Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

June 13, 2012

10 CFR 50.90

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Watts Bar Nuclear Plant, Unit 1
Facility Operating License No. NPF-90
NRC Docket No. 50-390

Subject: **Watts Bar Nuclear Plant Unit 1 - Application to Allow Selective Implementation of Alternate Source Term to Analyze the Dose Consequences Associated with Fuel Handling Accidents (WBN-TS-11-19)**

In accordance with the provisions of 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," and 10 CFR 50.67, "Accident source term," the Tennessee Valley Authority (TVA) requests a change (WBN-TS-11-19) to Watts Bar Nuclear Plant (WBN), Unit 1, Facility Operating License No. NPF-90.

The proposed amendment will:

1. Permit selective implementation of the Alternate Source Term (AST) methodology in accordance with 10 CFR 50.67 and Regulatory Guide (RG) 1.183, "Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." Implementation of the AST methodology will be limited to the analysis of Fuel Handling Accidents (FHAs) for WBN, Unit 1;

2. Add WBN, Unit 1 Technical Specification (TS) 3.9.10, "Decay Time," 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;

3. Modify WBN, Unit 1 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 in the containment or the fuel handling area;

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4. Eliminate TS 3.9.4, "Containment Penetrations," and TS 3.9.8, "Reactor Building Purge Air Cleanup Units;" and

5. Modify WBN, Unit 1 TS 5.7.2.20, "Control Room Envelope Habitability Program," to incorporate the Control Room dose limit defined in 10 CFR 50.67(b)(2)(iii).

Enclosure 1 to this letter provides a description, technical evaluation, regulatory evaluation, and discussion of environmental considerations of the proposed changes. Attachments 1 and 2 to the enclosure provide the existing TS and TS Bases pages marked-up to show the proposed changes. Attachments 3 and 4 to the enclosure provide the existing TS and TS Bases pages retyped to show the proposed changes. The TS Bases pages are provided to the NRC for information only.

Enclosure 2 to this letter provides the WBN, Unit 1 calculation utilized to determine the radiological consequences associated with a FHA.

Enclosure 3 provides the data files for the meteorological data that supports calculation of the atmospheric dispersion factors.

TVA requests approval of the proposed license amendment by June 13, 2013, with implementation within 60 days of issuance.

The WBN Plant Operations Review Committee and the WBN Nuclear Safety Review Board have reviewed this proposed change and determined that operation of WBN in accordance with the proposed change will not endanger the health and safety of the public.

Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and Enclosures to the Tennessee Department of Environment and Conservation.

This submittal does not contain any new regulatory commitments. Please address any questions regarding this request to Terry Cribbe, Corporate Licensing Manager, at 423-751-3850.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 13 day of June 2012.

Respectfully,

J. W. Shea
Manager, Corporate Nuclear Licensing

Enclosures:

1. Evaluation of Proposed Change
2. Calculation WBNTSR-009, Control Room Operator and Offsite Doses from a Fuel Handling Accident, Revision 14
3. Compact Disc with Meteorological Data File for Calendar Years 1991 – 2010 and the ARCON96 Output File

cc (Enclosures 1 and 2):

NRC Regional Administrator - Region II
NRC Resident Inspector – Watts Bar Nuclear Plant, Unit 1
NRC Resident Inspector – Watts Bar Nuclear Plant, Unit 2
Director, Division of Radiological Health - Tennessee State Department of Environment and Conservation

ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT, UNIT 1
EVALUATION OF PROPOSED CHANGE

Subject: **Application to Allow Selective Implementation of Alternate Source Term to Analyze the Dose Consequences Associated with Fuel Handling Accidents (WBN-TS-11-19)**

ATTACHMENTS

1.0 SUMMARY DESCRIPTION

The Tennessee Valley Authority (TVA) is proposing to amend Watts Bar Nuclear Plant (WBN), Unit 1, Facility Operating License No. NPF-90. The proposed change will selectively implement an Alternate Source Term (AST) methodology in accordance with Regulatory Position C.1.2.2 of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," by modifying the WBN, Unit 1 licensing basis for determining offsite and Control Room doses due to a Fuel Handling Accident (FHA). A license amendment is required for AST implementation in accordance with 10 CFR 50.67(b)(1) which states:

> "A licensee who seeks to revise its current accident source term in design basis radiological consequence analyses shall apply for a license amendment under Sec. 50.90. The application shall contain an evaluation of the consequences of applicable design basis accidents previously analyzed in the safety analysis report."

The proposed amendment will:

1. Add WBN, Unit 1 Technical Specification (TS) 3.9.10, "Decay Time," 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;

2. Modify WBN, Unit 1 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 in the containment or the fuel handling area;

3. Eliminate TS 3.9.4, "Containment Penetrations," and TS 3.9.8, "Reactor Building Purge Air Cleanup Units;" and

4. Modify WBN, Unit 1 TS 5.7.2.20, "Control Room Envelope Habitability Program," to incorporate the Control Room dose limit defined in 10 CFR 50.67(b)(2)(iii).

Mark-ups of the affected TS and TS Bases pages are included in Attachments 1 and 2. Final typed versions of the affected TS and TS Bases pages are included in Attachments 3 and 4. The TS Bases pages are provided to the NRC for information only.

2.0 DETAILED DESCRIPTION

2.1 Proposed Changes

The proposed change to the WBN, Unit 1 licensing basis involves the adoption of the AST methodology for calculating accident doses to Control Room personnel and offsite receptors following a FHA in accordance with 10 CFR 50.67 and RG 1.183. The following changes to the WBN, Unit 1 Technical Specifications are requested to support adoption of the new AST analysis of the FHA.

TS 3.3.6, "Containment Vent Isolation Instrumentation"

TVA proposes the following changes to TS 3.3.6:

1. Elimination of the specified condition "during movement of irradiated fuel assemblies within containment" from the Applicability Section;

2. Elimination of the Note from Condition B that states that it is only applicable in MODE 1, 2, 3, or 4;

3. Elimination of Condition C and the associated Required Actions;

4. Elimination of footnotes (a) and (b) from Table 3.3.6-1; and

5. Elimination of Allowable Values for the Containment Purge Exhaust Radiation Monitors that apply during movement of irradiated fuel assemblies within containment.

These changes reflect that: 1) the AST analysis of the FHA does not credit containment isolation, and 2) TS 3.9.10 is added 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.

TS 3.3.8, "Auxiliary Building Gas Treatment System (ABGTS) Actuation Instrumentation"

TVA proposes the following changes to TS 3.3.8:

1. Elimination of Condition C and the associated Required Actions;

2. Elimination of the reference to MODE 1, 2, 3, or 4 in Condition D;

3. Renumber Condition D, Required Action D.1, and Required Action D.2 as Condition C, Required Action C.1, and Required Action C.2;

4. Elimination of Surveillance Requirements (SRs) 3.3.8.1, 3.3.8.2, and 3.3.8.4;

5. Renumbering SR 3.3.8.3 as SR 3.3.8.1 in the Surveillance Requirements Section and Table 3.3.8-1;

6. Elimination of footnote (a) from Table 3.3.8-1; and

7. Elimination of the requirements regarding the Fuel Pool Area Radiation Monitors from Table 3.3.8-1.

These changes reflect that: 1) the AST analysis of the FHA does not credit actuation of the Auxiliary Building Gas Treatment Gas (ABGTS), and 2) TS 3.9.10 is added 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.

TS 3.7.12, "Auxiliary Building Gas Treatment System (ABGTS)"

TVA proposes the following changes to TS 3.7.12:

1. Elimination of the specified condition "during movement of irradiated fuel assemblies in the fuel handling area" from the Applicability Section;

2. Elimination of the references to MODE 1, 2, 3, or 4 in Condition B; and

3. Elimination of Conditions C and D and their associated Required Actions.

These changes reflect: 1) the AST analysis of the FHA does not credit actuation of the ABGTS, and 2) TS 3.9.10 is added 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.

TS 3.9.4, "Containment Penetrations"

TVA proposes to eliminate TS 3.9.4 in its entirety is proposed. This change reflects that: 1) the AST analysis of the FHA does not credit containment isolation, and 2) TS 3.9.10, is added 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.

TS 3.9.8, "Reactor Building Purge Air Cleanup Units"

TVA proposes to eliminate TS 3.9.8 in its entirety. This change reflects that: 1) the AST analysis of the FHA does not credit containment isolation, and 2) TS 3.9.10 is added 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.

TS 3.9.10, "Decay Time"

TVA proposes to add TS 3.9.10. TS 3.9.10 is added 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.

TS 5.7.2.20, "Control Room Envelope Habitability Program

TS 5.7.2.20 currently states:

> "...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 5 rem whole body or its equivalent to any part of the body for the duration of the accident..."

TS 5.7.2.20 would be revised to state:

> "...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 other accidents) for the duration of the accident..."

<u>Bases for Technical Specifications 3.3.6, 3.3.8, 3.7.12, 3.7.13, 3.9.4, 3.9.7, 3.9.8, and 3.9.10</u>

Conforming changes to the Bases for TSs 3.3.6, 3.3.8, 3.7.12, 3.9.4, and 3.9.8 are made to address the selective implementation of the AST for the FHA analysis. In addition, Bases for TS 3.9.10 are added, and changes to the Bases for TSs 3.7.13 and 3.9.7 are made to denote that 10 CFR 50.67 and Regulatory Position C.4.4 of RG 1.183 define the regulatory dose limits for the fuel handling accident.

2.2 Need for Proposed Changes

Section 15.5.6 of the WBN, Unit 1 Updated Final Safety Analysis Report (UFSAR) provides the current WBN, Unit 1 licensing basis for the radiological analyses of the FHA. The analysis is based on RG 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," and NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors." The dose was determined by utilizing dose equations from TID-14844. Dose conversion factors in International Commission on Radiation Protection (ICRP) Publication 30 were used to determine thyroid doses in place of those found in TID-14844.

On September 14, 2010, TVA discovered a conflict between the dose calculation for the FHA analysis and the calculation that determines the response time for the radiation monitor system, including the time for closure of the isolation damper. The dose calculation assumes that automatic isolation of the Auxiliary Building Ventilation System occurs, such that no unfiltered releases occur subsequent to a FHA. The response time calculation determined that the total response time for isolation of the Auxiliary Building Ventilation System was 12.5 seconds and the gas travel time between the spent fuel pool monitor and the Auxiliary Building Secondary Containment Enclosure (ABSCE) isolation damper was 6.3 seconds. Therefore, the potential exists for an unfiltered release to occur for 6.2 seconds following a FHA in the Auxiliary Building. Problem Evaluation Report (PER) 252012 was initiated to document the issue.

A functional evaluation was performed in accordance with NRC Inspection Manual, Part 9900: "Technical Guidance." The functional evaluation utilized an alternative analytical method in accordance with Appendix C.4 of NRC Inspection Manual, Part 9900 to establish that the offsite and Control Room doses would be within the regulatory limits without isolation of the Auxiliary Building following a FHA. The alternative method utilized the AST methodology.

10 CFR 50.67 was issued to allow holders of operating licenses to voluntarily revise the traditional accident source terms used in the design basis accident (DBA) radiological

consequence analyses with ASTs, because of advances made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents. RG 1.183 and NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," provide guidance on selective application of the AST methodology in revising the accident source terms used in design basis radiological consequences analyses, as allowed by 10 CFR 50.67.

The proposed changes to the WBN, Unit 1 licensing basis and TS are required to resolve the existing plant condition identified in PER 252012. The analysis of the dose consequences resulting from a FHA demonstrates that the regulatory acceptance criteria are met. NRC approval of the requested changes will establish the acceptability of the use of the AST methodology for WBN, Unit 1 to analyze the dose consequences associated with a FHA.

The proposed changes to WBN, Unit 1 TS 3.3.6, TS 3.3.8, TS 3.7.12, TS 3.9.4, TS 3.9.8, and TS 5.7.2.20 and the addition of TS 3.9.10 are made to reflect the assumptions made in the revised FHA analyses. The following FHA scenarios are addressed:

1. The drop of a single fuel assembly in the spent fuel pool/Auxiliary Building with no Auxiliary Building Isolation (ABI) and with unfiltered releases through the Auxiliary Building vent.

2. The drop of a single fuel assembly in the containment. The containment is assumed to be open, and an unfiltered release occurs through the Shield Building vent for 12.7 seconds until the Reactor Building Purge Ventilating System (RBPVS) is isolated. After 12.7 seconds, the remaining release occurs through the Auxiliary Building vent with no ABI and no filtration.

Both analyses assume that the fuel decays for 100 hours prior to the FHA occurring, Control Room isolation occurs within 40 seconds, and the Control Room Emergency Ventilation System (CREVS) filters the air provided to the Control Room.

The Case # 1 analysis of the FHA in the Auxiliary Building is the bounding accident, because it results in a higher dose to the occupants of the Control Room. The doses to the receptors at the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) are the same for both FHA analyses.

3.0 TECHNICAL EVALUATION

3.1 Introduction

3.1.1 Control Room Area Ventilation System Description

Section 9.4.1 of the WBN, Unit 1 UFSAR provides a description of the Control Room Area Ventilation System.

The Control Building air-conditioning systems are engineered safety features. Each pair of full-capacity (one redundant) water chillers and each redundant set of air handling units is served from a separate train of the emergency power system and from a coordinated separate loop of the Essential Raw Cooling Water System (ERCW).

The Control Building outside air intakes are provided with radiation monitors, and smoke detectors. Indicators are provided with the radiation monitors. Control room common annunciation is provided. Isolation of the Main Control Room Habitability Zone (MCRHZ) occurs automatically upon the actuation of a safety injection signal or upon indication of high radiation, or smoke concentrations in the outside air supply stream to the building.

Upon receipt of a signal for Main Control Room Habitability System (MCRHS) area isolation, Control Room Isolation (CRI), the following conditions are automatically implemented:

1. The Control Building emergency air cleanup fans operate to recirculate a portion of the MCRHS area air-conditioning system return air through the cleanup trains composed of HEPA filters and charcoal adsorbers.

2. The Control Building emergency pressurizing air supply fan operates to supply a reduced stream of outside air to the Control Room air-conditioning system to maintain the MCRHZ pressurized relative to outside and the adjacent areas. This fresh air is routed through the emergency air cleanup trains.

3. The Control Room electrical board rooms air handling units continue to draw outside air to maintain the lower floor spaces at atmospheric pressure.

4. The exhaust fan in the toilet rooms is stopped, and double isolation dampers are closed.

5. The spreading room supply and exhaust fans are stopped and any operating battery room exhaust fan continues to run.

6. Double isolation dampers in the spreading room supply duct and isolation dampers in the exhaust duct close.

7. The Auxiliary Building Elevation 757 shutdown board rooms pressurizing air supply fans are automatically de-energized.

8. Double isolation valves close to isolate the normal pressurizing supply to the MCRHZ.

MCRHZ isolation may be accomplished manually at any time by the Control Room operators.

The following building air-conditioning and ventilating system components are each provided with two 100% capacity units. Each meets the single failure criterion, and automatic switchover is assured if one of the units fails. These systems include the:

1. Control room air-conditioning system, water chillers, air handling units, and piping.

2. Control Building emergency air cleanup supply fans and filter assemblies.

3. Control Building emergency pressurizing air supply fans.

3.1.2 Fuel Handling Area Ventilation System Description

Section 9.4.2 of the WBN, Unit 1 UFSAR describes the Fuel Handling Area Ventilation System. It is a subsystem of the Auxiliary Building Ventilation System.

A FHA in the Auxiliary Building is detected by two gamma radiation monitors, mounted above the spent fuel pool. The high radiation signals via redundant trains will shut off the fuel handling and Auxiliary Building general supply and exhaust fans and start the ABGTS. To accomplish its safety function following a FHA, the fuel handling area ventilation system must:

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 (safety-related) located above the spent fuel pool 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 or the annulus is open to the ABSCE spaces, a Containment Vent Isolation (CVI) signal is procedurally configured to assure that a FHA in containment is promptly detected and the CIV signal is provided to each ventilation train.

3.1.3 Reactor Building Purge Ventilating System

Section 9.4.6 of the WBN, Unit 1 UFSAR describes the Reactor Building Purge Ventilating System (RBPVS).

The RBPVS consists of two trains, each designed to provide 50% of the capacity required for normal operation. Each train contains an air supply fan, an air exhaust fan, a cleanup filter unit, containment isolation valves, system air flow control valves, and all necessary ductwork. The system also includes single air supply distribution and air exhaust collection subsystems as well as an instrument room supply fan and an instrument room exhaust fan.

The filtered air is discharged to the outdoors by means of the Shield Building exhaust vent located in the annular space of the Reactor Building and extending through the roof of the Reactor Building. The purge air filtration units and associated exhaust ductwork provide a safety-related filtration path following a FHA.

Three signals cause the system to change from the normal purge mode to the accident isolation mode. These signals (i.e., manual, safety injection system auto-initiate, and high purge exhaust radiation (automatic)) initiate a CVI signal. Additionally, during refueling operations whenever containment or the annulus is open to the ABSCE spaces, a high radiation signal from the spent fuel pool accident radiation monitors automatically cause the system to change from the purge mode to the accident isolation mode.

3.1.4 Current Licensing Basis for the FHA

The current WBN, Unit 1 radiological analyses of the FHA is based on RG 1.25 and NUREG/CR-5009. The dose was determined by utilizing dose equations from TID-14844. Dose conversion factors in ICRP-30 were used to determine thyroid doses in place of those found in TID-14844.

Major assumptions of the current analysis include: 1) the accident occurs 100 hours after plant shutdown; 2) all of the fuel rods in one fuel assembly are damaged; 3) the 24 Tritium Producing Burnable Absorber Rods (TPBARs) in a single fuel assembly are damaged and release their entire tritium content to the environment; and 4) the release is filtered by the RBPVS or ABGTS filters.

3.1.5 Proposed Changes to the WBN, Unit 1 Current Licensing Basis

TVA proposes to revise the WBN, Unit 1 licensing basis to selectively implement the AST described in RG 1.183 through reanalysis of the radiological consequences of the FHA. 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) and Regulatory Position C.4.4 of RG 1.183 replace the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR 50, Appendix A, General Design Criterion (GDC) 19.
- The gap activity is revised to be consistent with the guidance of RG 1.183.
- An overall decontamination factor of 200 was applied to iodines in accordance with the guidance of RG. 1.183.
- The release of radioactive materials, including tritium, is assumed to be linear over a two-hour time frame to be consistent with the guidance of RG 1.183.
- Once or twice burned fuel assemblies may contain up to 24 TPBARs. All 24 TPBARs in a fuel assembly are assumed to break, with 25% of the tritium inventory being released to the environment.
- No filtration of the release by the RBPVS or ABGTS to the environment is assumed.
- No Auxiliary Building isolation is assumed.
- 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.
- The time to isolate the Control Room is increased from 20.6 seconds to 40 seconds.
- New onsite (Control Room) and offsite atmospheric dispersion factors based on more recent meteorological data (1991 through 2010) are used.

3.2 Computer Codes

Table 1 provides the computer codes used to perform the AST analysis of the FHA.

Table 1
Computer Codes Utilized in the AST Analysis of the FHA

Computer Code	Version	Purpose	Regulatory Precedence
ARCON96	June 25, 1997	Onsite Atmospheric Dispersion Factors	Safety Evaluation Report for License Amendment No. 59 to Facility Operating License No. NPF-90 for WBN, Unit 1, dated January 6, 2006
SCALE ORIGEN - S	4.3 3.0	Conventional Core Fission Product Inventory	NUREG/CR-0200, "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," September 1998 Safety Evaluation Report for License Amendment No. 293 to Renewed Facility Operating License No. NPF-6 for Arkansas Nuclear One, Unit No. 2, dated April 26, 2011
ORIGEN	2.1	Tritium Production Core Fission Product Inventory	Safety Evaluation Report for License Amendment Nos. 269 and 273 to Renewed Facility Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station, Units 2 and 3, dated September 5, 2008 Referenced in RG 1.183
STP	7	Activity Released after a FHA	Safety Evaluation Report for License Amendment No. 59 to Facility Operating License No. NPF-90 for WBN, Unit 1, dated January 6, 2006
COROD	7.1	Control Room Doses	Safety Evaluation Report for License Amendment No. 59 to Facility Operating License No. NPF-90 for WBN, Unit 1, dated January 6, 2006
FENCDOSE	5	Offsite Doses	Safety Evaluation Report for License Amendment No. 59 to Facility Operating License No. NPF-90 for WBN, Unit 1, dated January 6, 2006

The offsite X/Qs are determined utilizing a calculation methodology that conforms to RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants."

3.3 Accident Source Term

3.3.1 Fission Product Inventory

Table 2 provides the specific parameters used in the core inventory calculations.

Table 2
Parameters Used in Core Inventory Calculations

Parameter	Conventional Core	Tritium Production Core
Core Thermal Power	3459 Megawatts-thermal (MWt)	3459 MWt
Emergency Core Cooling System Uncertainty Factor	3.06%	0.6%
Number of Assemblies	193	193 (96 Once Burned, 96 Twice Burned, and 1 Thrice Burned)
Fuel Rods per Assembly	264	264
Burn-up	1500 Effective Full Power Days (EFPD)	510/1020/1530 EFPD
Enrichment	5 weight percent U-235	4.95 weight percent U-235
Core Average Assembly Power	18.47 MWt	18.03 MWt
Radial Peak to Average Ratio for Discharge Assembly	1.65	1.65

The analysis assumes that all of the fuel rods in a fuel assembly rupture. Thus, the fission product inventory of the damaged fuel assembly was determined by dividing the total core inventory by the number of fuel assemblies in the core. The values assumed for individual fission product inventories are calculated assuming full power operation at the end of core life immediately preceding shutdown with a radial peaking factor of 1.65 for the standard core assembly and TPC assembly, except tritium (discussed in Section 3.3.2). The factor of 1.65 is the maximum peaking factor allowed by the Core Operating Limit Report (COLR).

Table 3 provides the source terms for the 1500 EFPD maximum burn-up of a standard core utilized in an 18 month fuel cycle. Table 4 provides the source terms for the once burned, twice burned, and three-times burned assemblies for the Tritium Production Core (TPC).

The analysis assumes a decay time of 100 hours prior to the movement of spent fuel. The source terms presented in Tables 3 and 4 do not include this decay time, but it is accounted for in the STP model that is utilized to determine the activity released after a fuel handling accident.

The fission product inventory conforms to Regulatory Position C.3.1 of RG 1.183, except that an additional source of tritium is included.

Table 3
Source Terms for 1500 EFPD Maximum Burn-up of a Standard Core for an 18-Month Fuel Cycle

Nuclide Ci/Assembly	Nuclide Ci/Assembly	Nuclide Ci/Assembly
1 Kr-83m 5.20E+04	37 Rb-88 2.90E+05	76 Sb-127 4.69E+04
2 Kr-85m 1.04E+05	38 Rb-89 3.74E+05	77 Sb-129 1.65E+05
3 Kr-85 7.02E+03	39 Rb-90m 1.13E+05	78 Sb-130m 2.17E+05
4 Kr-87 2.06E+05	40 Rb-90 3.39E+05	79 Sb-130 5.45E+04
5 Kr-88 2.82E+05	41 Rb-91 4.65E+05	80 Sb-133 3.08E+05
6 Kr-89 3.44E+05	42 Se-84 9.25E+04	81 Te-125m 1.29E+03
7 Kr-90 3.64E+05	43 Sr-89 3.90E+05	82 Te-127m 7.91E+03
8 Xe-131m 5.64E+03	44 Sr-90 6.17E+04	83 Te-127 4.65E+04
9 Xe-133m 3.22E+04	45 Sr-91 5.06E+05	84 Te-129m 3.18E+04
10 Xe-133 9.63E+05	46 Sr-92 5.55E+05	85 Te-129 1.57E+05
11 Xe-135m 2.16E+05	47 Sr-93 6.46E+05	86 Te-131m 1.04E+05
12 Xe-135 2.90E+05	48 Sr-94 6.55E+05	87 Te-131 4.14E+05
13 Xe-137 9.15E+05	49 Y-90 6.56E+04	88 Te-132 7.06E+05
14 Xe-138 8.31E+05	50 Y-91m 2.94E+05	89 Te-133m 4.37E+05
15 Xe-139 5.99E+05	51 Y-91 5.24E+05	90 Te-133 5.35E+05
16 Xe-140 4.10E+05	52 Y-92 5.59E+05	91 Te-134 8.47E+05
	53 Y-93 4.39E+05	92 Ba-137m 8.45E+04
17 I-130 1.79E+04	54 Y-94 7.11E+05	93 Ba-139 8.66E+05
18 I-131 4.94E+05	55 Y-95 7.51E+05	94 Ba-140 8.71E+05
19 I-132 7.21E+05	56 Y-96 7.35E+05	95 Ba-141 7.80E+05
20 I-133 1.00E+06	57 Zr-95 8.05E+05	96 Ba-142 7.33E+05
21 I-134 1.10E+06	58 Zr-97 8.14E+05	97 La-140 9.43E+05
22 I-135 9.60E+05	59 Nb-95 8.11E+05	98 La-141 7.88E+05
23 I-136m 2.10E+05	60 Nb-97m 7.73E+05	99 La-142 7.64E+05
	61 Nb-97 8.20E+05	100 La-143 7.12E+05
24 Br-83 5.20E+04	62 Mo-99 9.16E+05	101 Ce-141 7.94E+05
25 Br-84m 2.63E+03	63 Tc-99m 8.06E+05	102 Ce-143 7.20E+05
26 Br-84 9.51E+04	64 Tc-99 0.00E+00	103 Ce-144 6.64E+05
27 Br-85 1.03E+05	65 Tc-101 8.48E+05	104 Ce-145 5.94E+05
28 Br-87 1.61E+05	66 Ru-103 8.48E+05	105 Pr-143 6.96E+05
	67 Ru-105 6.30E+05	106 Pr-144 6.69E+05
29 Cs-134 1.66E+05	68 Ru-106 3.85E+05	107 Pr-145 4.94E+05
30 Cs-135 0.00E+00	69 Ru-107 3.88E+05	108 Np-239 1.11E+07
31 Cs-136 4.90E+04	70 Rh-103m 8.46E+05	
32 Cs-137 8.90E+04	71 Rh-105m 1.79E+05	
33 Cs-138 9.09E+05	72 Rh-105 5.95E+05	
34 Cs-139 8.40E+05	73 Rh-106 4.08E+05	
35 Cs-140 7.52E+05	74 Rh-107 3.88E+05	
36 Cs-141 5.70E+05	75 Sn-130 1.62E+05	
Others (corrosion/activation products) Ci/Assembly		
Cr-51 1.65E+04	Fe-55 3.65E+03	Co-60m 9.82E+03
Mn-54 9.22E+02	Fe-59 2.74E+02	Ni-63 3.98E+02
Mn-56 2.41E+04	Co-58 6.07E+03	Ni-65 5.84E+02
Mn-57 3.21E+00	Co-60 6.60E+03	

Table 4 (Page 1 of 3)
Sources Terms for the Once Burned, Twice Burned, and Three Times Burned Fuel Assemblies for the TPC Core

WBN 96-Feed Equilibrium Core, End-of-cycle Operation at 3480 MWt for 510 days

		Average Assembly Inventories (Ci)			
Nuclide	Total Core Inventory	1X Burned	2X Burned	3X Burned	Core Avg.
Kr-83m	1.23E+07	7.63E+04	5.15E+04	6.13E+04	6.39E+04
Kr-85m	2.69E+07	1.69E+05	1.10E+05	1.25E+05	1.39E+05
Kr-85	8.81E+05	3.56E+03	5.54E+03	6.84E+03	4.56E+03
Kr-87	5.23E+07	3.31E+05	2.11E+05	2.36E+05	2.71E+05
Kr-88	7.38E+07	4.68E+05	2.97E+05	3.31E+05	3.82E+05
Kr-89	9.10E+07	5.81E+05	3.63E+05	3.97E+05	4.72E+05
Kr-90	9.01E+07	5.76E+05	3.59E+05	3.92E+05	4.67E+05
Xe-131m	9.54E+05	5.31E+03	4.56E+03	6.18E+03	4.94E+03
Xe-133m	5.80E+06	3.41E+04	2.60E+04	3.45E+04	3.01E+04
Xe-133	1.88E+08	1.11E+06	8.36E+05	1.09E+06	9.75E+05
Xe-135m	3.59E+07	2.08E+05	1.63E+05	2.19E+05	1.86E+05
Xe-135	4.96E+07	2.84E+05	2.30E+05	2.19E+05	2.57E+05
Xe-137	1.65E+08	9.75E+05	7.30E+05	9.51E+05	8.53E+05
Xe-138	1.59E+08	9.55E+05	6.93E+05	8.79E+05	8.24E+05
Xe-139	1.26E+08	7.57E+05	5.43E+05	6.83E+05	6.50E+05
Xe-140	8.32E+07	5.07E+05	3.54E+05	4.36E+05	4.31E+05
I-130	2.34E+06	9.02E+03	1.51E+04	3.14E+04	1.21E+04
I-131	9.01E+07	5.24E+05	4.09E+05	5.49E+05	4.67E+05
I-132	1.31E+08	7.63E+05	5.89E+05	7.87E+05	6.77E+05
I-133	1.88E+08	1.11E+06	8.35E+05	1.09E+06	9.75E+05
I-134	2.08E+08	1.23E+06	9.18E+05	1.19E+06	1.08E+06
I-135	1.76E+08	1.04E+06	7.81E+05	1.02E+06	9.09E+05
I-136m	5.05E+07	3.02E+05	2.21E+05	2.83E+05	2.62E+05
Br-83	1.23E+07	7.63E+04	5.14E+04	6.11E+04	6.38E+04
Br-84m	6.68E+05	3.86E+03	3.06E+03	4.13E+03	3.46E+03
Br-84	2.18E+07	1.37E+05	8.95E+04	1.03E+05	1.13E+05
Br-85	2.65E+07	1.67E+05	1.08E+05	1.23E+05	1.38E+05
Br-87	4.40E+07	2.79E+05	1.77E+05	1.98E+05	2.28E+05
Cs-134	1.12E+07	3.13E+04	8.41E+04	1.48E+05	5.81E+04
Cs-135	3.60E+01	1.30E-01	2.42E-01	3.48E-01	1.87E-01
Cs-136	3.67E+06	1.53E+04	2.24E+04	4.57E+04	1.90E+04
Cs-137	8.81E+06	3.34E+04	5.76E+04	7.66E+04	4.56E+04
Cs-138	1.75E+08	1.05E+06	7.67E+05	9.80E+05	9.09E+05
Cs-139	1.66E+08	9.95E+05	7.27E+05	9.28E+05	8.61E+05
Cs-140	1.49E+08	8.95E+05	6.53E+05	8.32E+05	7.74E+05
Cs-141	1.12E+08	6.75E+05	4.86E+05	6.14E+05	5.81E+05
Rb-88	7.48E+07	4.74E+05	3.02E+05	3.37E+05	3.88E+05
Rb-89	9.64E+07	6.13E+05	3.87E+05	4.29E+05	4.99E+05
Rb-90m	2.13E+07	1.33E+05	8.74E+04	1.01E+05	1.10E+05
Rb-90	9.41E+07	6.00E+05	3.76E+05	4.13E+05	4.88E+05
Rb-91	1.15E+08	7.28E+05	4.66E+05	5.24E+05	5.96E+05
Se-84	2.12E+07	1.33E+05	8.66E+04	9.93E+04	1.10E+05
Sr-89	1.02E+08	6.47E+05	4.06E+05	4.54E+05	5.26E+05
Sr-90	6.94E+06	2.78E+04	4.40E+04	5.45E+04	3.60E+04
Sr-91	1.23E+08	7.72E+05	4.99E+05	5.68E+05	6.35E+05
Sr-92	1.31E+08	8.14E+05	5.39E+05	6.27E+05	6.76E+05
Sr-93	1.45E+08	8.97E+05	6.08E+05	7.26E+05	7.53E+05
Sr-94	1.36E+08	8.39E+05	5.75E+05	6.92E+05	7.07E+05
Y-90	7.21E+06	2.89E+04	4.57E+04	5.78E+04	3.74E+04
Y-91m	7.11E+07	4.48E+05	2.90E+05	3.30E+05	3.68E+05
Y-91	1.29E+08	8.12E+05	5.23E+05	5.99E+05	6.67E+05

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Sources Terms for the Once Burned, Twice Burned, and Three Times Burned Fuel Assemblies for the TPC Core

WBN 96-Feed Equilibrium Core, End-of-cycle Operation at 3480 MWt for 510 days - continued

		Average Assembly Inventories (Ci)			
Nuclide	Total Core Inventory (Ci)	1X Burned	2X Burned	3X Burned	Core Avg.
Y-92	1.31E+08	8.17E+05	5.41E+05	6.30E+05	6.79E+05
Y-93	1.49E+08	9.16E+05	6.23E+05	7.46E+05	7.70E+05
Y-94	1.48E+08	9.07E+05	6.28E+05	7.63E+05	7.67E+05
Y-95	1.57E+08	9.54E+05	6.72E+05	8.31E+05	8.13E+05
Y-96	1.48E+08	8.99E+05	6.38E+05	7.93E+05	7.69E+05
Zr-95	1.64E+08	9.97E+05	7.05E+05	8.71E+05	8.51E+05
Zr-97	1.57E+08	9.38E+05	6.89E+05	8.83E+05	8.13E+05
Nb-95	1.66E+08	1.00E+06	7.13E+05	8.78E+05	8.59E+05
Nb-97m	1.49E+08	8.89E+05	6.53E+05	8.38E+05	7.72E+05
Nb-97	1.58E+08	9.45E+05	6.95E+05	8.92E+05	8.20E+05
Mo-99	1.68E+08	9.93E+05	7.48E+05	9.80E+05	8.72E+05
Tc-99m	1.47E+08	8.70E+05	6.55E+05	8.58E+05	7.63E+05
Tc-99	1.12E+03	4.36E+00	7.23E+00	9.15E+00	5.81E+00
Tc-101	1.54E+08	8.96E+05	6.94E+05	9.29E+05	7.95E+05
Ru-103	1.31E+08	7.25E+05	6.28E+05	8.97E+05	6.78E+05
Ru-105	8.13E+07	4.21E+05	4.19E+05	6.55E+05	4.21E+05
Ru-106	3.56E+07	1.37E+05	2.30E+05	3.49E+05	1.84E+05
Ru-107	4.33E+07	2.08E+05	2.38E+05	3.96E+05	2.24E+05
Rh-103m	1.18E+08	6.53E+05	5.66E+05	8.08E+05	6.10E+05
Rh-105m	2.28E+07	1.18E+05	1.17E+05	1.83E+05	1.18E+05
Rh-105	7.59E+07	3.92E+05	3.92E+05	5.92E+05	3.93E+05
Rh-106	3.94E+07	1.58E+05	2.49E+05	3.93E+05	2.04E+05
Rh-107	4.35E+07	2.09E+05	2.39E+05	3.98E+05	2.25E+05
Sn-130	3.14E+07	1.81E+05	1.43E+05	1.95E+05	1.63E+05
Sb-127	8.87E+06	4.88E+04	4.30E+04	6.26E+04	4.59E+04
Sb-129	2.77E+07	1.56E+05	1.30E+05	1.82E+05	1.43E+05
Sb-130m	4.16E+07	2.40E+05	1.90E+05	2.59E+05	2.16E+05
Sb-130	8.97E+06	5.04E+04	4.23E+04	5.96E+04	4.65E+04
Sb-133	5.52E+07	3.34E+05	2.38E+05	2.99E+05	2.86E+05
Te-125m	2.17E+05	7.83E+02	1.45E+03	1.95E+03	1.12E+03
Te-127m	1.15E+06	6.11E+03	5.81E+03	8.21E+03	5.97E+03
Te-127	8.77E+06	4.80E+04	4.27E+04	6.20E+04	4.54E+04
Te-129m	4.10E+06	2.32E+04	1.93E+04	2.68E+04	2.12E+04
Te-129	2.73E+07	1.54E+05	1.28E+05	1.79E+05	1.41E+05
Te-131m	1.27E+07	7.32E+04	5.87E+04	8.04E+04	6.60E+04
Te-131	8.00E+07	4.66E+05	3.62E+05	4.85E+05	4.15E+05
Te-132	1.29E+08	7.54E+05	5.80E+05	7.73E+05	6.67E+05
Te-133m	7.15E+07	4.34E+05	3.08E+05	3.82E+05	3.71E+05
Te-133	1.10E+08	6.48E+05	4.90E+05	6.43E+05	5.69E+05
Te-134	1.62E+08	9.81E+05	6.97E+05	8.70E+05	8.39E+05
Ba-137m	8.35E+06	3.17E+04	5.45E+04	7.26E+04	4.33E+04
Ba-139	1.71E+08	1.02E+06	7.49E+05	9.61E+05	8.84E+05
Ba-140	1.65E+08	9.88E+05	7.24E+05	9.30E+05	8.57E+05
Ba-141	1.55E+08	9.29E+05	6.79E+05	8.67E+05	8.05E+05
Ba-142	1.48E+08	8.92E+05	6.45E+05	8.17E+05	7.69E+05
La-140	1.69E+08	1.00E+06	7.48E+05	9.78E+05	8.77E+05
La-141	1.56E+08	9.33E+05	6.83E+05	8.71E+05	8.08E+05
La-142	1.52E+08	9.11E+05	6.61E+05	8.37E+05	7.86E+05
La-143	1.46E+08	8.85E+05	6.31E+05	7.89E+05	7.58E+05
Ce-141	1.59E+08	9.51E+05	6.94E+05	8.85E+05	8.23E+05
Ce-143	1.47E+08	8.90E+05	6.36E+05	7.96E+05	7.63E+05
Ce-144	1.17E+08	6.10E+05	6.00E+05	6.79E+05	6.06E+05
Ce-145	9.96E+07	6.00E+05	4.32E+05	5.45E+05	5.16E+05
Pr-143	1.46E+08	8.82E+05	6.30E+05	7.84E+05	7.56E+05
Pr-144	1.18E+08	6.15E+05	6.04E+05	6.85E+05	6.10E+05

Table 4 (Page 3 of 3)
Sources Terms for the Once Burned, Twice Burned, and Three Times Burned Fuel Assemblies for the TPC Core

WBN 96-Feed Equilibrium Core, End-of-cycle Operation at 3480 MWt for 510 days - continued

		Average Assembly Inventories (Ci)			
Nuclide	Total Core Inventory (Ci)	1X Burned	2X Burned	3X Burned	Core Avg.
Pr-145	9.97E+07	6.00E+05	4.32E+05	5.45E+05	5.16E+05
Np-239	1.53E+09	8.24E+06	7.61E+06	1.16E+07	7.95E+06
H-3	**2.68E+07**				

3.3.2 Tritium Inventory

WBN, Unit 1 is licensed to permit the production of tritium. Thus, in the analysis of the FHA, TVA includes a release of tritium, even though the FHA does not involve a temperature excursion that would result in boiling of the water covering the fuel assemblies. Note: RG 1.183 does not include the release of tritium as part of the analysis of the FHA, because standard plants do not include TPBARs.

TPBARs are installed in once and twice burned fuel assemblies, but they are not installed in fuel assemblies that are burned three times.

Following a FHA in the spent fuel pool, all 24 TPBARs in a TPC once or twice burned fuel assembly are assumed to break and release their tritium contents. Each TPBAR has 1.2 grams of tritium.

25% of the tritium released is assumed to be released to the environment following the FHA through evaporation of water. Tritium was assumed to evaporate at a constant rate over 2 hours.

100% of the tritium released from the TPBARs following a FHA will not be released to the environment, because the event does not involve temperatures that would result in boiling of the water covering the fuel assemblies.

The water tritium concentration is conservatively assumed to be 60 μCi/gm. At this concentration, the total tritium inventory would be 84,490 Ci. This is calculated as follows:

60μCi/gm * 372,000 gal * 3,785.4 cc/gal * 1 gm/cc * 1E-6 Ci/μCi = 84,490 Ci

25% of this value equals 21,123 Ci (84,490 Ci * 0.25)

If the temperature of the water in the spent fuel pool is maintained below the boiling point, a large fraction of the inventory will not evaporate in 2 hours. If the normal spent fuel pool cooling system is not in service, the spent fuel pool will not reach 212°F for at least 9 hours.

In the unlikely event that the spent fuel pool does boil, the boil off rate is 24,496.7 lb/hr, which is approximately 3,000 gallons/hr. Over a period of 2 hours, this would result in a total evaporation of 6,000 gallons of spent fuel pool water. This volume is less than 2%

of the total spent fuel pool water volume. Therefore, assuming that 25% of the spent fuel pool water evaporates in 2 hours or less is conservative.

For the containment analysis, assuming that 25% of the tritium is released to the environment following a FHA is conservative, because fuel movement would be terminated if any disruption in decay heat removal was experienced. Therefore, no boiling is expected to occur.

3.3.3 Release Fractions

The FHA analysis utilizes the following release fractions: I-131 = 0.08, Kr-85 = 0.10, and other noble gases and iodines = 0.05. Even though Table 3 of RG 1.183 specifies a gap activity for alkali metals of 12%, the FHA analysis assumes that no alkali metals are released, because particulates have essentially an infinite partition factor, which is consistent with Regulatory Position 3 of Appendix B of RG 1.183.

TVA confirmed the applicability of these release fractions, by ensuring that all of the fuel assemblies complied with Footnote 11 of RG 1.183 which states:

> "The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU."

Based on the previous cycles, burnups did not exceed 54 GWD/MTU except for a very small number of assemblies. The linear heat generation rates for these assemblies were much less than 6.3 kW/ft. The WBN fuel design guide is being updated to ensure that these limits are not exceeded. Thus, the use of the release fractions is appropriate for use at WBN.

The core inventory release fractions utilized in the FHA analysis conform to Regulatory Position C.3.2 of RG 1.183, Table 3 of RG 1.183, and Regulatory Positions 1.2 and 3 of Appendix B to RG 1.183.

3.3.4 Timing of Release Phases

For the FHA analysis, the release from the fuel gap and the fuel pellet are assumed to occur instantaneously with the onset of the projected damage. In addition, the releases to the environment are assumed to occur in a linear ramp manner over the duration of the event.

These assumptions conform to Regulatory Position C.3.3 of RG 1.183 and Regulatory Position 3 of Appendix B of RG 1.183.

3.3.5 Radionuclide Composition

As established in Section 3.3.3 of this enclosure, the core inventory release fractions utilized in the FHA analysis conform to Regulatory Positions C.3.2 and C.3.4 of RG 1.183, Table 3 of RG 1.183, and Regulatory Position 1.2 of Appendix B to RG 1.183.

3.3.6 Chemical Form

An overall effective decontamination factor of 200 is applied to the iodine in accordance with Regulatory Position 2 in Appendix B of RG 1.183. Thus, the chemical form of iodine was not expressly considered in the analysis. This conforms to Regulatory Position C.3.5 of RG 1.183.

3.3.7 Fuel Damage in FHA

All the fuel rods in a single fuel assembly (264 fuel rods) are assumed to be damaged in the FHA. This assumption is consistent with the current licensing basis established in Section 15.5.6 of the WBN, Unit 1 UFSAR. This conforms to Regulatory Position C.3.6 of RG 1.183 and Regulatory Position 1.1 of Appendix B to RG 1.183.

3.4 Dose Calculation Methodology

3.4.1 Offsite Dose Consequences

The offsite dose analysis of the FHA:

- Determines the TEDE for the most limiting person at the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ). The STP code is utilized to determine the activity following 100 hours of decay. The output from this code is used as input to computer code FENCDOSE to determine the offsite doses. The offsite atmospheric dispersion factors utilized in the dose analysis of the FHA are:

  Exclusion Area Boundary X/Q = 6.382E-4 sec/m^3

  Low Population Zone X/Q = 1.784E-4 sec/m^3

  This analysis conforms to the guidance of Regulatory Positions C.4.1.1, C.4.1.5 and C.4.1.6 of RG 1.183.

- Dose conversion factors (DCF) from Table 5-1 of EPA 400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, were utilized to calculate the TEDE. This total exposure DCF consists of 3 contributors, external exposure, inhalation exposure and exposure from ground deposition. These DCFs can be found in Tables 5-3, 5-4, and 5-5 of EPA 400-R-92-001. The Deep Dose Equivalent (DDE) component of this dose conversion factor utilizes the value from DOE/EH-0070, External Dose-Rate Conversion Factors for Calculation of Dose to the Public. The exposure to the committed effective dose equivalent (CEDE) component of these dose conversion factors are based on data provided in ICRP Publication 30.

  The DDE DCFs are an exception from the guidance of Regulatory Position C.4.1.4 of RG 1.183. The methodology utilized to calculate the offsite doses, including the exceptions to RG 1.183 noted above, is conservative (higher dose) when compared to using the RG 1.183 methodology. This is due to including the DCFs associated with ground deposition. (Appendix B of Enclosure 2)

- A breathing rate of 3.33E-4 m^3/s is embedded in the values in Table 5-4 of EPA 400-R-92-001 which is used to determine the overall DCF given in Table 5-1 as part of the dose conversion factor. This is an exception from the guidance of Regulatory Position C.4.1.3 of RG 1.183. The methodology utilized to calculate the offsite doses, including the exceptions to RG 1.183 noted above, is conservative (higher dose) when compared to using the RG 1.183 methodology. This is due to including the DCFs associated with ground deposition. (Appendix B of Enclosure 2)

- The analysis did not make any corrections for depletion of the effluent plume by deposition on the ground, but the DCF used does take into account the dose received from ground deposition. This conforms to the guidance of Regulatory Position C.4.1.7 of RG 1.183.

3.4.2 Control Room Dose Consequences

The Control Room dose analysis of the FHA:

- Determines the TEDE dose to the Control Room occupants due to the radioactive release associated with the FHA. The computer code COROD determines dose due to: 1) time dependent concentration of airborne activity in the Control Room; and 2) shine through the Control Room roof, Control Room ends, Auxiliary Building, Turbine Building, and Cable Spreading Room.

  The Control Room dose model includes a recirculation filter model along with filtered air intake, unfiltered air inleakage, and an exhaust path. Only one train of the CREVS is assumed to be in operation. Intake flow to the Control Room is assumed to be 3,200 cubic feet per minute (cfm) before isolation. Control room isolation occurs at 40 seconds. After isolation, the total recirculation flow rate into the Control Room is 3,600 cfm (711 cfm of pressurization flow and 2,889 cfm of recirculated flow) of filtered flow plus 51 cfm of unfiltered flow from various sources (e.g., open doors, leaky valves, etc).

  This conforms to the guidance of Regulatory Positions C.4.2.1 and C.4.2.4 of RG 1.183.

- Utilizes the same source term, transport, and release assumptions used for determining offsite doses to determine Control Room doses. This conforms to the guidance of Regulatory Position C.4.2.2 of RG 1.183.

- Utilizes the STP code to determine the activity released to the environment. The STP code output was utilized as input to the COROD code to determine the Control Room doses. This model is consistent with the current licensing basis described in Section 15.5.6 of the WBN, Unit 1 UFSAR. This conforms to the guidance of Regulatory Position C.4.2.3 of RG 1.183.

- Does not take any credit for the use of personal protective equipment or prophylactic drugs. This conforms to the guidance of Regulatory Position C.4.2.1 of RG 1.183.

- Assumes that the hypothetical maximum exposed individual was present in the Control Room for 100% of the time during the first 24 hours after the event, 60% of

the time between 1 and 4 days, and 40% of the time from 4 days to 30 days. A breathing rate of 3.33E-4 m^3/sec was utilized to calculate the CEDE. COROD utilizes Table 5-4 from EPA 400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, which has the breathing rate embedded in the DCF. With the exception of the breathing rate, these assumptions conform to the guidance of Regulatory Position C.4.2.6 of RG 1.183. The methodology utilized to calculate the control room doses is conservative (higher dose) when compared to using the RG 1.183 methodology. This is due to using the point kernal integration methodology as opposed to use of the DDE DCFs. (Appendix B to Enclosure 2)

- COROD calculates TEDE by summing up 100% of the gamma dose + 1% of the beta dose + concentration*DCF (Table 5-4 of EPA 400-R-92-001). The gamma dose is calculated using a point kernal integration method and the beta dose is calculated multiplying the average beta energy per disintegration * total concentration.

The parameters, including the atmospheric dispersion factors, assumed in the Control Room dose analysis of the FHA are provided in Table 5.

Table 5
Parameters Assumed in the Control Room Dose Analysis of the FHA

Parameter	Value	Basis
Control Room Volume	257,198 ft^3	Consistent with WBN, Unit 1 UFSAR Section 15.5.6
Control Room Intake Flow Prior to Isolation	3,200 cfm	Consistent with WBN, Unit 1 UFSAR Section 15.5.6
Control Room Makeup/Pressurization Flow	711 cfm	Consistent with WBN, Unit 1 UFSAR Section 15.5.6
Control Room Recirculation Flow	2,889 cfm	Consistent with WBN, Unit 1 UFSAR Section 15.5.6
Control Room Unfiltered Intake	51 cfm	Consistent with WBN, Unit 1 UFSAR Section 15.5.6
Control Room Filter Efficiency	95% first pass 70% second pass 0% for noble gases, and Tritium	Consistent with WBN, Unit 1 UFSAR Section 15.5.6
Control Room Isolation Time	40 seconds	Conservative Assumption – Change to WBN, Unit 1 UFSAR Section 15.5.6
Control Room Occupation Factors	0 – 24 hours – 100% 1 – 4 days – 60% 4 – 30 days – 40%	Consistent with WBN, Unit 1 UFSAR Section 15.5.6
Release Height	32.5 meters	Consistent with current licensing basis
Distance to Intake	41.4 meters	Consistent with current licensing basis
Intake Height	14.3 meters	Consistent with current licensing basis
Auxiliary Building Vent X/Q 0 – 2 hours	2.56E-3 sec/m^3	Change to WBN, Unit 1 UFSAR Table 15.5.-14
Shield Building Vent X/Q 0 – 2 hours	1.09E-3 sec/m^3	Change to WBN, Unit 1 UFSAR Table 15.5.-14

3.4.3 Meteorology Assumptions

3.4.3.1 Meteorological Data

Meteorological data over a 20-year period (1991 through 2010) were used in the development of the X/Qs used in the AST analysis of the FHA. The WBN, Unit 1 onsite meteorological measurements program is described in Section 2.3.3 of the WBN, Unit 1 UFSAR. It states:

> "The meteorological program has been developed to be consistent with the guidance given in RG 1.23 (Revision 1)..."

ARCON96 analyzes the meteorological data file used and lists the total number of hours of data processed and the number of hours of missing data in the case output. A meteorological data recovery rate may be determined from this information. The ARCON96 files present the number of hours of data processed as 175,320, and the number of missing data hours as 3,846. This yields a meteorological data recovery rate of 97.8%. Regulatory Position C.5 of RG 1.23 requires a 90% data recovery threshold for measuring and capturing meteorological data. The 97.8% valid meteorological data rate exceeds the 90% data recovery limit set forth by RG 1.23. With a data recovery rate of 97.8% and a total of 20 years of data, the contents of the meteorological data files are representative of the long-term meteorological trends at the WBN site.

3.4.3.2 Atmospheric Dispersion Factors

Section 2.3.4 of the WBN, Unit 1 UFSAR provides the current licensing basis regarding the derivation of the offsite X/Qs. The current offsite X/Qs are based on onsite meteorological data for the time period of 1974 through 1993 and a RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," calculation methodology.

Table 15.5-14 of the WBN, Unit 1 UFSAR provides the current onsite X/Qs. They are based on onsite meteorological data for the time period of 1974 through 1993 and were determined utilizing the ARCON96 computer code.

New WBN, Unit 1 onsite and offsite X/Qs are utilized in the FHA analysis. The onsite and offsite X/Qs are calculated consistent with the current licensing basis methodology, except the meteorological data was updated to reflect a more recent 20-year time period (1991 through 2010). The onsite and offsite X/Qs utilized in the dose analysis of the FHA are presented in Sections 3.4.1 and 3.4.2 of this evaluation.

The possible release pathways were considered from a FHA in either the Auxiliary Building (i.e., spent fuel pool) or containment. The most conservative pathway to the Control Room was modeled. The bounding pathway is an unfiltered release from the Auxiliary Building vent, which has the largest calculated Control Room X/Q. The EAB and LPZ X/Qs encompass all possible release points; therefore, they are bounding.

3.4.4 Acceptance Criteria

Offsite and Control Room doses must meet the guidance of Regulatory Position C.4.4 of RG 1.183 and the requirements of 10 CFR 50.67. Regulatory Position C.4.4 of RG 1.183 states:

> "The radiological criteria for the EAB, the outer boundary of the LPZ, and for the control room are in 10 CFR 50.67. These criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation, e.g., a large-break LOCA. The control room criterion applies to all accidents. For events with a higher probability of occurrence, postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6."

10 CFR 50.67(b)(2)(iii) provides the acceptance criterion for the Control Room. It states:

> "Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident."

Table 6 of RG 1.1.83 defines the EAB and LPZ dose criteria for the FHA as 6.3 rem TEDE.

3.5 Radiological Consequences – FHA Analysis

3.5.1 FHA Scenario Description

The following FHA scenarios are addressed:

1) The drop of a single fuel assembly in the spent fuel pool/Auxiliary Building with no Auxiliary Building Isolation (ABI) and with unfiltered releases through the Auxiliary Building vent.

2) The drop of a single fuel assembly in the containment. The containment is assumed to be open, and an unfiltered release occurs through the Shield Building vent for 12.7 seconds until the RBPVS is isolated. After 12.7 seconds, the remaining release occurs through the Auxiliary Building vent with no ABI and no filtration.

Case #1 above was determined to be the bounding accident, because the Atmospheric Dispersion Factors (X/Qs) for the Auxiliary Building are greater than the X/Qs for the Shield Building. As a result, no credit is taken for isolation of the RBPVS.

Sections 3.3 and 3.4 provide the assumptions regarding the accident source term and the dose calculation methodology. The assumptions are summarized in Table 6.

Table 6 (Page 1 of 2)
Parameters Utilized in the Analysis of the FHA

Parameter	AST Analysis	Basis
Delay Before Fuel Movement	100 hours	Consistent with the WBN, Unit 1 UFSAR Section 15.5.6 Consistent with proposed TS 3.9.10
Offsite and Control Room Breathing Rate	3.33E-4 m^3/sec for all time periods	This is an exception to Regulatory Positions C.4.1.3 and C.4.2.6 of RG 1.183. The breathing rate is embedded into the DCFs used for CEDE.
Dose Conversion Factors	Tables 5-1 (Offsite) and 5-4 (Control Room) of EPA 400-R-92-001	This is an exception to Regulatory Positions C.4.1.2 and C.4.1.4 of RG 1.183
Damage to Fuel Assembly	264 fuel rods (all fuel rods in one fuel assembly) are damaged	Consistent with WBN, Unit 1 UFSAR Section 15.5.6 Conforms to Regulatory Position 1.1 of Appendix B to RG 1.183
Fission Product Gap Fractions	I-131 = 0.08 Kr-85 = 0.10 Other noble gases and iodines = 0.05	Conforms to Regulatory Position C.3.2 of RG 1.183, Table 3 of RG 1.183, and Regulatory Position 1.2 of Appendix B to RG 1.183
Release Time for Gap Activity	Instantaneous	Conforms to Regulatory Position 1.2 of Appendix B to RG 1.183
Form of Iodine Activity Released to SFP	Specific form of radioiodine not considered due to use of an overall DCF of 200 for iodines	While Regulatory Position 1.3 of Appendix B to RG 1.183 specifies the chemical form of radioiodine to be released from the fuel, Regulatory Position 2 of Appendix B to RG 1.183 defines an overall DCF that should be utilized. Conforms to Regulatory Position 2 of Appendix B to RG 1.183
Water Level Above Damaged Fuel	Minimum water level of 23 feet is maintained above the damaged fuel for the duration of the event	Consistent with WBN, Unit 1 TS 3.7.13 Conforms to Regulatory Position 2 of Appendix B to RG 1.183

Table 6 (Page 2 of 2)
Parameters Utilized in the Analysis of the FHA

Parameter	AST Analysis	Basis
Decontamination Factor in SFP	Overall Iodine – 200 Noble Gases – 1 Particulates - Infinite	Conforms to Regulatory Positions 2 and 3 of Appendix B to RG 1.183
Release Time for Radioisotopes	All radioisotopes are linearly released to the environment within 2 hours	This conforms to Regulatory Positions 4.1 and 5.3 of Appendix B to RG 1.183
Filter Efficiencies in RBPVS and ABGST	No filtration by RBPVS or ABGTS assumed	Conservative assumption Conforms to Regulatory Position 4.2 and 5.4 of Appendix B to RG 1.183
Amount of Mixing of Activity in Containment or Auxiliary Building	None	Conforms to Regulatory Positions 4.3 and 5.5 of Appendix B to RG 1.183
Containment Isolation	The Containment is not isolated during fuel movement. The radiological consequences associated with an FHA in containment were determined to be bounded by an FHA in the Auxiliary Building	Conforms to Regulatory Positions 5.1, 5.2 and 5.3 of Appendix B to RG 1.183

3.5.2 Radiological Consequences

The methodology utilized by TVA to calculate the offsite and control room doses, including the exceptions to RG 1.183 noted above, is conservative (higher dose) when compared to using the RG 1.183 methodology. For the offsite doses, this is due to including the DCFs associated with ground deposition. For the control room doses, this is due to using the point kernal integration methodology as opposed to use of the DDE DCFs.

The radiological consequences are shown in Tables 7 and 8. The results for Control Room, EAB, and LPZ doses are within the appropriate acceptance criteria of 10 CFR 50.67(b)(2) and Table 6 of RG 1.183.

Table 7
Radiological Consequences for FHA in the Auxiliary Building

TEDE	Conventional Core (rem)	TPC Once Burned (rem)	TPC Twice Burned (rem)	TPC Thrice Burned (rem)	Limit (rem)
Control Room	1.015E+00	**2.869E+00**	2.602E+00	1.136E+00	5
EAB	2.383E+00	**2.834E+00**	2.268E+00	2.650E+00	6.3
LPZ (30-day)	6.660E-01	**7.923E-01**	6.339E-01	7.407E-01	6.3

Table 8
Radiological Consequences for FHA in Containment

TEDE	Conventional Core (rem)	TPC Once Burned (rem)	TPC Twice Burned (rem)	TPC Thrice Burned (rem)	Limit (rem)
Control Room	1.000E+00	**2.277E+00**	2.014E+00	1.119E+00	5
EAB	2.383E+00	**2.834E+00**	2.268E+00	2.650E+00	6.3
LPZ (30-day)	6.660E-01	**7.923E-01**	6.339E-01	7.407E-01	6.3

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met.

10 CFR 50, Appendix A, General Design Criterion (GDC) 19, "Control Room," states:

> "...Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident..."

The original WBN, Unit 1 licensing basis was established based on the whole body, thyroid, and skin dose limits of 10 CFR 100.11 as described in the WBN, Unit 1 UFSAR. As stated in RG 1.183, the applicable acceptance criterion to establish compliance with GDC 19 for facilities licensed with an AST is the 5 rem TEDE criterion of 10 CFR 50.67(b)(2)(iii). The new FHA analysis demonstrates that the WBN, Unit 1 Control Room dose complies with this 5 rem TEDE requirement.

4.2 Precedent

TVA evaluated precedent license amendment requests in which the NRC had approved implementation of the AST methodology to address FHAs. TVA identified the following precedents that were applicable, in part, to the changes TVA is proposing in this license amendment request:

- Arkansas Nuclear One, Unit No. 2 – Issuance of Amendment Re: Use of Alternate Source Term (TAC No. ME3678), dated April 26, 2011

- Peach Bottom Atomic Power Station, Units 2 and 3 - Issuance of Amendments Re: Application of Alternative Source Term Methodology (TAC Nos. MD6806 and MD6807), dated September 5, 2008

- Nine Mile Point Nuclear Station, Unit No. 2 - Issuance of Amendment Re: Implementation of Alternative Radiological Source Term (TAC No. MD5758), dated May 29, 2008

The approach used in the above license amendments regarding the application of the AST methodology to analyze the dose consequences of the FHA is similar to the proposed change to the WBN, Unit 1 licensing basis to selectively implement the AST methodology to analyze the dose consequences associated with FHAs. Sections 3.3, 3.4, and 3.5 of this enclosure describe WBN, Unit 1's compliance with the guidance of RG 1.183, including a description of any exceptions taken.

4.3 Significant Hazards Consideration

TVA is proposing to:

1. Permit selective implementation of the Alternate Source Term (AST) methodology in accordance with 10 CFR 50.67 and Regulatory Guide (RG) 1.183, "Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." Implementation of the AST methodology will be limited to the analysis of Fuel Handling Accidents (FHAs) for WBN, Unit 1;

2. Add WBN, Unit 1 Technical Specification (TS) 3.9.10 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;

3. Modify WBN, Unit 1 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;

4. Eliminate TS 3.9.4, "Containment Penetrations," and TS 3.9.8, "Reactor Building Purge Air Cleanup Units;" and

5. Modify WBN, Unit 1 TS 5.7.2.20 to incorporate the Control Room dose limit defined in 10 CFR 50.67(b)(2)(iii).

TVA concludes that these changes do not involve a significant hazards consideration. TVA's conclusion is based on its evaluation in accordance with 10 CFR 50.91(a)(1) of the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. *Does the proposed amendment involve a significant increase in the probability or consequence of an accident previously evaluated?*

   Response: No.

   The equipment affected by the proposed changes is mitigative in nature, and relied upon after an accident has been initiated. Application of the AST does not involve any physical changes to the plant design. While the operation of various systems will change as a result of these proposed changes, these systems are not accident initiators. Application of the AST is not an initiator of a design basis accident. The proposed changes to the TS, while they revise certain performance requirements, do not involve any physical modifications to the plant. As a result, the proposed changes do not affect any of the parameters or conditions that could contribute to the initiation of any accidents. As such, removal of operability requirements during the specified conditions will not significantly increase the probability of occurrence for an accident previously analyzed. Since design basis accident initiators are not being altered by adoption of the AST analysis of the FHA, the probability of an accident previously evaluated is not affected.

   The dose consequences of a FHA have been re-evaluated utilizing the AST methodology recognized by 10 CFR 50.67 and the guidance contained within Regulatory Guide 1.183. Based upon the results of this analysis, TVA has demonstrated that, with the requested changes, the dose consequences of the FHA are within the appropriate acceptance criteria of 10 CFR 50.67(b)(2) and Table 6 of RG 1.183. The AST involves quantities, isotopic composition, chemical and physical characteristics, and release timing of radioactive material for use as inputs to the dose analysis of the FHA. Selective implementation of the AST does not create any conditions that could significantly increase the consequences of any of the events being evaluated.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?*

   Response: No.

   The proposed changes would not require any new or different accidents to be postulated, since no changes are being made to the plant that would introduce any new accident causal mechanisms. This license amendment request does not impact any plant systems that are potential accident initiators. The AST methodology involves quantities, isotopic composition, chemical and physical characteristics, and release timing of radioactive material for use as inputs to the dose analysis of the FHA.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. *Does the proposed amendment involve a significant reduction in a margin of safety?*

   Response: No.

   TVA is proposing to modify the methodology for responding to a FHA. Selective implementation of the AST methodology is relevant only to the calculated dose consequences for the FHA. The radiological analysis of the FHA does not credit containment isolation, operation of the Auxiliary Building Gas Treatment System, or operation of the Reactor Building Purge Air Cleanup Units. The results of the revised dose consequences analysis demonstrate that the regulatory acceptance criteria regarding onsite and offsite doses are met for the FHA.

   In addition, the selective implementation of the AST methodology does not affect the transient behavior of non-radiological parameters (e.g., RCS pressure, Containment pressure) that are pertinent to a margin of safety.

   Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Based on the above, TVA concludes that the proposed amendments do not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusion

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

TVA determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental assessment needs to be prepared in connection with the proposed amendment.

ATTACHMENT 1

PROPOSED TS CHANGES (MARK-UPS) FOR WBN, UNIT 1

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.~~ ~~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.1~~ ~~Place and maintain containment purge and exhaust valves in closed position.~~ ~~OR~~ ~~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~~ ~~Immediately~~

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)]~~ ~~(8.41E+04 cpm)~~ ≤ 2.8E-02 μCi/cc~~[(b)]~~ (2.8E+04 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.~~

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~~
C ~~D~~. 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. AND C ~~D~~.2 Be in MODE 5.	6 hours 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	2	SR 3.3.8.~~3~~ 1	NA
		(a)	2	SR 3.3.8.~~3~~ 1	NA
2.	~~Fuel Pool Area Radiation Monitors~~ Deleted	~~(a)~~	~~2~~	~~SR 3.3.8.1~~ ~~SR 3.3.8.2~~ ~~SR 3.3.8.4~~	~~≤ 1161 mR/hr~~
3.	Containment Isolation - Phase A	Refer to LCO 3.3.2, Function 3.a., for all 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 Be in MODE 3. AND B.2 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~~ ~~Place OPERABLE ABGTS train in operation.~~ ~~OR~~ ~~C.2~~ ~~Suspend movement of irradiated fuel assemblies in the fuel handling area.~~	~~Immediately~~ ~~Immediately~~

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~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 and -0.5 inches water gauge with respect to atmospheric pressure during the post accident mode of operation at a flow rate ≥ 9300 and ≤ 9900 cfm.	18 months on a STAGGERED TEST BASIS

Deleted

3.9 REFUELING OPERATIONS

3.9.4 ~~Containment Penetrations~~

Deleted

~~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~~
~~A.~~ ~~One or more containment penetrations not in required status.~~	~~A.1~~ ~~Suspend movement of irradiated fuel assemblies within containment.~~	~~Immediately~~

Deleted

~~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~~

Deleted

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

Deleted

~~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 movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.10.1	Verify the reactor has been subcritical for ≥ 100 hours.	Prior to movement of irradiated fuel within containment

5.7 Procedures, Programs, and Manuals

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 ~~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:

Insert 5.7.2.20-1

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.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

(continued)

Insert 5.7.2.20-1

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 other accidents) for the duration of the accident.

ATTACHMENT 2

PROPOSED TS BASES CHANGES (MARK-UPS) FOR WBN, UNIT 1

B 3.3 INSTRUMENTATION

B 3.3.6 Containment Vent Isolation Instrumentation

BASES

BACKGROUND

Containment Vent Isolation (CVI) 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 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.~~

(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.

~~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.~~

~~As stated previously, provisions are provided to interconnect the ABI* and the CVI instrumentation to facilitate the actuation of an ABI* when a CVI is initiated and the actuation of a CVI when an ABI* is initiated. The following table specifies the radiation monitors and the associated actuation instrumentation that must be available when the ABI* and CVI functions are interconnected and fuel is being moved either inside containment or in the Auxiliary Building or if fuel movement is taking place in both areas:~~

~~Fuel Movement:~~	~~Required Radiation Monitors:~~	~~Required Actuation Instrumentation:~~	~~Containment Hatch:~~	~~Containment Penetrations:~~
~~Inside Containment~~	~~1-RE-90-130~~ ~~1-RE-90-131~~	~~LCO 3.3.6~~ ~~LCO 3.3.8~~	~~Closed in accordance with LCO 3.9.4~~	~~Containment open to Auxiliary Building in accordance with LCO 3.9.4~~
~~Auxiliary Building~~	~~0-RE-90-102~~ ~~0-RE-90-103~~	~~LCO 3.3.6~~ ~~LCO 3.3.8~~	~~Open or Closed~~	~~ABSCE maintained in accordance with LCO 3.3.8 and 3.7.12~~
~~Inside Containment and in Auxiliary Building~~	~~1-RE-90-130~~ ~~1-RE-90-131~~ ~~0-RE-90-102~~ ~~0-RE-90-103~~	~~LCO 3.3.6~~ ~~LCO 3.3.8~~	~~Closed in accordance with LCO 3.9.4~~	~~In accordance with LCO 3.9.4~~ ~~ABSCE maintained in accordance with LCO 3.3.8 and 3.7.12~~

~~* That portion of the ABI initiated by the Spent Fuel Pool Radiation Monitors, 0-RE-90-102 and 0-RE-90-103.~~

(continued)

BASES

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.

This AV is

The specified AV is

Table 3.3.6-1 specifies the~~-two~~ Allowable Value~~s~~ (AV~~s~~) for the Containment Purge Exhaust Radiation Monitors. ~~One AV is applicable in MODES 1, 2, 3, and 4 and the second AV is applicable during the movement of irradiated fuel assemblies inside containment when the potential for a fuel handling accident (FHA) exists. Both of these AVs are~~ based on expected concentrations for a small break LOCA, which is more restrictive than the 10 CFR 100 limits. ~~In addition, the source term for an FHA is significantly greater than the source term for a small break LOCA which would result in the containment purge monitors responding more quickly to the higher FHA source term (Ref. 5). Both of the specified AVs are~~ 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 AV~~s~~. If the setpoint does not exceed the applicable AV, 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.

(continued)

APPLICABILITY (continued)

While in MODES 5 and 6 ~~without fuel handling in progress~~, the Containment ~~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.

(continued)

BASES

ACTIONS (continued)

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 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.~~

(continued)

ACTIONS (continued)

~~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 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.

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.

(continued)

B 3.3 INSTRUMENTATION

B 3.3.8 Auxiliary Building Gas Treatment (ABGTS) Actuation Instrumentation

BASES

BACKGROUND

The ABGTS ensures that radioactive materials in the fuel building atmosphere following ~~a fuel handling accident or~~ a loss of coolant accident (LOCA) are filtered and adsorbed prior to exhausting to the environment. The system is described in the Bases for LCO 3.7.12, "Auxiliary Building Gas Treatment System." 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.

A

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

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

BASES (continued)

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.

~~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.~~

~~As stated previously, provisions are provided to interconnect the ABI* and the CVI instrumentation to facilitate the actuation of an ABI* when a CVI is initiated and the actuation of a CVI when an ABI* is initiated. The following table specifies the radiation monitors and the associated actuation instrumentation that must be available when the ABI* and CVI functions are interconnected and fuel is being moved either inside containment or in the Auxiliary Building or if fuel movement is taking place in both areas:~~

~~Fuel Movement:~~	~~Required Radiation Monitors:~~	~~Required Actuation Instrumentation:~~	~~Containment Hatch:~~	~~Containment Penetrations:~~
~~Inside Containment~~	~~1-RE-90-130~~ ~~1-RE-90-131~~	~~LCO 3.3.6~~ ~~LCO 3.3.8~~	~~Closed in accordance with LCO 3.9.4~~	~~Containment open to Auxiliary Building in accordance with LCO 3.9.4~~
~~Auxiliary Building~~	~~0-RE-90-102~~ ~~0-RE-90-103~~	~~LCO 3.3.6~~ ~~LCO 3.3.8~~	~~Open or Closed~~	~~ABSCE maintained in accordance with LCO 3.3.8 and 3.7.12~~
~~Inside Containment and in Auxiliary Building~~	~~1-RE-90-130~~ ~~1-RE-90-131~~ ~~0-RE-90-102~~ ~~0-RE-90-103~~	~~LCO 3.3.6~~ ~~LCO 3.3.8~~	~~Closed in accordance with LCO 3.9.4~~	~~In accordance with LCO 3.9.4~~ ~~ABSCE maintained in accordance with LCO 3.3.8 and 3.7.12~~

~~* That portion of the ABI initiated by the Spent Fuel Pool Radiation Monitors, 0-RE-90-102 and 0-RE-90-103.~~

(continued)

BASES (continued)

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~~

Deleted

~~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.~~

(continued)

BASES

LCO (continued)

3. Containment Phase A Isolation

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

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 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~~

(continued)

BASES

ACTIONS

~~C.1 (continued)~~

~~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.~~

C

C

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

C

Condition ~~D~~ 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.~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS

~~SR 3.3.8.1 (continued)~~

~~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.~~

~~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 setpoints shall be left consistent with the unit specific calibration procedure tolerance. 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.~~

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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~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.~~

REFERENCES

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

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 1 ECCS components and Unit 1 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 1 ECCS pump rooms, Unit 1 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. 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).

(continued)

BASES (continued)

LOCA

APPLICABLE SAFETY ANALYSES

The ABGTS design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a ~~fuel handling accident~~. ~~The analysis of the fuel handling accident, given in Reference 3, assumes that all fuel rods in an assembly are damaged.~~ The analysis of the LOCA assumes that radioactive materials leaked from the Emergency Core Cooling System (ECCS) are filtered and adsorbed by the ABGTS. The DBA analysis ~~of the fuel handling accident~~ assumes that only one train of the ABGTS is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the one remaining train of this filtration system. The amount of fission products available for release from the ABSCE is determined ~~for a fuel handling accident and~~ for a LOCA. ~~The assumptions and the analysis for a fuel handling accident follow the guidance provided in Regulatory Guide 1.25 (Ref. 5) and NUREG/CR-5009 (Ref. 11).~~ The assumptions and analysis for a LOCA follow the guidance provided in Regulatory Guide 1.4 (Ref. 6).

The ABGTS satisfies Criterion 3 of the NRC Policy Statement.

~~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.~~

~~As stated previously, provisions are provided to interconnect the ABI* and the CVI instrumentation to facilitate the actuation of an ABI* when a CVI is initiated and the actuation of a CVI when an ABI* is initiated. The following table specifies the radiation monitors and the associated actuation instrumentation that must be available when the ABI* and CVI functions are interconnected and fuel is being moved either inside containment or in the Auxiliary Building or if fuel movement is taking place in both areas:~~

~~Fuel Movement:~~	~~Required Radiation Monitors:~~	~~Required Actuation Instrumentation:~~	~~Containment Hatch:~~	~~Containment Penetrations:~~
~~Inside Containment~~	~~1-RE-90-130~~ ~~1-RE-90-131~~	~~LCO 3.3.6~~ ~~LCO 3.3.8~~	~~Closed in accordance with LCO 3.9.4~~	~~Containment open to Auxiliary Building in accordance with LCO 3.9.4~~
~~Auxiliary Building~~	~~0-RE-90-102~~ ~~0-RE-90-103~~	~~LCO 3.3.6~~ ~~LCO 3.3.8~~	~~Open or Closed~~	~~ABSCE maintained in accordance with LCO 3.3.8 and 3.7.12~~
~~Inside Containment and in Auxiliary Building~~	~~1-RE-90-130~~ ~~1-RE-90-131~~ ~~0-RE-90-102~~ ~~0-RE-90-103~~	~~LCO 3.3.6~~ ~~LCO 3.3.8~~	~~Closed in accordance with LCO 3.9.4~~	~~In accordance with LCO 3.9.4~~ ~~ABSCE maintained in accordance with LCO 3.3.8 and 3.7.12~~

~~* That portion of the ABI initiated by the Spent Fuel Pool Radiation Monitors, 0-RE-90-102 and 0-RE-90-103.~~

(continued)

BASES (continued)

LCO

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

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

a. Fan is OPERABLE;

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

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.~~

BASES (continued)

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

~~In MODE 1, 2, 3, or 4, 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.

~~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 handling accident. This does not preclude the movement of fuel assemblies to a safe position.~~

(continued)

BASES

~~ACTIONS~~

~~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.

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). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

(continued)

BASES

REFERENCES (continued)

Deleted.

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

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

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

10. Watts Bar Drawing 1-47W605-242, "Electrical Tech Spec Compliance Tables."

Deleted.

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

Insert B 3.7.13-1

1.183 (Ref. 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 within 10 CFR 50.67 (Ref. 7) and Regulatory Position C.4.4 of Regulatory Guide 1.183 (Ref. 6) for a fuel handling accident.

Insert B 3.7.13-2

6. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

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

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 (Ref. 3).

Insert B 3.7.13-1

APPLICABLE SAFETY ANALYSES

The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide ~~1.25 (Ref. 4). The resultant 2-hour thyroid dose per person at the exclusion area boundary is a small fraction of the 10 CFR 100 (Ref. 5) limits.~~

6 6

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

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1 (continued)

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.

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 15.4.5, "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."~~ Deleted.

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

Insert B. 3.7.13-2

B 3.9 REFUELING OPERATIONS

B 3.9.4 ~~Containment Penetrations~~ Deleted

~~BASES~~

~~BACKGROUND~~

~~During movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.~~

~~The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.~~

~~The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.~~

~~The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During movement of irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain capable of being closed.~~

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(continued)

~~BASES~~

~~BACKGROUND (continued)~~	~~The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted to within regulatory limits.~~ ~~The Reactor Building Purge Ventilation System operates to supply outside air into the containment for ventilation and cooling or heating, to equalize internal and external pressures, and to reduce the concentration of noble gases within containment prior to and during personnel access. The supply and exhaust lines each contain two isolation valves. Because of their large size, the 24 inch containment lower compartment purge valves are physically restricted to ≤ 50 degrees open. The Reactor Building Purge and Ventilation System valves can be opened in MODES 5 and 6, but are closed automatically by the Engineered Safety Features Actuation System (ESFAS). In MODE 6, large air exchanges are necessary to conduct refueling operations. The normal 24 inch purge system is used for this purpose. The ventilation system must be either isolated or capable of being automatically isolated upon detection of high radiation levels within containment.~~ ~~The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during fuel movements (Ref. 1). Closure by other valves or blind flanges may be used if they are similar in capability to those provided for containment isolation. These may be constructed of standard materials and may be justified on the basis of either normal analysis methods or reasonable engineering judgment (Ref. 4).~~
~~APPLICABLE SAFETY ANALYSES~~	~~During movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 2). Fuel handling accidents, analyzed in Reference 3, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.7, "Refueling Cavity Water Level," in conjunction with a minimum decay time of 100 hours prior to irradiated fuel movement with containment closure capability ensures that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. Standard Review Plan, Section 15.7.4, Rev. 1 (Ref. 3), defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values. The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).~~

(continued)

~~BASES~~

~~APPLICABLE SAFETY ANALYSES (continued)~~

~~Containment penetrations satisfy Criterion 3 of the NRC Policy Statement.~~

~~LCO~~

~~This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE Reactor Building Purge and Ventilation System penetrations, and the containment personnel airlocks. For the OPERABLE Reactor Building Purge and Ventilation System penetrations, this LCO ensures that these penetrations are isolable by the Containment Ventilation Isolation System. The OPERABILITY requirements for this LCO ensure that the automatic purge and exhaust valve closure times specified in the FSAR can be achieved and, therefore, meet the assumptions used in the safety analysis to ensure that releases through the valves are terminated, such that radiological doses are within the acceptance limit.~~

~~The containment personnel airlock doors may be open during movement of irradiated fuel in the containment provided that one door is capable of being closed in the event of a fuel handling accident and provided that ABGTS is OPERABLE in accordance with TS 3.7.12. Should a fuel handling accident occur inside containment, one personnel airlock door will be closed following an evacuation of containment. The LCO is modified by a Note allowing penetration flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative controls. Administrative controls ensure that 1) appropriate personnel are aware of the open status of the penetration flow path during movement of irradiated fuel assemblies within containment, 2) specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident, 3) penetration flow paths, penetrating the Auxiliary Building Secondary Containment Enclosure (ABSCE) boundary, are limited to less than the ABSCE breach allowance, and 4) the ABGTS is OPERABLE in accordance with TS 3.7.12. Operability of ABGTS is required to alleviate the consequences of a FHA inside containment resulting in leakage of airborne radioactive material past the open airlock or penetration flow paths prior to their closure.~~

(continued)

~~BASES (continued)~~

~~APPLICABILITY~~

~~The containment penetration requirements are applicable during movement of irradiated fuel assemblies within containment because this is when there is a potential for the limiting fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when movement of irradiated fuel assemblies within containment is not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.~~

~~ACTIONS~~

~~A.1~~

~~If the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, including the Containment Ventilation Isolation System not capable of automatic actuation when the purge and exhaust valves are open, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.~~

~~SURVEILLANCE REQUIREMENTS~~

~~SR 3.9.4.1~~

~~This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. Also the Surveillance will demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed by an OPERABLE automatic containment ventilation isolation signal.~~

~~The Surveillance is performed every 7 days during movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO. As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of significant fission product radioactivity to the environment in excess of those recommended by Standard Review Plan Section 15.7.4 (Reference 3).~~

(continued)

~~BASES~~

~~SURVEILLANCE REQUIREMENTS (continued)~~

~~SR 3.9.4.2~~

~~This Surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 18 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. LCO 3.3.6, Containment Ventilation Isolation Instrumentation requires a CHANNEL CHECK every 12 hours and a COT every 92 days to ensure the channel OPERABILITY during refueling operations. Every 18 months a CHANNEL CALIBRATION is performed. The system actuation response time is demonstrated every 18 months, during refueling, on a STAGGERED TEST BASIS. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.~~

~~REFERENCES~~

~~1. "Use of Silicone Sealant to Maintain Containment Integrity - ITS," GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.~~

~~2. Watts Bar FSAR, Section 15.4.5, "Design Basis Fuel Handling Accidents."~~

~~3. NUREG-0800, Standard Review Plan, Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents," Rev. 1, July 1981.~~

~~4. Generic Letter 88-17, "Loss of Decay Heat Removal."~~

Insert B 3.9.7-1

the limits defined in 10 CFR 50.67 (Ref. 7) and Regulatory Position C.4.4 of Regulatory Guide 1.183 (Ref. 8)

Insert B 3.9.7-2

(Regulatory Position 2 of Appendix B to Regulatory Guide 1.183 (Ref. 8)) allows an overall iodine decontamination factor of 200

Insert B 3.9.7-3

8% of the I-131, 10% of the Kr-85, and 5% of the other noble gases and iodides from the total fission product inventory in accordance with Regulatory Position 3.1 of Regulatory Guide 1.183 (Ref. 8)

Insert B 3.9.7-4

without containment closure or Auxiliary Building isolation

Insert B 3.9.7-5

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

8. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

B 3.9 REFUELING OPERATIONS

B 3.9.7 Refueling Cavity Water Level

and 8

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. ~~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~~.

Insert B 3.9.7-1

Insert B 3.9.7-2

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~~ (Refs. 2 and 8) analysis of a fuel handling accident in containment~~, as postulated by Regulatory Guide 1.25 (Ref. 1)~~. A minimum water level of 23 ft (Regulatory Position ~~C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1)~~ to be used in the accident analysis for iodine. This relates to the assumption ~~that 99~~% of the total iodine released from the pellet to cladding gap of ~~all the~~ dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain ~~10% of the total fuel rod iodine inventory (Ref. 1) except for I-131 which is assumed to be 12% (Ref. 6)~~.

99.5

Insert B 3.9.7-3

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

Insert B 3.9.7-4

in conjunction with

7 and 8

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

(Continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. Deleted. ~~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 15.4.5, "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. Deleted. ~~Malinowski, D. D., Bell, M. J., Duhn, E., and Locante, J., WCAP-7828, Radiological Consequences of a Fuel Handling Accident, December 1971.~~

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

Insert B 3.9.7-5

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B 3.9 REFUELING OPERATIONS

B 3.9.8 ~~Reactor Building Purge Air Cleanup Units~~

Deleted

~~BASES~~

~~BACKGROUND~~

~~The Reactor Building Purge Air Cleanup Units are an engineered safety feature of the Reactor Building Purge Ventilation System which is a non-safety feature ventilation system. The air cleanup units contain prefilters, HEPA filters, 2-inch thick charcoal adsorbers, housings and ductwork. Anytime fuel handling operations are being carried on inside the primary containment, either the containment ventilation will be isolated or the Reactor Building Purge air cleanup units will be OPERABLE (Ref. 1).~~

~~The Reactor Building Purge Ventilation System provides mechanical ventilation of the primary containment, the instrument room located within the containment, and the annulus. The system is designed to supply fresh air for breathing and contamination control to allow personnel access for maintenance and refueling operations. The exhaust air is filtered by the Reactor Building Purge Air Cleanup Units to limit the release of radioactivity to the environment.~~

~~The containment upper and lower compartments are purged with fresh air by the Reactor Building Purge Ventilation System before occupancy. The annulus can be purged with fresh air during reactor shutdown or at times when the annulus vacuum control system of the Emergency Gas Treatment System is shut down. The instrument room is purged with fresh air during operation of the Reactor Building Purge Ventilation System or is separately purged by the Instrument Room Purge Subsystem. All purge ventilation functions are non-safety related.~~

~~The Reactor Building Purge Ventilation System is sized to provide adequate ventilation for personnel to perform work inside the primary containment and the annulus during all normal operations. In the event of a fuel handling accident, the Reactor Building Purge Ventilation System is isolated. The Reactor Building Purge Air Cleanup Units are always available as passive inline components to perform their function immediately after a fuel handling accident to process activity contained in exhaust air before it reaches the outside environment.~~

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(continued)

~~BASES~~

~~BACKGROUND (continued)~~

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

~~APPLICABLE SAFETY ANALYSES~~

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

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

~~In addition, during movement of irradiated fuel in the Auxiliary Building when containment is open to the Auxiliary Building spaces, a high radiation signal from the spent fuel pool accident radiation monitors will initiate a CVI.~~

~~LCO~~

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

~~The plant design basis requires that when moving irradiated fuel in the Auxiliary Building and/or Containment with the Containment open to the Auxiliary Building ABSCE spaces, a signal from the spent fuel radiation monitors 0-RE-90-102 and -103 will initiate a CVI in addition to their normal function. In addition, a signal from the containment purge radiation monitors 1-RE-90-130, and -131 or other~~

(continued)

~~BASES~~

~~LCO (continued)~~

~~CVI signal will initiate that portion of the ABI normally initiated by the spent fuel pool radiation monitors. In addition, the ABGTS must remain operable if these containment penetrations are open to the Auxiliary Building during movement of irradiated fuel in side containment.~~

~~APPLICABILITY~~

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

~~ACTIONS~~

~~A.1 and A.2~~

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

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

~~B.1~~

~~With two Reactor Building Purge Air Cleanup Units inoperable, movement of irradiated fuel assemblies within containment must be suspended. This precludes the possibility of a fuel handling accident in containment with both Reactor Building Purge Air Cleanup Units inoperable. Performance of this action shall not preclude moving a component to a safe position.~~

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

~~SURVEILLANCE REQUIREMENTS~~

~~SR 3.9.8.1~~

~~The Ventilation Filter Testing Program (VFTP) encompasses the Reactor Building Purge Air Cleanup Unit filter tests in accordance with Regulatory Guide 1.52 (Ref. 3). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of~~

(continued)

~~BASES~~

~~SURVEILLANCE REQUIREMENTS~~	~~SR 3.9.8.1 (continued)~~ ~~the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.~~
~~REFERENCES~~	~~1. Watts Bar FSAR, Section 15.5.6, "Environmental Consequences of a Postulated Fuel Handling Accident."~~ ~~2. Watts Bar FSAR, Section 9.4.6, "Reactor Building Purge Ventilating System."~~ ~~3. Regulatory Guide 1.52 (Rev. 02), "Design, Testing and Maintenance Criteria for Post-Accident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light Water-Cooled Nuclear Power Plants."~~

B 3.9 REFUELING OPERATIONS

B 3.9.10 Decay Time

BASES

BACKGROUND	Section 15.5.6 of the WBN, Unit 1 UFSAR (Ref. 1) defines the assumptions of the fuel handling accident radiological analysis, including a minimum decay time for irradiated fuel assemblies prior to movement. This assumption ensures that the inventory of radioactive isotopes is at a level that supports the safety analysis assumptions. To ensure that irradiated fuel assemblies have decayed for the appropriate period of time, a limitation is established to require the reactor core to be subcritical for a time period at least equivalent to the minimum decay time assumption in the fuel handling analysis prior to allowing irradiated fuel to be moved. Given that no irradiated fuel assembly will be moved outside of the containment until the minimum decay time requirement is met, this requirement also ensures that any irradiated fuel assemblies that are moved outside of the containment meet the decay time assumption in the radiological analysis of the fuel handling accident.
APPLICABLE SAFETY ANALYSES	The radiological analysis of the fuel handling accident (Ref. 1) assumes a minimum decay time prior to movement of irradiated fuel assemblies. The requirements of LCO 3.3.7, "Control Room Emergency Ventilation System (CREVS) Actuation Instrumentation," LCO 3.7.10, "Control Room Emergency Ventilation System (CREVS)," LCO 3.7.11, "Control Room Emergency Air Temperature Control System (CREATCS)," and LCO 3.9.7, "Refueling Cavity Water Level," in conjunction with a minimum decay time of 100 hours prior to irradiated fuel movement ensures that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are within the requirements of 10 CFR 50.67 (Ref. 2) and Regulatory Position C.4.4 of Regulatory Guide 1.183 (Ref. 3). The decay time satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO	A minimum decay time of 100 hours is required prior to moving irradiated fuel assemblies within containment. This preserves an assumption in the fuel handling accident analysis (Ref. 1), and ensures that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits.
APPLICABILITY	This LCO applies during movement of irradiated fuel assemblies within the containment, since the potential for a release of fission products exist.
ACTIONS	<u>A.1</u> When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the reactor is subcritical for < 100 hours, movement of irradiated fuel assemblies within containment must be suspended. This action precludes the possibility of a fuel handling accident in containment. This action does not preclude moving a fuel assembly to a safe position. The immediate Completion Time is consistent with the required times for actions to be performed without delay and in a controlled manner.
SURVEILLANCE REQUIREMENTS	<u>SR 3.9.10.1</u> This SR verifies that the reactor has been subcritical for at least 100 hours prior to moving irradiated fuel assemblies by confirming the date and time of subcriticality. This ensures that any irradiated fuel assemblies have decayed for at least 100 hours prior to movement. The Frequency of "Prior to movement of irradiated fuel in the containment" is appropriate, because it ensures that the decay time requirement has been met just prior to moving the irradiated fuel.
REFERENCES	1. Watts Bar UFSAR, Section 15.5.6, "Environmental Consequences of a Postulated Fuel Handling Accident." 2. Title 10, Code of Federal Regulations, 10 CFR 50.67, "Accident Source Term." 3. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

ATTACHMENT 3

PROPOSED TS CHANGES (FINAL TYPED) FOR WBN, UNIT 1

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. 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

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
---NOTE--- This surveillance is only applicable to the actuation logic of the ESFAS instrumentation. ---		
SR 3.3.6.2	Perform ACTUATION LOGIC TEST.	92 days on a STAGGERED TEST BASIS
---NOTE--- This surveillance is only applicable to the master relays of the ESFAS instrumentation. ---		
SR 3.3.6.3	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 OR 18 months for Westinghouse type AR relays
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	≤ 2.8E-02 μCi/cc (2.8E+04 cpm)
4. Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.		

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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. <u>AND</u> C.2 Be in MODE 5.	6 hours 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 3.3.8.1	NA
	(a)	2	SR 3.3.8.1	NA
2. Deleted				
3. Containment Isolation - Phase A	Refer to LCO 3.3.2, Function 3.a., for all initiating functions and requirements.			

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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. <u>OR</u> Two ABGTS trains inoperable.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.12.1	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 and -0.5 inches water gauge with respect to atmospheric pressure during the post accident mode of operation at a flow rate ≥ 9300 and ≤ 9900 cfm.	18 months on a STAGGERED TEST BASIS

3.9 REFUELING OPERATIONS

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3.9 REFUELING OPERATIONS

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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 movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.10.1	Verify the reactor has been subcritical for ≥ 100 hours.	Prior to movement of irradiated fuel within containment

5.7 Procedures, Programs, and Manuals

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 other accidents) 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.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

(continued)

ATTACHMENT 4

PROPOSED TS BASES CHANGES (FINAL TYPED) FOR WBN, UNIT 1

B 3.3 INSTRUMENTATION

B 3.3.6 Containment Vent Isolation Instrumentation

BASES

BACKGROUND

Containment Vent Isolation (CVI) 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."

(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. 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)

BASES

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.

Table 3.3.6-1 specifies the Allowable Value (AV) for the Containment Purge Exhaust Radiation Monitors. This AV is based on expected concentrations for a small break LOCA, which is more restrictive than the 10 CFR 100 limits. The specified AV 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 AV. If the setpoint does not exceed the applicable AV, 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. 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.

(continued)

APPLICABILITY (continued)

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.

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.

(continued)

BASES

ACTIONS (continued)

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 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.

BASES

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.

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.

(continued)

B 3.3 INSTRUMENTATION

B 3.3.8 Auxiliary Building Gas Treatment (ABGTS) Actuation Instrumentation

BASES

BACKGROUND

The ABGTS ensures that radioactive materials in the fuel building atmosphere following a 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." The system initiates filtered exhaust of air from the fuel handling area, ECCS pump rooms, and penetration rooms automatically following receipt of 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).

(continued)

BASES (continued)

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 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 (continued)

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. Deleted

(continued)

BASES

LCO (continued)

3. Containment Phase A Isolation

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

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.

(continued)

BASES

ACTIONS (continued)

A.1

Condition A applies to the actuation logic train function from the Phase A Isolation and the manual 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.

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.

(continued)

BASES

ACTIONS

C.1 and C.2

Condition C applies when the Required Action and associated Completion Time for Condition A or B have not been met. 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

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."

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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 1 ECCS components and Unit 1 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 1 ECCS pump rooms, Unit 1 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)

BASES (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. 6).

The ABGTS satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

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

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

a. Fan is OPERABLE;

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

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.

(continued)

BASES (continued)

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.

BASES

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. 8). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

(continued)

BASES

REFERENCES (continued)

5. Deleted.

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

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

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

10. Watts Bar Drawing 1-47W605-242, "Electrical Tech Spec Compliance Tables."

11. Deleted.

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 (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 (Ref. 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 within 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 6, 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 nonconservatism, 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

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1 (continued)

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.

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 15.4.5, "Fuel Handling Accident."
4. Deleted.
5. Deleted.
6. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000.
7. Title 10, Code of Federal Regulations, 10 CFR 50.67, “Accident Source Term.”

B 3.9 REFUELING OPERATIONS

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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 (Refs. 2 and 8). 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 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 iodides 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 without containment closure or Auxiliary Building isolation, 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)

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. Deleted.

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

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

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

5. Deleted.

6. Deleted.

7. Title 10, Code of Federal Regulations, 10 CFR 50.67, “Accident Source Term.”

8. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000.

B 3.9 REFUELING OPERATIONS

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B 3.9 REFUELING OPERATIONS

B 3.9.10 Decay Time

BASES

BACKGROUND	Section 15.5.6 of the WBN, Unit 1 UFSAR (Ref. 1) defines the assumptions of the fuel handling accident radiological analysis, including a minimum decay time for irradiated fuel assemblies prior to movement. This assumption ensures that the inventory of radioactive isotopes is at a level that supports the safety analysis assumptions. To ensure that irradiated fuel assemblies have decayed for the appropriate period of time, a limitation is established to require the reactor core to be subcritical for a time period at least equivalent to the minimum decay time assumption in the fuel handling analysis prior to allowing irradiated fuel to be moved. Given that no irradiated fuel assembly will be moved outside of the containment until the minimum decay time requirement is met, this requirement also ensures that any irradiated fuel assemblies that are moved outside of the containment meet the decay time assumption in the radiological analysis of the fuel handling accident.
APPLICABLE SAFETY ANALYSES	The radiological analysis of the fuel handling accident (Ref. 1) assumes a minimum decay time prior to movement of irradiated fuel assemblies. The requirements of LCO 3.3.7, “Control Room Emergency Ventilation System (CREVS) Actuation Instrumentation,” LCO 3.7.10, “Control Room Emergency Ventilation System (CREVS),” LCO 3.7.11, “Control Room Emergency Air Temperature Control System (CREATCS),” and LCO 3.9.7, "Refueling Cavity Water Level," in conjunction with a minimum decay time of 100 hours prior to irradiated fuel movement ensures that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are within the requirements of 10 CFR 50.67 (Ref. 2) and Regulatory Position C.4.4 of Regulatory Guide 1.183 (Ref. 3). The decay time satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO	A minimum decay time of 100 hours is required prior to moving irradiated fuel assemblies within containment. This preserves an assumption in the fuel handling accident analysis (Ref. 1), and ensures that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits.
APPLICABILITY	This LCO applies during movement of irradiated fuel assemblies within the containment, since the potential for a release of fission products exist.
ACTIONS	A.1 When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the reactor is subcritical for < 100 hours, movement of irradiated fuel assemblies within containment must be suspended. This action precludes the possibility of a fuel handling accident in containment. This action does not preclude moving a fuel assembly to a safe position. The immediate Completion Time is consistent with the required times for actions to be performed without delay and in a controlled manner.
SURVEILLANCE REQUIREMENTS	SR 3.9.10.1 This SR verifies that the reactor has been subcritical for at least 100 hours prior to moving irradiated fuel assemblies by confirming the date and time of subcriticality. This ensures that any irradiated fuel assemblies have decayed for at least 100 hours prior to movement. The Frequency of “Prior to movement of irradiated fuel in the containment” is appropriate, because it ensures that the decay time requirement has been met just prior to moving the irradiated fuel.
REFERENCES	1. Watts Bar UFSAR, Section 15.5.6, "Environmental Consequences of a Postulated Fuel Handling Accident." 2. Title 10, Code of Federal Regulations, 10 CFR 50.67, “Accident Source Term.” 3. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000.