-85, and 5% of the other noble gases and iodines from the total fission product inventory in accordance with Regulatory Position 3.1 of Regulatory Guide 1.183 (Ref. 8).

The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft in conjunction with a minimum decay time of 100 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Refs. 7 and 8).

Refueling cavity water level satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO	A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits, as provided by the guidance of Reference 3.
APPLICABILITY	LCO 3.9.7 is applicable when moving irradiated fuel assemblies within containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.13, "Fuel Storage Pool Water Level."
ACTIONS	<u>A.1</u> With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving movement of irradiated fuel assemblies within the containment shall be suspended immediately to ensure that a fuel handling accident cannot occur. The suspension of fuel movement shall not preclude completion of movement of a component to a safe position. <u>A.2</u> In addition to immediately suspending movement of irradiated fuel, actions to restore refueling cavity water level must be initiated immediately.
SURVEILLANCE REQUIREMENTS	<u>SR 3.9.7.1</u> Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2). The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.

(continued)

BASES (continued)

REFERENCES

1. Deleted

2. Watts Bar FSAR, Section 15.5.6, "Fuel Handling Accident."

3. NUREG-0800, "Standard Review Plan," Section 15.7.4, "Radiological Consequences of Fuel-Handling Accidents," U.S. Nuclear Regulatory Commission.

4. Title 10, Code of Federal Regulations, Part 20.1201(a), (a)(1), and (2)(2), "Occupational Dose Limits for Adults."

5. Deleted

6. Deleted

7. Title 10, Code of Federal Regulations 50.67, "Accident Source Term."

8. Regulatory Guide 1.183, "Alternate Source Terms for Evaluation Design Basis Accidents at Nuclear Power Reactors", July 2000.

(continued)

Enclosure 10
WBN Unit 2 – Revised Technical Requirements Manual Section 3.9.1

~~TR 3.9 REFUELING OPERATIONS~~

~~TR 3.9.1 Decay Time~~

~~TR 3.9.1~~ ~~The reactor shall be subcritical for ≥ 100 hours.~~

~~APPLICABILITY:~~ ~~During movement of irradiated fuel in the reactor vessel.~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. Reactor subcritical for < 100 hours.~~	~~A.1 Suspend all operations involving movement of irradiated fuel in the reactor vessel.~~	~~Immediately~~

~~TECHNICAL SURVEILLANCE REQUIREMENTS~~

	~~SURVEILLANCE~~	~~FREQUENCY~~
~~TSR 3.9.1.1~~	~~Verify the reactor has been subcritical for ≥ 100 hours by confirming the date and time of subcriticality.~~	~~Prior to movement of irradiated fuel in the reactor vessel~~